Recovery Operations in the Event of a Nuclear Accident or Radiological Emergency
RECOVERY OPERATIONS
IN THE EVENT
OF A NUCLEAR ACCIDENT OR
RADIOLOGICAL EMERGENCY
The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN  HAITI  PARAGUAY
ALBANIA  HOLY SEE  PERU
ALGERIA  HUNGARY  PHILIPPINES
ARGENTINA  ICELAND  POLAND
AUSTRALIA  INDIA  PORTUGAL
AUSTRIA  INDONESIA  QATAR
BANGLADESH  IRAN, ISLAMIC REPUBLIC OF  ROMANIA
BELGIUM  IRAQ  SAUDI ARABIA
BOLIVIA  IRELAND  SENEGAL
BRAZIL  ISRAEL  SIERRA LEONE
BULGARIA  ITALY  SINGAPORE
BYELORUSSIAN SOVIET SOCIALIST REPUBLIC  JAMAICA  SOUTH AFRICA
CAMEROON  JAPAN  SPAIN
CANADA  JORDAN  SRI LANKA
CHILE  KENYA  SUDAN
CHINA  KOREA, REPUBLIC OF  SWEDEN
COLOMBIA  KUWAIT  SWITZERLAND
COSTA RICA  LEBANON  SYRIAN ARAB REPUBLIC
COTE D’IVOIRE  LIBERIA  THAILAND
CUBA  LIBYAN ARAB JAMAHIRIYA  TUNISIA
CYPUS  LIECHTENSTEIN  TURKEY
CZECHOSLOVAKIA  LUXEMBOURG  UGANDA
DEMOCRATIC KAMPUCHEA  MADAGASCAR  UKRAINIAN SOVIET SOCIALIST REPUBLIC
DEMOCRATIC PEOPLE’S REPUBLIC OF KOREA  MALAYSIA  UNION OF SOVIET SOCIALIST REPUBLICS
DENMARK  MALI 
DENMARK  MAURITIUS
DOMINICAN REPUBLIC  MEXICO  UNITED ARAB EMIRATES
ECUADOR  MONACO  UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
EGYPT  MONGOLA  
EL SALVADOR  MOROCCO
ETHIOPIA  MYANMAR  UNITED REPUBLIC OF TANZANIA
FINLAND  NAMIBIA
FRANCE  NETHERLANDS  UNITED STATES OF AMERICA
FRANCE  NEW ZEALAND  URUGUAY
GABON  NICARAGUA  VENEZUELA
GERMAN DEMOCRATIC REPUBLIC  NIGER  VIET NAM
GERMANY, FEDERAL REPUBLIC OF  NIGERIA  YUGOSLAVIA
GHANA  NORWAY  ZAIRE
GREECE  PAKISTAN  ZAMBIA
GUATEMALA  PANAMA  ZIMBABWE

The Agency’s Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is “to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world”.

© IAEA, 1990

Permission to reproduce or translate the information contained in this publication may be obtained by writing to the International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria.

Printed by the IAEA in Austria
May 1990
RECOVERY OPERATIONS
IN THE EVENT
OF A NUCLEAR ACCIDENT OR
RADIOLOGICAL EMERGENCY

PROCEDINGS OF AN INTERNATIONAL SYMPOSIUM
ON RECOVERY OPERATIONS IN THE EVENT
OF A NUCLEAR ACCIDENT OR RADIOLOGICAL EMERGENCY
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, 6-10 NOVEMBER 1989

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1990
RECOVERY OPERATIONS IN THE EVENT
OF A NUCLEAR ACCIDENT OR RADIOPHYSICAL EMERGENCY
IAEA, VIENNA, 1990
STI/PUB/826
ISBN 92-0-020290-X
ISSN 0074-1884
FOREWORD

Much progress has been made over the last decade in the field of emergency planning and preparedness, including the development of guidance, criteria, training programmes, regulations and comprehensive plans in support of nuclear facilities. The international conventions on early notification and assistance in the event of a nuclear accident or radiological emergency are in effect and the International Atomic Energy Agency is continuing its efforts to improve and facilitate the practical implementation of these conventions in order to maximize the efficacy of these agreements.

It is well recognized that the late or recovery phase of a possible accident needs to be firmly addressed within the framework of advance emergency planning and preparedness. Several events that have occurred have not only prompted renewed interest but have served to highlight the complexities and challenging circumstances encountered after a nuclear accident or radiological emergency during the protracted period of restoring matters to normal.

To provide a forum for international review and discussion of actual experiences gained and lessons learned from the different aspects of recovery techniques and operations in response to serious accidents at nuclear facilities and accidents associated with radioactive materials, the IAEA organized the International Symposium on Recovery Operations in the Event of a Nuclear Accident or Radiological Emergency. The symposium was held from 6 to 10 November 1989 in Vienna, Austria, and was attended by over 250 experts from 35 Member States and 7 international organizations.

Although the prime focus was on on-site and off-site recovery from nuclear reactor accidents and on recovery from radiological accidents unrelated to nuclear power plants, development of emergency planning and preparedness resources was covered as well. From the experiences reported, lessons learned were identified. While further work remains to be done to improve concepts, plans, materials, communications and mechanisms to assemble quickly all the special resources needed in the event of an accident, there was general agreement that worldwide preparations to handle any possible future radiological emergencies had vastly improved.

A special feature of the symposium programme was the inclusion of a full session on an accident involving a chemical explosion in a high level waste tank at a plutonium extraction plant in the Southern Urals in the USSR in 1957. Information was presented on the radioactive release, its dissemination and deposition, the resultant radiation situation, dose estimates, health effects follow-up, and the rehabilitation of contaminated land.

This volume contains the full text of the 49 papers presented at the symposium together with a concise summary based on the deliberations of the chairmen of the scientific sessions.
The present proceedings complement those published by the IAEA in 1986 of an International Symposium on Emergency Planning and Preparedness for Nuclear Facilities.

EDITORIAL NOTE

The papers and discussions have been edited by the editorial staff of the International Atomic Energy Agency to the extent considered necessary for the reader's assistance. The views expressed and the general style adopted remain, however, the responsibility of the named authors or participants. In addition, the views are not necessarily those of the governments of the nominating Member States or of the nominating organizations.

The use in these Proceedings of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of specific companies or of their products or brand names does not imply any endorsement or recommendation on the part of the IAEA.

Authors are themselves responsible for obtaining the necessary permission to reproduce copyright material from other sources.
CONTENTS

RECOVERY OPERATIONS FROM ACCIDENTS WITH
RADIOACTIVE MATERIALS (Session I)

Aspects of the initial and recovery phases of the
radiological accident in Goiânia, Brazil (IAEA-SM-316/10) ............... 3
J.J. Rozental, C.E. de Almeida, A.H. Mendonça

Emergency response to a spill of tritiated heavy water — the interface
between emergency response, routine monitoring and research
(IAEA-SM-316/9) ........................................................................ 23
D.R. Champ, R.M. Brown, E.L. Cooper, R.J. Cornett

Decontamination of the highly contaminated sites in the Goiânia
radiological accident (IAEA-SM-316/13) ........................................ 39
L.A. Vinhas

Waste management in the Goiânia accident — the contribution of the
Waste Treatment Division of the Nuclear Technology Development
Centre (IAEA-SM-316/16) .......................................................... 49
S.T.W. Miaw, M.F.R. Guzella, L.C.A. Reis, P.O. Santos,
E.M.P. Silva, C.C.O. Tello

Operational failure and plan for the recovery of a cobalt-60 source
(IAEA-SM-316/3) ........................................................................ 57
F.A. Khan, M.A. Rab Molla, K.O. Awal, S.M.F. Karim,
M.A. Mannan, M.A.T. Ali

ON-SITE RECOVERY OPERATIONS (NUCLEAR FACILITIES)
(Session II)

The Three Mile Island Unit 2 recovery: a decade of challenge
(IAEA-SM-316/23) ........................................................................ 65
J.E. Hildebrand

United States Nuclear Regulatory Commission’s regulatory oversight
of cleanup operations at the Three Mile Island Unit 2 station
(1979–1989) (IAEA-SM-316/7) ......................................................... 79
W.D. Travers

Обеспечение радиационной безопасности при сооружении объекта
“Укрытие” на Чернобыльской АЭС
(IAEA-SM-316/40) ........................................................................ 87
A.P. Панфилов, Л.Ф. Беловодский, В.И. Гришмановский
(Radiation safety during construction of the encapsulation at the
Chernobyl nuclear power plant: A.P. Panfilov, L.F. Belovodskij,
V.I. Grishmanovskij)
Радиационный контроль при сооружении "Укрытия" четвертого энергоблока Чернобыльской АЭС

(IAEA-SM-316/38) .............................................................. 105
Л. Ф. Беловодский, И. И. Андреев, Ю. А. Болотов,
В. К. Гаевой, В. И. Гришмановский, А. П. Панфилов

Методы радиационной разведки и защиты при сооружении "Укрытия" четвертого энергоблока Чернобыльской АЭС

(IAEA-SM-316/39) .............................................................. 125
Л. Ф. Беловодский, И. А. Беляев, Л. А. Лебедев,
С. Г. Михеенко, А. А. Строганов
(Radiation reconnaissance and protection methods during construction of the encapsulation for Unit 4 of the Chernobyl nuclear power plant: L.F. Belovodskij, I.A. Belyaev, L.A. Lebedev, S.G. Mikheenko, A.A. Stroganov)

Оптимизация дозовых затрат при ликвидации последствий крупных аварий на АЭС

(IAEA-SM-316/41) .............................................................. 135
Л. Ф. Беловодский, И. А. Беляев, Л. А. Лебедев,
С. Г. Михеенко, А. А. Строганов
(Optimization of dose commitments during cleanup of major accidents at nuclear power plants: L.F. Belovodskij, I.A. Belyaev, L.A. Lebedev, S.G. Mikheenko, A.A. Stroganov)

Preparation for cleanup activities after severe accidents:
a Swedish perspective (IAEA-SM-316/50) ................................ 145
G. Hultqvist

Формирование доз облучения в период ликвидации последствий крупной радиационной аварии

(IAEA-SM-316/36) .............................................................. 151
О. А. Кочетков, В. П. Крючков, В. А. Кутьков
Л. Г. Лапа, Д. П. Осанов, В. И. Попов
(Accumulation of the exposure dose during operations to deal with the consequences of a serious radiation accident: O.A. Kochetkov, V.P. Kryuchkov, V.A. Kyt'kov, L.G. Lapa, D.P. Osanov, V.I. Popov)

Operaciones de recuperación en el emplazamiento tras una pérdida de agua pesada de un canal de refrigeración de un sistema presurizado (IAEA-SM-316/45) ........................................... 167
O.E. Agatiello
OFF-SITE RECOVERY OPERATIONS (NUCLEAR FACILITIES)
(Session III)

Некоторые аспекты послеаварийных работ в контролируемой зоне Чернобыльской АЭС

(IAEA-SM-316/42) ....................................................... 185
С.Т. Беляев, А.А. Боровой, Ю.П. Бузулуков, Ю.Л. Добрынин, А.Ю. Гагаринский, Н.Н. Долгин, Ю.И. Сульдин, А.Д. Шрамченко

Use of dose assessment models to facilitate off-site recovery operations for accidents at nuclear facilities (IAEA-SM-316/4) ................................................. 203
M.H. Dickerson, K.T. Foster

TACTUS: a code for simulation of the flow of caesium-137 in urban surroundings (IAEA-SM-316/48) ...................................................... 217
K.G. Andersson

Peculiarities of nuclear fuel inside the confinement system of the damaged reactor at Chernobyl (IAEA-SM-316/60) ................................................. 229
S.T. Belyaev, A.A. Borovoj, A.Yu. Gagarinskij

Из опыта работы правительства УССР по ликвидации последствий аварии на Чернобыльской АЭС

(IAEA-SM-316/56) .................................................................. 235
Е.В. Качаловский
(Experience of the Government of the Ukrainian SSR in cleaning up the accident at the Chernobyl nuclear power plant: E.V. Kachalovskij)

L’organisation nationale française en cas d’urgence radiologique

(IAEA-SM-316/25) ................................................................. 249
Y. Mourès

Effect of administering stable iodine to the Warsaw population to reduce thyroid content of iodine-131 after the Chernobyl accident

(IAEA-SM-316/19) ................................................................. 257
P. Krajewski

Planning and emergency measures established in Turkey after the Chernobyl accident (IAEA-SM-316/6) ...................................................... 273
Ö.A. Soyberk

Cleanup of areas contaminated as a result of a nuclear accident

(IAEA-SM-316/12) ................................................................. 283
M.A. Feraday
Uso del análisis costo-beneficio en la toma de decisión durante la etapa de recuperación de accidentes nucleares (IAEA-SM-316/46) 301

E. Palacios, H. Bruno, J.J. Kunst

Radiation protection standards and monitoring in the event of an accident at a nuclear facility (IAEA-SM-316/37) 305
G.M. Avetisov, L.A. Buldakov, O.A. Kochetkov, D.P. Osanov, A.V. Barabanova

Recovery operations after the Chernobyl accident: The intervention criteria of the USSR’s National Commission on Radiation Protection (IAEA-SM-316/57) 313
A.J. Gonzalez

Les mesures radiologiques post-accidentelles: Carte de la contamination (IAEA-SM-316/28) 347
J. Cortella, C. Bourgeois, R. Chastel, A. Rosenberg

Decontamination of urban areas after nuclear accidents (IAEA-SM-316/44) 355
H. de Witt, W. Goldammer, H.D. Brenk, R. Hille, H. Jacobs, K. Frenkler

Décontamination des voies routières par un véhicule spécialisé (IAEA-SM-316/29) 365
R. Gros, C. Le Drean, L. Roux

THE ACCIDENT IN THE SOUTHERN URALS IN 1957
(Special Session)

THE ACCIDENT IN THE SOUTHERN URALS IN 1957
(Special Session)

Radiony cia zavar na Yu johnom Urale v 1957 godu i likvidatsiya ee posledstviy (IAEA-SM-316/55) 373


Formation of dose burdens to man and the environment: Long term management of an area subjected to radioactive contamination as a result of a radiation accident beyond the design basis (IAEA-SM-316/55-1) 405
G.N. Romanov
LESSONS LEARNED FROM RADIOLOGICAL ACCIDENTS
(Session IV)

International guidance on off-site post-accident assessment and recovery operations: development and state of the art
(IAEA-SM-316/24) ................................................................. 441
S. Chakraborty, Y.G. Gonen

The psychological impact of the radiological accident in Goiânia
(IAEA-SM-316/18) ................................................................. 463
A.B. Carvalho

The role of the United States Department of Energy in recovering from the Three Mile Island accident (IAEA-SM-316/1) ......................... 479

Le programme de recherche RESSAC (IAEA-SM-316/32) .................. 495
A. L'Homme, N. Parmentier, B. Legrand, P. Fache

Premiers résultats expérimentaux du programme RESSAC sur les essais in situ de décontamination/fixation et études de migration des radionucléides dans les sols (IAEA-SM-316/33) ......................... 507
B. Legrand, P. Fache, M. Hamoniaux, H. Camus, D. Gauthier

Lessons learned during the recovery operations in the Ciudad Juárez accident (IAEA-SM-316/53) .................................................. 517
G. Molina

The technical history of the Three Mile Island Unit 2 cleanup: Factors, options and decisions (IAEA-SM-316/22) ....................................... 525
W.C. Holton, C.A. Negin, R.W. Lambert

The Nuclear Emergency Service Company in the Federal Republic of Germany (IAEA-SM-316/43) .................................................. 541
G. Brudermüller, W. Neumann
The new approach of the radiological emergency response team at the Brazilian national Nuclear Energy Commission's Institute of Radiation Protection and Dosimetry after the Goiânia accident (IAEA-SM-316/14) ................................................................. 553
C.A. Nogueira de Oliveira, J.C.A. Récio, J. Hunt, I.A. Sachett

Intervention télécommandée en cas d'accident nucléaire (IAEA-SM-316/31) ................................................................. 559
J. Couture, B. Noc, A. Khairallah, P. Vesseron

Le plan post-accidental relatif à un accident à caractère radiologique (IAEA-SM-316/35) ................................................................. 567
M. Genesco

The radiological accident in San Salvador (IAEA-SM-316/54) ............ 575
J.R. Crofi, P. Zuniga-Bello, A. Kenneke

Applying lessons learned from a variety of events to the management of radiological recovery operations in the USA (IAEA-SM-316/8) .... 585
B.H. Weiss, G.G. Zech

Lessons learned from the radiological accident in Goiânia (IAEA-SM-316/17) ................................................................. 593
J.J. Rozental, C.E. de Almeida, J.J. Labone}

The contribution of the World Health Organization to international co-operation in medical preparedness for radiation emergencies (IAEA-SM-316/49) ................................................................. 599
I. Riaboukhine

The influence of seasonality on accident consequences and emergency response planning (IAEA-SM-316/34) ................................................................. 607
C. Viktorsson, G. Boeri

Panel ................................................................. 625

Chairmen of Sessions ................................................................. 635

Secretariat of the Symposium ................................................................. 635
List of Participants ................................................................. 637
Author Index ................................................................. 653
Transliteration Index ................................................................. 655
Index of Papers by Number ................................................................. 657
RECOVERY OPERATIONS FROM ACCIDENTS
WITH RADIOACTIVE MATERIALS

(Session I)

Chairman

R. CIBRIAN
Commission of the European Communities
Invited Paper

ASPECTS OF THE INITIAL AND RECOVERY PHASES OF THE RADIOLOGICAL ACCIDENT IN GOIÂNIA, BRAZIL

J.J. ROZENTAL, C.E. DE ALMEIDA, A.H. MENDONÇA
Comissão Nacional de Energia Nuclear, Rio de Janeiro, Brazil

Abstract

ASPECTS OF THE INITIAL AND RECOVERY PHASES OF THE RADIOLOGICAL ACCIDENT IN GOIÂNIA, BRAZIL.

In September 1987, the removal of the rotating assembly of the shielding head of a teletherapy unit and the dismantling of the capsule containing 50.9 TBq (1375 Ci) of $^{137}$Cs resulted in a widespread contamination of central Goiânia, a Brazilian city of one million inhabitants. Notwithstanding the recommendations contained in publications concerning emergency response planning and preparedness, this radiological accident showed that several adverse vectors not mentioned in the literature were a reality. Thus, not only social, political, economic and technical problems had to be faced but also psychological aspects had to be dealt with. Of these, discrimination against the victims was the most important. The paper draws attention to several aspects of the actions that were necessary to minimize the exposure of the population to external radiation and discusses the initial procedures undertaken by the National Nuclear Energy Commission (CNEN) in Goiânia.

1. ACKNOWLEDGEMENT OF THE STATE OF EMERGENCY

The National Nuclear Energy Commission (CNEN) learned about the radiological accident in Goiânia from a telephone call to the Department of Nuclear Installations (DIN) made by physicist W.F. at 3.00 p.m. on 1987-9-29 at the request of the Secretary of Health of the State of Goiás. The DIN was informed that several persons were at the Hospital for Tropical Diseases or were camped down in tents at the Olympic Stadium and presented symptoms characteristic of radiation syndrome.

The description of the facts and the collection of data regarding the radioactive installations in Goiânia suggested that the occurrence had resulted from the loss of the sealing protection of radioactive material coming from some radiotherapy equipment. After consulting its records, the DIN telephoned the Goiânia Institute for
Radiotherapy (IGR) and was informed by one of its owners, a radiotherapy physician and the physicist in charge of radiation protection, that the occurrence was related, possibly, to a source of $^{137}$Cs from its Institute.

The CNEN arranged for its Institutes and Headquarters to provide immediate assistance, and at 6.00 p.m. that day, the director of DIN and technicians of the Institute for Nuclear and Energy Research (IPEN) left for Goiânia to make an initial assessment of the situation. They arrived in Goiânia at half past midnight on 1987–9–30.

**FIG. 1. Principal sites of contamination.**
Together with the directors of IGR, they went to 1587 Paranaiba Street, where the source was presumed to be. The site was found to be in a state of semi-destruction, unguarded, and with its electric power turned off. After illumination had been improvised, it was found that the $^{137}$Cs source was not there and the exposure rates were about the same as from natural radiation. More detailed investigation of the site was carried out later, in daylight.

Next, the group, accompanied by members of the Civil Defence, went to the sites presumably contaminated, where the situation was found to be serious.

At dawn of the same day, contact was made with the Executive Director I of CNEN and the Director of the Institute of Radiation Protection and Dosimetry (IRD), who dispatched teams to Goiânia immediately with the following duties:

- radiological protection
- environmental control
- radioactive waste management and
- medical aid.

The preliminary evaluation indicated that some individuals might require specialized treatment. Therefore, the Marcilio Dias Naval Hospital was placed on the alert.

At daybreak of 30 September, the principal foci (Fig. 1) were isolated. They were:

- Rua 57 (57th street), house No. 58
  Downtown District (Setor Central)
- Rua 17-A, block No. 70, lot 26-B
  Airport District (Setor Aeroporto)
- Rua 6, block Q, lot 18
  North Railroad District (Setor Norte Ferroviário)
- Rua P-19, block 92, lot 4
  Civil Service Workers District (Setor dos Funcionários)
- Rua-16-A, No. 792
  Public Hygiene Control Unit (Vigilância Sanitária)
- Rua 63, house No. 179
  Downtown District
- Rua 26-A, block 2, lot 30
  Airport District.

Because of the high exposure rates in the neighbourhood of the Public Hygiene Control building — occasioned by the $^{137}$Cs source contained in the metal part that had been brought there previously — it was determined that the part should be shielded with maximum urgency.

The values measured on this occasion were higher than 10 Sv/h near the surface and about 0.4 Sv/h one metre from the source. On 1987-10-1, in conjunction
with the Departments of Health and of Transport of the State of Goiás, steps were taken to shield the source of $^{137}$Cs with concrete.

The priorities established for the tasks to be undertaken beginning on the afternoon of 30 September were:

— medical care for the contaminated persons
— washing and changing of the clothing of the individuals contaminated
— provision of food
— monitoring of people
— informing the public.

A technical meeting of the task group in Goiânia was held on the evening of September 30, when it was decided that these teams should join forces to make it possible to provide:

— a reconstruction of the accident as accurately as possible;
— identification of all the foci of contamination;
— monitoring of the people at the Olympic Stadium;
— radiometric surveys;
— environmental monitoring in a joint effort with the Environmental Department of Goiás (SGMAGO);
— care for the victims at the General Hospital of Goiânia (HGG), the transfer of six patients to the Naval Hospital, and the inspection of the screening of persons presumably irradiated and/or contaminated;
— accurate information to the public by means of the Health Organization of the State of Goiás (OSEGO);
— aeroradiometric tracing of the city of Goiânia.

The severity of the clinical picture of the patients, the alarming information about the contamination of the water, and the number of areas affected made for a great confluence of the population at the Olympic Stadium.

2. SCREENING AND CARE OF PEOPLE

The persons who had handled the source or part of it and consequently presented a higher level of irradiation had been evaluated clinically and, with the diagnosis being uncertain, had been admitted to the Hospital for Tropical Diseases (HDT) and to the Santa Maria Hospital. The Olympic Stadium was chosen by OSEGO to shelter the persons vacated from their homes in light tents. In the main foci, the people were evacuated for clinical and radiometric treatments and according to the seriousness of the findings they were sent to the HGG, the State Foundation for the Well-being of Minors (FEBEM), or to the House of the Good Shepherd (Albergue Bom Samaritano).
The persons who lived in places adjacent to the identified foci of contamination or who had had some type of contact with the victims were instructed to go to the Olympic Stadium, where they would be monitored. The levels of contamination that they presented were considerably lower than those of the preceding group.

In addition to these people, 112,000 members of Goiânia’s population reported to the Olympic Stadium to be monitored even though their direct relation to the incident had not been established.

The reason for their coming to the stadium was apparently that they happened to be in the city at that time.

The purpose of the monitoring was:

— to identify any contamination
— to apply preliminary decontamination measures as well as to evaluate the efficiency of the procedure
— to refer cases of persisting contamination to a team of specialists for medical follow-up.

In this phase of first aid, contamination was found in 249 persons. Among these, 120 presented contamination of clothing and shoes only; 129 presented external and/or internal contamination and were, therefore, placed under direct medical supervision.

Within the first few post-accident days, 50 other persons were referred to the medical team for clinical and laboratory evaluation.

Among these were some from the Public Hygiene Control Unit, the Military Police, the Fire Department, or from among the close relatives of the victims. Of a total of 100 persons, 21 required hospitalization for intensive medical treatment for their haematological condition and radiodermatitis. Ten of these patients were in a serious condition; four of these died and one underwent amputation of the forearm.

The initial screening undertaken by the State government and the procedure adopted by the CNEN eliminated the possibility of contamination and exposure of persons while they waited to be monitored.

Simultaneously with the technical work being done, professionals from the field of social services gave explanations and support to the public to assuage the anxiety of the population, even though contamination had been found to be restricted to one group of individuals, confirming the initial working hypothesis of the task force.

3. THE SEARCH TO ISOLATE THE CONTAMINATION

After the first day of activity in Goiânia, the technical teams established that the dissemination of $^{137}$Cs could be attributed principally to the following factors:

— social contacts maintained by persons directly contaminated through the inappropriate handling of radioactive material
FIG. 2. Principal sites of contamination and sampling points.
— commercial use of contaminated material that came from the junkyards involved
— distribution of fragments of the radioactive source
— dispersion by wind and rain
— The interval of time between splitting open the capsule containing $^{137}$Cs on 1987–9–13 and the wider disseminations of the material.

Smaller foci and less contaminated persons were identified in distant regions. This led the CNEN to add aeroradiometric surveying to the inland search that was being conducted.

3.1. Land search

The identification of the contaminated material proceeded according to the following scheme:

— systematic tracking of exposure rates higher than those of natural background radiation;
— on-site verification of information furnished by contaminated persons, their family, friends, and the population.

Seven main foci of contamination were immediately identified and the respective areas isolated. The radiological survey in one of the foci is shown in Fig. 2.

Two weeks later, in October 1987, other sites, some of them located outside the region of Goiânia, had been identified, presenting, however, only a slight amount of radioactive contamination. These sites all underwent initial decontamination, and the resulting radioactive waste was treated and removed to the Olympic Stadium temporarily.

3.2. Air search

On 7–8 October, tracing of the urban area of Goiânia, including complexes at its outskirts, the campus of the Federal University of Goiás (UFGO) and the Autodromo (auto race track), resulted in the finding of only one new focus of contamination in addition to those already known. The Meia Ponte River was examined from north, near the university campus, to south, the auto race track, without any trace of contamination above background being detected.

This survey, performed in two days, ascertained that the $^{137}$Cs contamination was concentrated chiefly in the foci isolated and already under control, confirming the initial task team’s hypotheses as to the extent of the mishap.
3.3. Other investigations

In addition, points of residual contamination were identified in 42 residences, which may be divided into two groups for the sake of simplicity:

- 20 residences that neighboured on the principal foci and had to be vacated temporarily;
- 22 residences that were distant from these foci.

In the first group, exposure of the persons concerned occurred during the period 13–29 September, at which time they were removed from their residences.

In the second group, in the residences of relatives, friends, or other persons connected with the contaminated homes, the average exposure rate was from 0.1 mR/h to not more than 1 mR/h at any site.

The transit of contaminated persons or part of the source resulted in the contamination of public places at residual levels. A programme for the decontamination of these places was immediately activated.

The lapse of time between the occurrence of the accident and its becoming known, with the persons externally contaminated being identified, indicated the necessity for immediately controlling the money in circulation in Goiânia.

A systematic check-up of the money and local bank agencies was made. Among the 10 240 000 bills monitored, only 68 (0.00066%) were found to be contaminated by $^{137}$Cs.

As other bills in contact with those contaminated presented no signs of contamination, with no signs of transference being found when smear tests were performed, it became evident that the contamination was not readily transferred by the simple handling of these bills. Furthermore, the external exposure that might supervene from their being carried or handled would be, for the most part, insignificant (3 $\mu$Sv/h). The contaminated bills were of course withdrawn from circulation.

4. ENVIRONMENTAL EVALUATION

4.1. The behaviour of caesium

On coming into contact with water courses, caesium chloride in its soluble form is rapidly and strongly retained by the sediment lying at the bottom and by particles in suspension, which then become its chief means of transport. In its airborne dispersion, caesium is deposited on the surface of the soil and plants and can be absorbed by the latter through their roots, leaves and other exposed parts.

Owing to the characteristics of the accident and the affected sites in Goiânia, the chief potential means by which the population in the neighbourhood of the foci
could be exposed to radiation were inhalation, ingestion (fruit and vegetables) and external irradiation.

To evaluate the need for operational measures such as the removal of soil, a ban on the consumption of certain foods or the pruning of trees, $^{137}$Cs concentration limits more restrictive than those recommended internationally for radiological emergency situations were used. The methodology for estimating the doses coming from different potential pathways was immediately established.

The intervention level established was aimed at keeping the annual dose from exceeding 300 mrem (3 mSv).

By the beginning of October, samples had been collected from the environment and sent to the IRD for analysis, while other steps were taken to establish a laboratory where measuring could be done in Goiânia.

Contact was made with the State Waterworks Department of Goiás to obtain the hydrological information necessary for characterizing the region.

With the participation of SEMAGO, it was possible to plan the steps necessary for making a preliminary assessment of the environment, including the rain water, sewer, and potable water systems; groundwater; foods; and evaluation of the environmental dose levels.

4.2. Potable water supply system

Samples were taken from the waters of the João Leite River at the point where the city’s water supply is drawn off, situated on the bank opposite the Meia Ponte River, and from different points in the drinking water supply system, as well as from the water and mud of the filter bed of the filtration plant. The radiometric results indicated that the potable water of the city of Goiânia was not contaminated by $^{137}$Cs. The detection limit was 1.5 Bq/L.

4.3. Sewage and rainwater systems

By studying the hydrographic network of the region affected, it was found that its contamination by $^{137}$Cs could occur first through the emptying of the sewage and rain water into the Capim Puba Creek, later affecting sections of the Botafogo Creek, the Anicuns Stream, and the Meia Ponte River.

Consequently, with the co-operation of technicians of the State Technology and Basic Sanitation Company of São Paulo (CETESB) careful tracing was carried out of the rainwater galleries and sewer system, consistently downstream from the areas whose surfaces had been found to be contaminated.

It was concluded that the concentrations of $^{137}$Cs measured with a sodium iodide detector in the sewer and rain water were due particularly to the waters coming from 57th, 63rd, and 26-A Streets. However, from the point of view of radiological safety, the values found did not present any risk to the population.
4.4. Analysis of sediments

The results obtained from the samples collected from the sediment of the bed of the Meia Ponte River indicated $^{137}$Cs concentration values of 100 to 800 Bq/kg in the sections downstream from the mouth of the Anicuns Stream. This observation was confirmed by the tracing done with a sodium iodide probe by the staff of the Nuclear Development and Technology Centre/Brazilian Nuclear Enterprises Inc. (CDTN/NUCLEBRAS).

The results obtained after this period showed a rapid decrease in the concentration in the sediment, indicating that the exposures due to $^{137}$Cs were within the typical limits of variation of natural exposure.

4.5. Study of the groundwater

As the opening of the source and its initial handling took place in non-cemented areas, rainfall in the region favoured the penetration of $^{137}$Cs into the soil despite its argillaceous composition.

In view of this, measurements were made of the:

— concentration of $^{137}$Cs in soils at different depths (soil profiling)
— concentration of $^{137}$Cs in the water of wells near the foci of contamination.

Preparation of the soil profile was first entrusted to the School of Agriculture of the Federal University of Goiás (UFGO) and later IPEN. The results obtained from the samples indicated that the major part of the $^{137}$Cs, until 30 October, was concentrated in the top 20 centimetres, this pattern being similar in all the areas studied.

4.6. Measurement of the environmental dose rate

To furnish data regarding the environmental dose rate to furnish data regarding the environmental exposure characteristics for the city of Goiânia, various measuring stations equipped with thermoluminescent dosimeters (TLDs) were installed in these places:

— OSEGO
— SEMAGOG headquarters
— Zoological Garden
— Jardim das Palmeiras Cemetery
— Water reservoir of SEMAGO.

To evaluate the rate of exposure in the areas close to the foci of contamination, TLD stations were installed within a radius of 180 metres from each focus and in nearby residences. Analyses were planned to be made every three months.
4.7. Rain and aerosols

Eleven stations for the collecting of the complete deposition of rainwater and dust were installed in the Airport District. Caesium-137 was not detected in any of the samples collected.

4.8. Food products

Samples of the food products sold in the markets near the areas isolated as well as in the region that produced them, were collected. The presence of $^{137}\text{Cs}$ was not detected in any of them.

In the areas close to the sites most affected, 216 samples of vegetables, and also fruit and greens cultivated in home gardens from within a radius of 180 metres, were analysed. The results showed that the farther the plants were from the foci of contamination the less $^{137}\text{Cs}$ activity there was.

Based on the limits derived for $^{137}\text{Cs}$ in vegetables and fruits, as a precaution it was recommended that the fruit and greens be removed and that some trees that could present additional risk to the population be pruned. All the vegetables and greens were gathered then, and trees were isolated. A few trees whose levels of radioactivity could be of concern to the population were removed. Only in a few cases did these levels exceed the values allowed by the national and international regulations. Finally, a long term environmental follow-up programme was established in conjunction with SEMAGO.

5. IDENTIFICATION AND CLASSIFICATION OF WASTE RESULTING FROM THE ACCIDENT

The waste generated during the process of decontaminating persons and sites was classified according to the categories established in the experimental norm: ‘Management of Radioactive Waste in Radioactive Installations’ approved by CNEN resolution 19/85 of 11/27/85 — Official Government Report D.O.U. 12/17/85:

(a) *Non-radioactive solids*
    Those that presented specific activity below 74 kBq/kg of material.

(b) *Solids with low level radiation*
    Those that presented an exposure rate less than or equal to 2 mSv/h on the surface of the package.

(c) *Solids with medium level radiation*
    Those that presented exposure rates from 2 mSv/h up to 20 mSv/h on the surface of the package.

Liquid waste was solidified with cement and classified as solid waste, obeying the criteria described above.
5.1. Packaging of waste

The packaging containers used were industrial carbon steel drums (barrel part, bottom, and lid of 18 gauge, of 40 litres, 100 litres, and 200 litres capacity) and metal chests. The 40 litre and 100 litre drums were normally encapsulated in 200 litre drums or in metal cases.

6. TRANSPORTATION OF THE WASTE

The transportation of the radioactive waste was effected according to the Presidential Decrees No. 2063 of 1983–10–6 and No. 88821 of 1983–10–6, adopted by the CNEN and dealing with highway transportation of dangerous freight and products; the CNEN Resolution 5/81 of 1981–7–27, which adopts the International Atomic Energy Agency regulations concerning the safe transportation of radioactive material; and, wherever pertinent, CNEN Norm 06/77, ‘The Physical Protection of Operational Units in the Nuclear Area’ of 1981–7–17.

Once closed, these packages were inspected so as to determine any residual contamination of their exterior and marked accordingly after monitoring to determine surface exposure rates and those at one metre distance. Later, they were dispatched in trucks, transported under ‘completely loaded’ conditions, that is, the vehicles were exclusively for this use. The drivers and their helpers were duly instructed as to the obligatory procedures both in normal and emergency situations.

To prevent an accident or to be able to act in the event of one, the following measures were taken:

— the vehicles proceeded in convoys accompanied by a police escort
— radiation protection technicians accompanied the vehicles on their route with instruments and auxiliary material
— a system of communication by radio was maintained to keep the central station informed about the various stages en route
— the speed of the vehicles was restricted to 20 km/h in cities and 40 km/h on highways.

The measures stipulated for possible emergency situations during transportation were later put into practice when a vehicle loaded with four cases containing radioactive waste toppled over. The radiological protection team accompanying the transportation operations immediately isolated the spot with the aid of the escort and, with the assistance of a rescue team, restored normal conditions for the transportation to proceed.

After unloading at the repository, the trucks were completely monitored to determine whether they could be released into immediate circulation or not. If there was contamination, the vehicle was decontaminated until the level permitted for conventional transportation was attained.
7. TEMPORARY REPOSITORY

The waste generated during the initial process of decontamination made it imperative to determine a site for storage outside the area affected in order that the local levels of radiation not be increased by the accumulation of packages stored in the isolated areas.

The matter was discussed with the State government, which indicated two alternative areas, on the understanding that this storage would be of a temporary nature. Considerable time was taken to reach a feasible solution, and this affected the effectiveness of the clean-up operation.

The area selected is of about 2 hectares (almost 5 acres) located 20 km from the centre of Goiânia and 2.5 km from the city of Abadia de Goiás.

The technical group co-ordinated by CNEN, working jointly with the Department of Transport of Goiás, prepared an engineering project for the construction of nine concrete platforms, $60 \times 18 \times 0.2$ m where the packages would be placed. During construction, aspects that guaranteed the quality of drainage, physical safety, illumination, sanitary installations, and access for vehicles were taken into consideration. To ensure the safety of the workers and preservation of the environment, occupational and environmental radiological protection programmes were implemented. To achieve this, samples of the soil, vegetation, sediment, surface water, rain water and aerosols were collected. In addition, five TLD stations were installed along the fence that delimited the areas, and a well was sunk to collect underground water as part of the routine programme of monitoring the environment.

8. INVENTORY OF THE SOURCE

An inventory of the source was carried out (Heilbron, 1988), the starting point being fundamentally the contents of the chests of waste collected in the city of Goiânia, since the dispersion of radioactive material was restricted to few sites and the material was handled by a group of persons who had been identified. The best estimate for the source recovery was 1200 Ci or 44 TBq.

The results of the extensive environmental monitoring formed the primary basis for the assurance that there was no significant residual hazard.

9. TASK TEAMS AND TECHNICAL AID

The CNEN was able to co-ordinate the work of several members of their professional staff, in the areas of radiological protection, waste, decontamination and environmental monitoring and for providing co-ordination and maintenance. In addition, professionals from FURNAS, NUCLEBRAS, the Centro de Desenvolvimento
da Tecnologia Nuclear (CDTN), NUCLEÍ, the Ministry of the Navy, and EsIEX (Special Army Division) were added to the working team.

Teams at the headquarters and institutes of the CNEN provided continuous support to the activities in Goiânia by effecting:

- analyses of the faeces and urine of the patients hospitalized in Rio de Janeiro and Goiânia;
- cytogenetic dosimetry of the persons who suffered major exposure and contamination;
- evaluation of internal contamination, utilizing whole body counters (for both technicians and patients);
- calculations of the dose of internal contaminations in the patients, using data obtained from the analyses of excreta;
- analyses of environmental samples;
- radiation protection at the Marcilio Dias Hospital;
- the preparation of radioactive standards;
- calibration of equipment and maintenance;
- preparation of material for decontamination;
- management of the waste from the hospital;
- manufacture of special equipment;
- control and storage of the waste;
- evaluation of the doses to technicians, using various types of personal dosimeters.

Detailed technical publications in each field have been published in proceedings of national and international meetings.

In addition to these professionals, hundreds of individuals from the State of Goiás, hired firms, universities, the Civil Defence of the State of Rio de Janeiro, and local volunteers collaborated directly in various supporting activities, transportation, and civil engineering.

10. QUALITY CONTROL OF EQUIPMENT AND RECOMMENDATIONS FOR USE

The equipment employed in the emergency actions in Goiânia was submitted on site to quality control tests and checked for proper functioning and maintenance.

The quality control programme established during the second week of work included various measuring procedures, depending on the type of equipment and its purpose.

Owing to the multiplicity of types of equipment, the many different manufacturers, and the turnover of users, recommendations included such considerations as: distance, energy and directional dependency, types of radiation that might be detected, response time, calibration factors, and conditions for equipment use.
The accident in Goiânia had a great psychological impact on the Brazilian population owing to its association with the accident at the Chernobyl nuclear power station in the USSR in 1986. Many people feared contamination, irradiation and damage to health; worse still, they feared incurable and fatal diseases.

Some of the inhabitants of Goiânia were discriminated against, even by their relatives. Sales of cattle, cereals and other agricultural products, the main economic product of Goiás State, fell by a quarter in the period after the accident.

In order to allay these fears, the working team was encouraged to explain to people what they were doing and why, and, for example, to accept offers of drinking water and food from people's houses. They thus gained people's confidence and raised the credibility of official statements. Team workers made frequent appearances on television. A considerable effort was made by the co-ordinators to use the media to answer all questions and to treat the problem in an open and friendly manner. Their approach was to draw analogies, using simple language, with common applications of radiation, such as for medical X rays, and to recount as much as was known of the situation at the time.

However, the co-ordinators found it difficult to explain some aspects of radiological protection to the press. Talks were given for journalists, explaining in basic terms the applications and the biological effects of radiation. A pamphlet was produced jointly with the Health State Authority entitled "What You Should Know About Radioactivity and Radiation", and 250,000 copies were distributed. A telephone service operated 24 hours a day to answer inquiries and receive information about other possibly contaminated people or sites.

Several talks were given to different sections of the population and to community groups in order to restore confidence so that public life could proceed normally.

There were two distinct phases in the reaction of the communications media (the press, radio and television). The first was characterized by sensationalism, misinformation and criticism of the authorities.

In the second phase there was a much more mature coverage of events, seeking to inform the public and describing more clearly what was happening and what actions were being taken by CNEN and the Federal and State Governments. Later, CNEN decided to install a professional scheme of communication, taking away this extra burden from the local co-ordinators.

To encourage a more responsible presentation of events, CNEN personnel went to great lengths to clarify matters for the communication media, demonstrating and explaining their work.

News reporters could accompany CNEN technicians engaged in decontamination work and attending to casualties.
12. NATIONAL AND INTERNATIONAL CO-OPERATION

During the initial phase of handling the accident in Goiânia several experts were requested or sent through international agencies or through the existing bilateral agreement.

The Brazilian authorities informed the IAEA of the accident soon after its discovery, and requested assistance under the terms of the International Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency. The assistance given included the provision of experts and equipment.

In addition, several members of the Brazilian scientific community were present at the accident site, providing assistance and expertise.

13. LESSONS LEARNED

(1) The physical and chemical properties of a radioactive source are important factors in an accident. The records of sealed sources should contain that information. It is suggested that physical and chemical properties of sources should be taken into account in the licensing for manufacture of such a source.

(2) An adequate system of social and psychological support should be provided following a radiological accident. The psychological support should be provided to those individuals directly and indirectly affected and the personnel working in response to the emergency. A psychologist should be available for counselling, joining the group responsible for making quick decisions and planning action to be taken, and evaluating the possible stress to the casualties.

(3) The effectiveness of international assistance following a radiological accident depends on the substructure of the country concerned and on the professional profile of the expert and his will to get involved in the actual work instead of producing paperwork of questionable value. Emergency training courses should be held in developing countries as well as in developed countries where facilities are available and work well. In general, these programmes deal with emergencies responded to by strong organizations under a priori known conditions. In many countries circumstances are very different, equipment is diverse, the climate is adverse and matters are administered differently.

(4) Instrumentation should be capable of being adjusted to withstand field conditions so that it can be used in high humidities, high temperatures and unstable environmental conditions. Personnel using instruments should be trained to be able to obtain a clear indication of dose rate response for a wide range of doses; and to know the most suitable equipment in different conditions and its calibration factors.
(5) The provision of a temporary waste storage site near the area affected by a radiological accident is considered indispensable. A delay, usually a political one, in the decision on where to construct a site could permit greater dispersion of radioactive material in the environment.

(6) In general, a programme of inspection of radiological equipment and facilities is very important. However, it is only effective if coupled with some kind of enforcement system, such as assigning civil or professional liability in licensing sources.

14. FINAL REMARKS

The rule that radiation sources must be secure and under control must be given expression by competent national authorities in an appropriate regulatory system supported by appropriate rules and regulatory inspections, which was indeed the case in Brazil.

However, such a system cannot diminish the responsibility of the person designated as liable for a radioactive source. The regulatory system cannot and must not detract from managerial responsibilities; in particular, it cannot substitute for the licensee's responsibility for safety.

If, in spite of all precautions, an accident does occur and a radiological hazard is foreseen, there should be a well understood chain of information and command. In particular, in order to respond to a serious accident, a country would probably need to engage many of its qualified personnel, possibly from many widely separated establishments, and make use of much of the equipment available to it. An emergency plan should anticipate the need for integration, and this command structure should have been set up in advance.

In this regard it is important not only that responsibilities are assigned, but also that the necessary authority to obviate bureaucracy is conferred. For instance, the accident in Goiânia was remote from the centres of radiological expertise. The logistics of mobilizing personnel and arranging for material were a major difficulty (air transport was found to be essential). A clear chain of command will facilitate the provision of means necessary during emergencies, including means for enabling immediate mobilization. It follows that preparedness to respond to radiological emergencies should extend not only to nuclear accidents but to the entire range of possible radiological accidents.

In the accident in Goiânia, a number of practical problems were encountered in carrying out surveying and decontamination. However, two observations are worthy of note here:

(a) emergency equipment must be capable of operating in adverse ambient conditions;
(b) there will almost certainly be a need to engage workers without previous experience of radiological work, and even professional staff may not have had relevant operational experience.

Provision for training should therefore be made within emergency plants. The dissemination of information to the media, the public and, indeed, the response force is particularly important.

The accident in Goiânia was one of the most serious radiological accidents to have occurred to date. It resulted in the injury by radiation of many people, of whom four died, and the radioactive contamination of parts of the city. Radiological accidents are rare events; but this should give no grounds for complacency. No radiological accident is acceptable, and the public must feel confident that the competent authorities and individuals are doing all in their power to prevent them. Part of this process is to learn the lessons of the accident in Goiânia.

Annex I

THE SHIELDING OF THE REMNANT SOURCE CAPSULE

The rotating metal assembly holding the remains of the $^{137}$Cs source capsule was enveloped by a cloth bag and placed on a chair, as shown in Fig. 3, which in turn, was taken to the corner of the Health Department courtyard.

The dose rates measured at the surface of the bag were greater than 10 Gy/h and equal to 0.4 Gy/h at a distance of 1 metre, indicating a radioactive content of about 4.5 TBq (120 Ci), which represented less than 10% of the original source (i.e. 1375 Ci in Sep. 1987).

The radiation levels in the nearest sidewalk reached values of up to 30 mSv/h and the neighbourhood was quite apprehensive, not only because of the external radiation field but especially because of the possibility of the spread of contamination through rain and the falling of the source from the chair.

It was thus decided that immediate measures were to be taken to shield the whole set (chair + source) not only to reduce the exposure rates and therefore to minimize the number of evacuated houses but also recognizing the depressive state of the local inhabitants.

The best solution thought of at the time was to place a concrete sewer pipe over the chair, with the help of a truck, generally used for civil engineering purposes, equipped with a crane. Some difficulties had to be overcome during this operation, for instance:

- the chair was behind a 2 metre wall and very close to it (less than 20 cm away)
- the pipe oscillated when lifted
- the pipe had to be moved downwards slowly to prevent the chair from turning over.
FIG. 3. The chair after removal from the office. The dose rate at the surface of the cloth bag was 10 Gy/h, and at a distance of one metre 0.4 Gy/h.

FIG. 4. The shielded source and chair. The operation was completed in 30 minutes.
With the help of the Secretaries of the Health and the Transport Departments of the State of Goiás, it was possible to select experienced workers and successfully conduct the above described steps.

The sewer pipe was then filled with concrete, from a distance, pumping it through a hose over the wall. The whole operation was completed in 30 minutes and the radiation levels in the street were reduced to values less than 100 μGy/h.

In the following two weeks, the shielded source was repacked, to reduce the exposure rates at its surface even further, and a few days later it was transferred to the storage site (Fig. 4).

The decontamination of the courtyard and office was carried out over 15 days and the normal activities of the Health Department were resumed.

An estimated dose of 1.3 Gy was calculated for the technician who received the bag, placed it on his desk and left it there for some time, before removing it to the courtyard. Since the $^{137}$Cs remanent source remained in the bag his intake was negligible.

**BIBLIOGRAPHY**


EMERGENCY RESPONSE TO A SPILL OF TRITIATED HEAVY WATER — THE INTERFACE BETWEEN EMERGENCY RESPONSE, ROUTINE MONITORING AND RESEARCH

D.R. CHAMP, R.M. BROWN, E.L. COOPER, R.J. CORNETT
Environmental Research Branch,
Atomic Energy of Canada Limited,
Chalk River Nuclear Laboratories,
Chalk River, Ontario,
Canada

Abstract

EMERGENCY RESPONSE TO A SPILL OF TRITIATED WATER — THE INTERFACE BETWEEN EMERGENCY RESPONSE, ROUTINE MONITORING AND RESEARCH.

On 8 December 1988 approximately 4000 kg of heavy water spilled into the basement of the National Research Universal (NRU) research reactor at the Chalk River Nuclear Laboratories. As a result of the accident, tritium was released to the atmosphere, through the reactor stack as well as from the NRU building itself, and to the Ottawa River through a previously unsuspected pathway from the reactor basement to a process sewer. Although the incident was of little radiological consequence, it served as a useful test of our emergency response. Most notably, the emergency response activities illustrated the value of maintaining a close linkage between the routine monitoring, environmental research and emergency response programmes.

1. INTRODUCTION

Prediction of the off-site radiological consequences of an accident in nuclear facilities is critically dependent upon the availability of appropriate environmental monitoring data and the availability of valid environmental transport and dose estimation models. At the Chalk River Nuclear Laboratories (CRNL) responsibility for these monitoring and modelling activities resides in the Environmental Research Branch. The same staff are involved in research, routine monitoring and emergency response functions. The overlapping responsibilities have been very beneficial in establishing an efficient and effective emergency response capability. The emergency response to a recent accident at CRNL is illustrative of the value of closely linked functions. The discussion will focus on actions related to off-site emergency response to a loss of radioactive heavy water from the NRU research reactor at the CRNL site.
2. BACKGROUND TO THE ACCIDENT AND EMERGENCY RESPONSES

2.1. CRNL site and facilities

CRNL is located on the Ottawa River approximately 150 km upstream from Ottawa, Ontario (Fig. 1). The CRNL property covers approximately 37 km², with the bulk of the laboratory facilities located in a smaller Inner Area close to the river. The largest contributing sources of radioactive emissions are the National Research Experimental (NRX) and NRU research reactors and associated operations, such as decontamination and waste management activities. Emissions from these sources are released to the Ottawa River via liquid effluent streams and to the atmosphere via stacks and roof vents.

The NRU reactor is a heavy water moderated and cooled reactor. The heavy water primary coolant is circulated through the reactor fuel core and eight parallel shell and tube heat exchangers. The heat exchanger secondary coolant is Ottawa River water. NRU commenced operation in 1957 using natural uranium fuel but was converted to enriched uranium fuel in 1964. A versatile engineering research reactor, it provides facilities for basic research in condensed matter physics, production of long and short lived radioisotopes, material irradiations, power reactor fuel-performance testing and state-of-the-art severe fuel damage testing in support of the nuclear power reactor programme.
The process sewer is the major liquid effluent stream leaving the CRNL site. Since it carries secondary coolant water from the reactors, the flow rates are quite large. Tritium from a number of sources is released via the process sewer as HTO (tritiated water). The process sewer is sampled with an automatic sampler at the point of release. Samples are picked up from the sampler on a daily basis. A portion from each sample is analysed daily for fission products but not for tritium (daily tritium analysis was instituted subsequent to the accident). Other portions are used to make composite samples which are measured either weekly or monthly. Tritium is measured in weekly composite samples. Permanent records of the results of routine radiological monitoring are kept. Results of radiological monitoring are reported in a variety of ways. Releases from CRNL are not only regulated by the derived release limits (DRLs) but also by Administrative Levels and Warning Levels which are set at a small fraction of the DRL. The DRLs represent upper limits for routine emissions of radioactivity from CRNL to the surrounding environment. If a Warning Level is exceeded in an effluent, then the appropriate facility owner is notified promptly so that remedial action or an investigation can be initiated. If Administrative Levels are exceeded, then the owner is required to notify the Chalk River Environmental Authority, the group responsible for ensuring that site operations meet licensing requirements. Appropriate actions can then be taken by the Authority. Prompt response when a Warning Level is exceeded often results in remedial action, which avoids releases that exceed the Administrative Level. Very large releases would generally be first detected by the alarm system associated with the major facilities on the site. In most cases the Warning Levels are based on weekly average releases, while the Administrative Levels and DRLs for liquid effluents are based on monthly averages.

2.2. Routine radiological monitoring at CRNL

The radiological monitoring programme carried out by the Environmental Research Branch incorporates both liquid effluent monitoring and environmental monitoring. The major objectives of this programme are: to assess actual or potential doses to critical groups and populations from radioactive materials or radiation fields in the environment arising from normal operations or accidents; to demonstrate compliance with authorized limits and legal requirements and to provide public assurance. As part of the site licensing, DRLs have been set so that no person will receive more than 5 mSv (500 mrem) per year from the radioactive releases. Three distinct activities are required to meet the monitoring programme's objectives: source monitoring to measure radioactivity in airborne and liquid effluents at the point of release, environmental monitoring to provide relevant data along the transfer pathways to humans and meteorological measurements to define the pathways of airborne releases under both normal and abnormal conditions.
Through source monitoring, releases can be compared with the DRLs to demonstrate compliance with regulatory limits. More importantly, for the present discussion, source monitoring provides a warning of unusual or unforeseen conditions; timely warnings can lead to more rapid remedial actions if and when required; no protective actions were required in this instance.

Environmental monitoring provides data closer to the ends of the pathways (i.e. closer to humans) and may yield better estimates of the actual doses. Since data is obtained closer to the ends of the pathways, the uncertainty in dose predictions may be reduced because of the removal of the uncertainties associated with the transfer parameters used in the dose modelling. On the other hand, the uncertainty in dose predictions could increase, if the doses are low, because of the increased uncertainty resulting from measurements of lower concentrations of radionuclides. The measurements can then be used to verify the results of source monitoring and the models used in the DRL calculations, or more generally to help verify environmental transfer pathways and models. Environmental monitoring for atmospheric, terrestrial and aquatic pathways may be required for various radionuclides and sources. Both atmospheric and aquatic pathways were important in the accident and actions were taken to assess both pathways.

Meteorological measurements provide data on the long term average conditions which are used in the calculation of DRLs and in the design of the monitoring programme. In the case of abnormal or accident conditions, meteorological measurements are needed to predict doses to people. Data for both meteorological parameters and $^{41}$Ar releases from the reactor stack at CRNL have been collected for approximately 30 years. These data have been used to develop and test the atmospheric dispersion models used at CRNL and elsewhere [1, 2]. These atmospheric dispersion models were used to predict both on-site and off-site doses from the accident and to determine appropriate response actions.

2.3. Emergency response

Since CRNL is a major nuclear installation, there is the potential risk that an accident on the site may lead to significant doses off the site. Although the probability of this is very small, CRNL maintains an off-site contingency plan to cover such accidents. During an emergency, the CRNL Environmental Research Branch is called upon to carry out environmental monitoring both on and off the site. In addition, the meteorological data plus source term information is used to model the plume due to airborne releases.

2.4. The accident

On 8 December 1988, during a scheduled shutdown of the NRU reactor, an oil line failure caused a bearing failure in a main heavy water pump that in turn
caused a pump-shaft seal failure. The resultant heavy water release in the shielded pump room overflowed through wall penetrations to lower basement areas.

To assist the recovery and clean-up operation, a long precautionary stay-in was initiated during the afternoon of 8 December. During that time, venting of tritium from the reactor building was attempted.

Routine analysis of the process sewer weekly composite sample late on 12 December indicated a high tritium level. Subsequent analysis and river monitoring confirmed that almost immediately after the spill had occurred, 565 kg of heavy water containing HTO had been pumped inadvertently, as part of a normal sump draining procedure, to the Ottawa River.

3. RESULTS OF THE EMERGENCY RESPONSE ACTIVITIES

3.1. Airborne releases

Two actions were taken to assess and predict the consequences of an airborne release. First, as part of the standard procedures followed upon initiation of an emergency, data on the release height, meteorological conditions and a default value for the source were input to our Gaussian plume model, to predict plume dispersion and potential doses. The modelling program has access to meteorological data that is continuously acquired and updated. Source information can be entered as available from on-line monitors or other monitoring outputs. The modelling activities are supported by several decades of work linking research on atmospheric dispersion and atmospheric monitoring.

It was evident at the beginning of the emergency that tritium was the only radiological hazard associated with the heavy water spill and further that off-site consequences from atmospheric releases were likely to be negligible. On that basis the atmospheric dispersion modelling served mainly to define the probable path of the plume during the venting operation and the consequent potential for exposure of workers in the various buildings at CRNL. The modelling results were used to guide the environmental monitoring programme for airborne releases; this monitoring was carried out in conjunction with on-site monitoring of building vents and for building contamination.

The environmental monitoring deployed two sampling systems. The first comprised active samplers that pumped the air through columns of molecular sieve (MS)4A; these samplers were developed and used previously in a research programme to study the fate of tritiated hydrogen (HT) released to the atmosphere [3]. The second sampling system was based on passive samplers that are being assessed as part of a monitoring programme for occupational exposures; for the environmental sampling, the samplers used water and ethylene glycol as the adsorbent for HTO [4]. The active samplers were used during the period of venting
FIG. 2. Location map for air sampling sites. The arrow indicates the predominant wind direction (from the NW at approximately 300°) on 8 December 1988.
TABLE I. RESULTS OF ATMOSPHERIC HTO SAMPLING FOR NRU RELEASE — ACTIVE SAMPLERS ON 8 DECEMBER 1988

<table>
<thead>
<tr>
<th>Location</th>
<th>Time</th>
<th>Air conc. (kBq/m³)</th>
<th>% of DAC&lt;sup&gt;a&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>On-site 1</td>
<td>09:42–13:40&lt;sup&gt;b&lt;/sup&gt;</td>
<td>0.008</td>
<td>0.001</td>
</tr>
<tr>
<td></td>
<td>13:38–15:55</td>
<td>0.001</td>
<td>0.000</td>
</tr>
<tr>
<td>On-site 2</td>
<td>09:38–13:00</td>
<td>24.9</td>
<td>3.11</td>
</tr>
<tr>
<td></td>
<td>13:26–15:52</td>
<td>18.8</td>
<td>2.35</td>
</tr>
<tr>
<td>On-site 3</td>
<td>09:45–13:23</td>
<td>24.7</td>
<td>3.09</td>
</tr>
<tr>
<td></td>
<td>13:23–15:58</td>
<td>35.9</td>
<td>4.49</td>
</tr>
<tr>
<td>On-site 4</td>
<td>09:52–11:10</td>
<td>6.2</td>
<td>0.77</td>
</tr>
<tr>
<td></td>
<td>12:24–13:21</td>
<td>12.9</td>
<td>1.61</td>
</tr>
<tr>
<td></td>
<td>13:21–16:00</td>
<td>13.7</td>
<td>1.71</td>
</tr>
<tr>
<td>On-site 5</td>
<td>09:49–12:25</td>
<td>3.4</td>
<td>0.429</td>
</tr>
<tr>
<td></td>
<td>13:19–16:00</td>
<td>3.7</td>
<td>0.466</td>
</tr>
<tr>
<td>On-site 6</td>
<td>10:53–12:28</td>
<td>0.036</td>
<td>0.004</td>
</tr>
<tr>
<td></td>
<td>13:16–16:22</td>
<td>0.053</td>
<td>0.007</td>
</tr>
<tr>
<td>Plant boundary</td>
<td>09:50–16:40</td>
<td>0.026</td>
<td>0.003</td>
</tr>
<tr>
<td>1 km downwind</td>
<td>12:05–15:43</td>
<td>0.42</td>
<td>0.052</td>
</tr>
<tr>
<td>of site</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<sup>a</sup> DAC = Derived Air Concentration (800 kBq/m³). Multiply % by 40 to apply to the general population beyond the plant boundary.

<sup>b</sup> Venting occurred between 13:00 and 16:00; bearing failure occurred at approximately 06:00.

of the reactor building, but subsequent sampling was done using the passive samplers because the active samplers are battery powered and less suitable for prolonged sampling.

Six sampling locations were set up on the site in the prevailing wind direction; additional sampling locations were established near the boundary of the site and in local population centres.
### TABLE II. RESULTS OF ATMOSPHERIC HTO SAMPLING FOR NRU RELEASE — PASSIVE SAMPLERS

<table>
<thead>
<tr>
<th>Location</th>
<th>Start time</th>
<th>Sample time (d)</th>
<th>Air conc. (kBq/m³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>On-site 1</td>
<td>15:57 Dec. 8</td>
<td>0.84</td>
<td>0.41</td>
</tr>
<tr>
<td></td>
<td>12:05 Dec. 9</td>
<td>3.10</td>
<td>0.92</td>
</tr>
<tr>
<td>On-site 2</td>
<td>15:54 Dec. 8</td>
<td>0.84</td>
<td>3.24</td>
</tr>
<tr>
<td></td>
<td>12:08 Dec. 9</td>
<td>3.11</td>
<td>2.25</td>
</tr>
<tr>
<td>On-site 3</td>
<td>15:52 Dec. 8</td>
<td>0.85</td>
<td>1.16</td>
</tr>
<tr>
<td></td>
<td>12:12 Dec. 9</td>
<td>3.10</td>
<td>5.17</td>
</tr>
<tr>
<td>On-site 4</td>
<td>15:50 Dec. 8</td>
<td>0.85</td>
<td>0.26</td>
</tr>
<tr>
<td></td>
<td>12:14 Dec. 9</td>
<td>3.09</td>
<td>5.77</td>
</tr>
<tr>
<td>On-site 5</td>
<td>15:48 Dec. 8</td>
<td>0.85</td>
<td>0.15</td>
</tr>
<tr>
<td></td>
<td>12:17 Dec. 9</td>
<td>3.09</td>
<td>2.88</td>
</tr>
<tr>
<td>On-site 6</td>
<td>15:42 Dec. 8</td>
<td>0.84</td>
<td>−0.05&lt;sup&gt;a&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td>11:57 Dec. 9</td>
<td>3.10</td>
<td>3.11</td>
</tr>
<tr>
<td>Plant boundary</td>
<td>17:15 Dec. 8</td>
<td>0.80</td>
<td>−0.09</td>
</tr>
<tr>
<td></td>
<td>12:21 Dec. 9</td>
<td>3.10</td>
<td>1.01</td>
</tr>
<tr>
<td>1 km downwind</td>
<td>16:07 Dec. 8</td>
<td>0.82</td>
<td>−0.35</td>
</tr>
<tr>
<td>of site</td>
<td>11:45 Dec. 9</td>
<td>3.01</td>
<td>−0.11</td>
</tr>
</tbody>
</table>

<sup>a</sup> The negative values result from the background correction.

The results obtained from the active samplers are given in Table I; the sampling sites are shown in Fig. 2. Background HTO concentrations on the site are typically 0.001 to 0.003 kBq/m³. The highest concentrations (~25 to 35 kBq/m³) were observed near on-site location 3. This is consistent with the prevailing wind direction during the sampling periods. There was also some indication of tritium 1 km from the reactor building. The highest air concentration corresponded to about 4.5% of the International Commission on Radiological Protection (ICRP) derived air concentration (DAC) for radiation workers.
Results obtained with the passive samplers are summarized in Table II. The results for 1988-12-08/09 show a maximum concentration near on-site location 2. This is consistent with the meteorological conditions up to about 18:00 on 1988-12-08. After 18:00 the wind shifted from the NW (315°) to the SE (135°) and any tritium released would have been blown away from the samplers.

The results for the period 1988-12-09 to 1988-12-12 showed a maximum concentration near on-site location 4, but the plume is more widely spread. This is also consistent with the wind directions, which were largely from the WNW to N (~300° to 360°), but which did shift to other directions as well. This accounts for the observed tritium concentrations near on-site location 6 and the plant boundary site.

3.2. Liquid releases — monitoring

Routine releases of radioactivity through the process sewer are monitored with a composite sampler that collects about 50 mL of sample every 15 minutes using a purge-sample-purge cycle.

Composite samples are prepared on a daily basis for analysis of total beta and gross alpha activities, and the excess sample is archived. The daily analyses are deemed adequate to identify any events in the reactor that may lead to radiologically significant releases. Weekly composites (Monday to Monday) are prepared for tritium analysis and analysis of gamma emitters. Other analyses are carried out on monthly composites.

The first indication of a tritium release following the heavy water spill came when the weekly composite for the period 1988-12-05/12 was analysed. Since the tritium levels were much higher than normal, the archived daily samples for the same period were analysed individually. The highest concentration appeared in the sample collected from the morning of 1988-12-07 to the morning of 1988-12-08. Subsequent samples also showed elevated levels, but the levels were not as high as for the period 1988-12-07/08.

<table>
<thead>
<tr>
<th>Date</th>
<th>Concentration (MBq/m³)</th>
<th>Total flow (L/d)</th>
<th>Total release GBq (Ci)</th>
</tr>
</thead>
<tbody>
<tr>
<td>07/08</td>
<td>10 849</td>
<td>71.8 x 10⁶</td>
<td>780 000 (21 000)</td>
</tr>
<tr>
<td>08/09</td>
<td>117.6</td>
<td>72.7 x 10⁶</td>
<td>8 600 (230)</td>
</tr>
<tr>
<td>09/12</td>
<td>62.6</td>
<td>238.6 x 10⁶</td>
<td>15 000 (400)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Total: 803 600 (21 630)</td>
</tr>
</tbody>
</table>
Flow rates in the process sewer were calculated using flow figures from the CRNL pump house. These flow rates were used to calculate the total amount of tritium released. This introduces some uncertainty into the release estimates which are given in Table III. The uncertainty arises because the pump house measurements reflect the total volume of water pumped from the river to service the site's water requirements, including the reactor cooling water. The estimated total release over the period was 803,600 GBq (21,630 curies).

The major source of uncertainty in the estimated releases is due to our inability to answer the question "How representative are the collected samples?". Since the sampler cycles every 15 minutes, the release would have to have been prolonged to provide a representative sample. In this incident it is suspected that the release took place through a sump which is pumped out in 10 to 15 seconds. If the sampler happened to sample just as a small volume of tritiated heavy water was being released, then the concentration in the sample collected would be higher than the average concentration over the 24 hour composite period. Likewise, the sampler could just as easily miss a short pulse of activity and this would give low release estimates.

Sampling of the Ottawa River was initiated after it became apparent that an abnormal amount of tritium had been released. Sampling was started at three locations downstream: the pump house at Canadian Forces Base Petawawa, the pump house in Pembroke and the hydro dam at Chenaux (Fig. 1). The pump houses provide the water supply for Petawawa and Pembroke. The Chenaux samples were particularly useful, since the tritium was well mixed at Chenaux and flow measurements were available so that the integrated amount of tritium could be calculated.

The tritium concentrations measured in the samples collected at Petawawa are plotted in Fig. 3 as a function of time after the spill. The concentration declined after the start of sampling, which most likely means that the peak in concentration had already passed Petawawa. After day 8 the concentration rose slightly and fluctuated around 300 Bq/L until day 11. This was probably due to poor mixing caused by the changing ice conditions during this period. After day 11 the concentrations continued to decline.

Computer simulation of tritium levels in the Ottawa River was performed to provide an early prediction of the levels which could be expected downstream. A major uncertainty in comparing predicted concentrations with measured concentrations is the completeness of mixing of the released tritium into the river water. In order to check for completeness of mixing at Pembroke, samples were taken from the bridges around Allumette Island, just below the city of Pembroke (Fig. 1), where the river splits into four channels. The results for 1988-12-16 and -19 are given in Table IV. Although the results for all four locations are not exactly the same, they do indicate substantial mixing, particularly in the main channels (the middle two sampling points).

Tritium concentrations for the samples collected at Chenaux are presented in Fig. 4. At this location sampling was started before the peak in concentration,
FIG. 3. Tritium concentrations measured in Ottawa River. Samples collected from pump house at Petawawa.

TABLE IV. TRITIUM CONCENTRATIONS IN WATER FROM OTTAWA RIVER CHANNELS. (Ottawa River at bridges from Pembroke to Quebec side of river)

<table>
<thead>
<tr>
<th>Point</th>
<th>Date</th>
<th>Time</th>
<th>Days after Dec. 8</th>
<th>Bq/L</th>
<th>SD</th>
</tr>
</thead>
<tbody>
<tr>
<td>South Channel</td>
<td>Dec. 16</td>
<td>12:15</td>
<td>8.2</td>
<td>639</td>
<td>19</td>
</tr>
<tr>
<td>South-Middle Channel</td>
<td></td>
<td>12:20</td>
<td>8.2</td>
<td>799</td>
<td>17</td>
</tr>
<tr>
<td>North-Middle Channel</td>
<td></td>
<td>12:30</td>
<td>8.2</td>
<td>781</td>
<td>13</td>
</tr>
<tr>
<td>North Channel</td>
<td></td>
<td>12:45</td>
<td>8.2</td>
<td>613</td>
<td>17</td>
</tr>
<tr>
<td>South Channel</td>
<td>Dec. 19</td>
<td>8:35</td>
<td>11.1</td>
<td>481</td>
<td>7</td>
</tr>
<tr>
<td>South-Middle Channel</td>
<td></td>
<td>8:44</td>
<td>11.1</td>
<td>322</td>
<td>11</td>
</tr>
<tr>
<td>North-Middle Channel</td>
<td></td>
<td>8:55</td>
<td>11.1</td>
<td>318</td>
<td>5</td>
</tr>
<tr>
<td>North Channel</td>
<td></td>
<td>9:10</td>
<td>11.1</td>
<td>135</td>
<td>2</td>
</tr>
</tbody>
</table>
although the initial samples do show concentrations slightly above normal background levels. Sampling was continued long enough to define the shape of the tritium pulse. By using the flows measured during the same period, the total amount of tritium in the pulse was calculated to be $4.1 \times 10^{14}$ Bq (10 987 Ci). The net (corrected for background tritium levels) pulse resulting from the NRU spill was $3.81 \times 10^{14}$ (10 404 Ci). This is about half what had been estimated earlier from the liquid effluent samples. In view of the uncertainties in the estimates of the releases from the effluent sampler, the best estimate is that based on the samples from Chenaux. Initial estimates on December 8, by reactor operations personnel, indicated that approximately 2% of the heavy water released from the reactor may have been lost to the environment. This represented approximately 80 kg of D$_2$O, containing 53 780 GBq (1450 Ci) of tritium.

3.3. Liquid releases — simulation

Following the detection of elevated levels of tritiated water in process sewer effluents, a simulation model of HTO transport in the river was used to provide a rapid initial estimate of the movement of the contaminated plume down the river and to aid in quantifying the release. The simulation model had been developed in our
FIG. 5. Initial Simulated (-P) and observed (-O) HTO Concentrations at Petawawa and Pembroke Water Filtration Plants. □ PETAWAWA-P, + PETAWAWA-O, ○ PEMBROKE-P, △ PEMBROKE-O.

research programmes on surface-water transport development and testing of the model was greatly facilitated by the availability of historical data from the decades of monitoring on the Ottawa River [5].

The initial simulations assumed a HTO release of $7 \times 10^5$ GBq (21 000 Ci) based upon the process sewer monitoring release estimate. The average discharge rate (1988-12-8/15) was obtained from Ontario Hydro’s measurements at the Rapid des Joachim Power Dam on the Ottawa River, which is located approximately 30 km upriver of CRNL. Discharge from the Petawawa River (the major tributary in the river reach before Pembroke) was determined at the Water Survey of Canada monitoring station to be less than 4% of the Ottawa River flow. This dilution was ignored in the analysis. Bathymetric areas were calculated in previous studies from bathymetric sounding charts prepared by the Canadian Hydrographic service. The dispersion coefficient was estimated from previous studies [5]. Distances to points downstream were measured from topographic maps.

The simulations predicted that the peak concentrations (~1100 Bq/L) occurred at the Petawawa pump house on 12 December prior to the initiation of an intensive sampling programme (Fig. 5). Peak concentrations downstream were predicted to be
lower, as the plume of HTO was diluted by tributary inflows and by longitudinal dispersion. Eight measurements were available to check the simulation results on 16 December: the first three points on the Pembroke curve and the first five points on the Petawawa curve (Fig. 5). The simulated HTO concentrations agreed favourably with the few measurements available at that time. The best estimate of the HTO released into the river was 90% of the estimate obtained from the monitoring records for the process sewer. This estimate was obtained by integrating the area under the concentration curve and multiplying it by the mean discharge measured at the des Joachim Dam.

The simulations and measurements of HTO concentrations agreed closely (Fig. 6) when refined parameter values were used to account for the differences in area of different river reaches, different flow rates and dispersion coefficients resulting from the changes in roughness of the river morphometry (e.g. rapids). The peak concentrations predicted at each site are very similar to those estimated in the initial analysis. However, with the changes in model parameters in the final analysis it was clear that the total activity at Pembroke was overestimated in the initial simulations about twofold (Fig. 5). The initial simulation estimate was based on the preliminary
data derived from the process sewer monitoring and limited river sampling; the initial simulation provided a conservative estimate of the release. Better estimates were derived from later data obtained at Chenaux, as noted in Section 3.2. In the refined simulation the simulated total activity of HTO at Chenaux agrees (within experimental error) with the measured integral of \(3 \times 10^5\) GBq. The only major difference at Petawawa and Pembroke between the initial and final simulations was the longitudinal dispersion. Based on this analysis, the mean integrated dose to an individual from the HTO released during the entire event was approximately 0.15 \(\mu\)Sv (0.015 mrem).

4. SUMMARY AND CONCLUSIONS

The off-site radiological consequences of HTO released to the atmosphere were predicted to be inconsequential very early on in the accident. Confirmation relied upon the results of the atmospheric dispersion and dose modelling, and the environmental monitoring for HTO. The quality of the predictions from the atmospheric modelling was greatly enhanced by the involvement of the same personnel in both emergency response and research. Decades of extensive monitoring and meteorological data have been gathered and used to develop and test the models as part of our research programmes. The rapid environmental monitoring response for atmospheric HTO was largely due to the availability of skilled research personnel with suitable equipment that originated in the research programmes.

Rapid and appropriate response to the liquid release in the emergency was greatly aided by the close linkage of the research programme, through the development and testing of models for surface-water transport of contaminants, and the routine monitoring programme, through the availability of decades of monitoring data. The river simulations provided a rapid initial estimate of the movement of the contaminated plume down the river so that the accidental release, its consequences and appropriate responses (e.g. sampling schedules) could be assessed. In addition, the simulations integrated information on the quantity of HTO released, river flow velocities and individual concentration measurements, thus providing an overall picture of events.

The Health Protection Branch of Health and Welfare Canada also instituted a special monitoring programme on the Ottawa River on 15 December. Their results confirmed those of CRNL. The data collected confirmed that there was no health risk from the release. All concentrations measured were less than 2% of the maximum acceptable concentration of 40 000 Bq/L specified in the Canadian Drinking Water Guidelines [6]. The dose to an average Ottawa resident was estimated to be 0.2 \(\mu\)Sv (0.02 mrem) [1].

The routine environmental monitoring programme and emergency response actions were appropriate and adequate for radiation protection purposes. The only
change to the routine monitoring programme, following the accident, was the initiation of daily analysis for tritium in the process sewer effluent. However, if a release occurred, radionuclides other than tritium would be of much more concern and their presence would be readily detectable by the daily analysis of routine monitoring samples.

ACKNOWLEDGEMENT

The authors greatly appreciated the critical review and instructive comments of Dr. D.K. Myers.

REFERENCES


DECONTAMINATION OF THE HIGHLY CONTAMINATED SITES IN THE GOIÂNIA RADIOLOGICAL ACCIDENT

L.A. V I N H A S
Instituto de Pesquisas Energéticas e Nucleares,
Comissão Nacional de Energia Nuclear,
São Paulo,
Brazil

Abstract

DECONTAMINATION OF THE HIGHLY CONTAMINATED SITES IN THE GOIÂNIA RADIOLOGICAL ACCIDENT.

In September 1987 a very serious radiological accident occurred in Goiânia, Brazil, when a 50.9 TBq \(^{137}\)Cs source was inadvertently removed from a therapy unit and dismantled by junk dealers, contaminating several persons and sites in the city. Following radiation surveys, seven main foci of contamination were identified, which had resulted from the direct handling of the source or parts of it, either during its dismantling or subsequently. The paper describes the work carried out to decontaminate these highly contaminated sites. Details of the decontamination programme, including the radiation protection plant, the teams' organization, the sequence of operations and the procedures are presented. This decontamination programme was established in order to avoid the spread of the radioactive material and to optimize the rate of decontamination, keeping the workers' exposures as low as possible. During the operations seven houses, which presented high levels of contamination, had to be demolished and large amounts of soil had to be removed, as determined by soil profile measurements. The total volume of waste removed was 3100 m\(^3\), which was packaged in more than 3500 containers. Although the remedial actions were carried out under adverse conditions due to the site characteristics, the very high exposure rates with levels up to 1.1 Sv/h at one metre, and the social and political pressures involved, the objectives were entirely achieved. Indeed, the decontamination of the main foci was performed in only forty days; the workers involved in the cleanup operations were not exposed in excess of the authorized dose limits (1.5 mSv/day, 5 mSv/week and 15 mSv/month) and their internal contamination was insignificant.

1. INTRODUCTION

In September 1987 a very serious radiological accident occurred in Goiânia, Brazil, when a 50.9 TBq \(^{137}\)Cs source was inadvertently removed from a therapy unit and dismantled by junk dealers, contaminating several persons and sites in the city. As the result of the direct handling of the source or parts of it, either during its dismantling or subsequently, seven sites became strongly contaminated, these
being identified by radiation surveys. The total area of these seven main foci was about 5000 m². These sites were distributed within an area of about two km² in the central districts of the urban area of Goiânia.

The main objective of this paper is to describe the work carried out to decontaminate these main foci. Other aspects of the Goiânia radiological accident are described in detail elsewhere [1, 2].

The remedial actions took place under unusual conditions due to the sites' characteristics and their location in an urban area of a large town. It should be emphasized that because of political and social pressures, the decontamination had to be concluded in the minimum time, attaining extremely low levels.

2. DESCRIPTION OF THE MAIN FOCI

The seven main foci were: “Roberto’s house” (57th Street), Junkyards I, II and III, COPEL, “Wagner’s house” (63rd Street), and “the Cesspool house”. The main characteristics of the first five sites, which were highly contaminated, are described in order to illustrate the unique conditions in which the decontamination work was done.

In the site named Roberto’s house (57th Street) there were two houses, occupying about 50% of the 550 m² area. The source was broken in this site, transferring large amounts of radioactive material to the ground. By the action of a heavy rainfall, a planar source of 2 × 2 m was formed, resulting in the most highly contaminated hot spot with a dose rate around 1100 mSv/h at one metre. Three other hot spots were found in the yard with dose rates around 50 mSv/h. Furthermore, generalized contamination was also found in the yard and houses with dose rates between 0.2 and 5 mSv/h.

The Junkyard I site had an area of 900 m². In this area three houses and a metallic shed were located. It was characterized by piles of waste paper and metallic junk, and several bundles of papers distributed in disorderly fashion over the area. Seven hot spots were identified in this site, with dose rates between 50 and 1500 mSv/h, besides a generalized contamination with levels below 2 mSv/h.

The Junkyard II site had an area of 1250 m² (25 × 50 m). It comprised a house, a wooden shack and a square metallic structure (shed). These occupied around 30% of the yard. Several bundles and piles of paper, plastic wastes, debris and metal junk were spread all over the yard. The initial survey indicates a generalized contamination with dose rates below 3 mSv/h and twelve hot spots with dose rate levels up to 2000 mSv/h.

The Junkyard III consisted of a shed of 400 m², where 60 t of metal junk were stored. These materials and the floor presented a generalized contamination with levels up to 2 mSv/h, and only one hot spot on the floor with 50 mSv/h was identified.
COPEL was a used paper trading company. When the accident took place, this company had in its warehouse about 300 contaminated bundles of papers, each one weighing 270 kg, with a maximum dose rate of 50 mSv/h.

3. ISOLATION OF CONTAMINATED SITES AND CONTROL OF ACCESS AND EGRESS

The contaminated sites had been cordoned off using ropes immediately after their identification by radiation surveys. Even before the starting of the actual decontamination, the standard procedures for the control of access to the contaminated sites were adopted [3, 4]. Each controlled area was isolated by means of timber fences or existing barriers such as walls and gates. In the most highly contaminated sites, 57th Street, Junkyards I and II, it was necessary to isolate also part of the street at the entrance to the site in order to minimize the number of people who might be exposed, to avoid contamination of vehicles and to prevent interference in the decontamination operations.

At each site a checkpoint was established. At this checkpoint, a timber shed was constructed to serve as a radiological control station for individuals involved in the operation. It comprised an access control room, a dressing room, personal monitoring and decontamination rooms for the workers leaving the site and a small stockroom for materials and equipment used in the decontamination service. These rooms were distributed over an area of about 40 m², as can be seen in Fig. 1.
In cases where the controlled area was in the open, the checkpoint was located in the upwind direction.

4. THE DECONTAMINATION PROGRAMME AND ITS IMPLEMENTATION

The programme for the decontamination of the main foci was established with the following objectives in mind:

— to avoid the spread of contamination;
— to conclude the decontamination in the shortest period of time, attaining extremely low levels;
— to minimize the workers’ exposure and to avoid their internal contamination.

In the planning of the decontamination process the following aspects were taken into account:

— the characteristics of the contaminated sites
— the contamination level
— the high levels of exposure rates
— the dimensions of the contaminated areas
— the kind and volume of contaminated material to be removed.

It is interesting to note that a great part of the decontamination work was carried out in the open, under typical tropical weather conditions: intense sunshine with temperatures up to 38°C and sometimes heavy rains.

![Diagram of working zones](image-url)
In order to implement the decontamination programme in the most efficient way, and to optimize the rate of decontamination by speeding the cleanup, each highly contaminated site and its surroundings were divided into five working zones, shown in Fig. 2.

The main operations in each zone are described in the following sections.

4.1. Zone A

Zone A comprised the radiological control station and the checkpoint. Therefore, control of access to the contaminated area and the monitoring of persons leaving the controlled area were performed here. It was also used as a dressing room where the workers changed their clothes and shoes for protective clothing and working shoes. All persons entering the controlled area were provided with a personal dosimeter (film badge) and a direct reading dosimeter (quartz fibre electrometer (QFE) pen dosimeter). Each person leaving the controlled area was checked, the survey including the facial areas and the inside of the face mask. Surface contamination monitors with large end window Geiger-Müller detectors (‘pancake’ type) were used for this purpose.

4.2. Zone B

In Zone B, the empty packages were received and wrapped with plastic sheets. This procedure was adopted in order to reduce the probability of external contamination of the packages when they were transported to the contaminated area and during their filling with contaminated materials. Two types of containers were used to package the radioactive wastes: industrial drums of 0.2 m³ made of 18 gauge carbon steel and ribbed metal boxes with capacity of 1.7 m³ and a maximum load of 5 t [1]. A fork lift (auto lifter) was used to move ribbed metal boxes and filled drums.

4.3. Zone C

Zone C comprised the highly contaminated site itself.

In the highly contaminated sites presenting very high dose rate hot spots and generalized contamination with relatively high levels, it was necessary to remove everything, including houses and soil layers. Roberto’s house and Junkyards I and II were in this situation.

Experience showed that the best order in which to carry out the cleanup operations of these sites was:

— removal, whenever possible, of the hot spots;
— removal of the loose paper, plastic waste and so forth;
— removal of the bundles of papers;
— cutting and removal of the trees;
— removal of the furniture from the houses;
— demolition of houses and wooden shacks and removal of debris;
— demolition of metal structures, cutting the girders and removing the pieces;
— removal of contaminated layers of soil according to soil activity profile measurements;
— covering the areas with padding soil or concrete.

Before the start of the decontamination actions, and periodically during the operations, area monitoring was performed using teletectors (Geiger–Müller detectors). The hot spots were identified and marked. Warning signs were posted near these points. In the final phase of the cleanup operations, surface contamination monitors (pancake type) were used to find the remaining bits to be removed.

At the beginning of the operations, a lot of materials spread over the yard, e.g. loose papers, metal junk and plastic waste, had to be removed by hand, in order to allow the use, in Zone C, of heavy machinery such as excavators (back and front loaders/motor scraper), mechanical shovels and fork lifts.

During the decontamination operations in these three main foci, seven houses which presented high levels of generalized contamination had to be demolished and large amounts of soil had to be removed. The excavators proved suitable for these operations.

Junkyard III and COPEL were closed areas. In these sites the cleanup operations comprised the removal of the materials and decontamination of the buildings. Parts of the floor with fixed contamination were removed and the new floor was made with a layer of concrete.

In Junkyard III, the metal junk was cut up using an acetylene torch and motor saw in order to reduce its volume and to allow it to be packed in ribbed metal boxes.

At COPEL, there was a large volume of low level waste in the form of contaminated bundles of paper. The bundles were wrapped with plastic sheets and packaged in roll-on-roll-off shipping containers with 32 m³ capacity.

The total volume of waste removed from the contaminated sites was 3100 m³, which was placed in more than 3500 packages, including drums, metal boxes and shipping containers [1].

4.4. Zone D

Zone D, part of the street at the entrance to the site, was included in the controlled area. In this zone, the full packages were closed and externally decontaminated.

The check for external contamination was performed by the usual wipe test technique. A pancake type detector was used for monitoring the wiping samples.
The external decontamination was done using water and a weak solution of acetic acid (vinegar).

4.5. Zone E

The packages, closed and free of external contamination, were carried out to Zone E, to be prepared for shipment according to the Regulations for the Safe Transport of Radioactive Material [5].

For each package the maximum radiation levels at the package surface and at one metre from the surface were determined. Thus, the category of the package and its Transport Index were established.

The packages were labelled and loaded on lorries which were monitored. Warning placards were put on the vehicles.

During the decontamination operations 275 lorry loads of waste were transported to the temporary storage site.

5. DECONTAMINATION TEAMS' ORGANIZATION

The decontamination teams working in different highly contaminated sites were similarly organized. They comprised about 24 persons, half of them professionals dealing with radiation protection. Each member of the team with radiation protection experience was responsible for one of the following functions:

(a) co-ordination of the actions at the site (team leader)
(b) radiation protection control (radiation protection officer)
(c) dosimetric control and access control
(d) dressing the workers with personal protective equipment
(e) supervision of the work in Zone С
(f) monitoring of Zone С
(g) monitoring of Zone D
(h) packages decontamination
(i) shipment of the packages
(j) monitoring and decontamination of persons leaving the controlled area.

The members of the decontamination teams belonged to the following organizations: CNEN and its Institutes, NUCLEBRAS (INB), FURNAS, the Brazilian Army and a large private construction company.

6. PERSONAL PROTECTIVE MEASURES

The very high exposure rates found in the highly contaminated sites gave rise to a potentially abnormal exposure of the workers involved in the cleanup operations.
In order to avoid the primary dose limits being exceeded, the following derived limits, authorized by the competent authority, were adopted for the occupational exposure of the working staff [1]:

- daily limit: 1.5 mSv
- weekly limit: 5.0 mSv
- monthly limit: 15.0 mSv
- quarterly limit: 25.0 mSv

Each worker entering the controlled area used a pen dosimeter, beside his film badge, in order to ensure that these limits were observed.

Every day at the end of the work period, or after each risk operation, the pen dosimeter was read and the dose was registered in an individual file and reported to a control centre.

Every day the workers with the lowest accumulated doses were chosen to work in Zones C and D in order to obtain better dose distribution over the team.

The decontamination actions were carefully planned in order to enable doses from external irradiation to be controlled by the usual combination of shielding, distance and exposure time. In this context, the use, whenever possible, of heavy machinery in the decontamination operations contributed significantly to reduce the individual doses.

In order to prevent contamination of skin, internal contamination by inhalation and ingestion of radioactive materials, the workers wore protective clothing and equipment. These items were selected considering the operations to be carried out and the hazard present in the working zone, bearing in mind their influence on the user’s performance.

By taking into account these factors, the most appropriate protective devices for each working zone could be determined.

In working Zone A, B and E the persons wore a cloth coverall, boots or working shoes and gloves; in Zone D, the same protective clothing was used plus overshoes and a disposable plastic coverall; in Zone C, the highly contaminated site, the workers wore a cloth coverall, boots or working shoes, gloves, overshoes and, in addition, disposable plastic coveralls, a second pair of overshoes gloves and a full face mask.

A worker leaving Zone C would remove the external overshoes at a border between Zones C and D, go to the radiological control station, where he would remove the plastic coverall, the other overshoes and the face mask and would be monitored.

This procedure helped prevent the spread of contamination to the other working zones.
7. CONCLUDING REMARKS

Although the remedial actions were carried out under unusual and adverse conditions due to the site characteristics, the very high exposure rates, the large volume of waste to be removed, the unfavourable weather conditions, and the social and political pressures, the objectives of the decontamination programme were entirely achieved.

The decontamination of the main foci was performed in only forty days, restoring exposure levels similar to the natural background. In this period, a volume of waste of 3100 m³ was removed from these sites.

The measures adopted for personal protection proved effective. The workers involved in the decontamination operations were not exposed in excess of the authorized dose limits and their internal contamination was insignificant.

Finally, it should be emphasized that the skill, courage and determination of the decontamination teams' members and their mutual co-operation were essential for the successful implementation of the decontamination programme.

REFERENCES


WASTE MANAGEMENT IN THE GOIÂNIA ACCIDENT — THE CONTRIBUTION OF THE WASTE TREATMENT DIVISION OF THE NUCLEAR TECHNOLOGY DEVELOPMENT CENTRE

S.T.W. MIAW, M.F.R. GUZELLA, L.C.A. REIS,
P.O. SANTOS, E.M.P. SILVA, C.C.O. TELLO
Centro de Desenvolvimento da Tecnologia Nuclear (CDTN), Belo Horizonte, Minas Gerais, Brazil

Abstract

WASTE MANAGEMENT IN THE GOIÂNIA ACCIDENT — THE CONTRIBUTION OF THE WASTE TREATMENT DIVISION OF THE NUCLEAR TECHNOLOGY DEVELOPMENT CENTRE.

Radioactive wastes were generated in Goiânia, Brazil, by the accidental breakage of a $^{137}$Cs radiotherapy source (50.9 TBq) in September 1987. The Divisão de Tratamento de Rejeitos Radioativos (Waste Treatment Division) of the Centro de Desenvolvimento da Tecnologia Nuclear (CDTN), the Nuclear Technology Development Centre, was requested to perform tasks of general planning, establishment of waste management and specific procedures, identification of the national infrastructure, installation of treatment systems, decontamination of the critical areas and provision of interim storage. Of great value were the experience of the staff with waste management and the results obtained from R&D carried out by the Division.

1. INTRODUCTION

A vast amount of radioactive contaminated material was generated in Goiânia, Brazil, in September 1987, by the accidental breakage of a $^{137}$Cs radiotherapy source (50.9 TBq). It gave rise to an unprecedented health risk to the population in the area and created an unusual radioactive decontamination problem.

Upon the request of the Comissão Nacional de Energia Nuclear — CNEN (National Nuclear Energy Commission), the Divisão de Tratamento de Rejeitos Radioativos (Waste Treatment Division) of the Centro de Desenvolvimento da Tecnologia Nuclear — CDTN (Nuclear Technology Development Centre) participated in general planning, establishment of waste management and specific procedures, identification of the available infrastructure in Brazil, the installation of waste treatment systems, decontamination of the critical areas, and provision of interim storage.
The main objectives were to decontaminate the areas affected, and to obtain waste products that would best meet the transportation and storage criteria set by the CNEN, considering the limitations of the Brazilian infrastructure. All the tasks were performed under CNEN supervision.

The experience accumulated by the Division enabled its staff to provide efficient and prompt assistance in Goiânia. The R&D carried out by the Division includes the CDTN’s waste management, development and operation of different waste treatments and facilities, qualification of final waste products, development and testing of packages for storage, and transportation of radioactive materials [1].

2. EXPERIENCE OF THE WASTE TREATMENT DIVISION USED IN GOIÂNIA

2.1. Waste management

Based on the infrastructure available in Brazil, the following waste management strategy was suggested:

— to segregate the waste as far as possible
— to condition/treat the waste according to its type and activity and packages available
— to obtain final products suitable for transportation and storage.

These measures were all very important, mainly on account of the uncertainties of further treatment and of the characteristics and location of the repository. The factors associated with the quality of the final product would influence the cost-benefit analysis and the impact on the environment.

A waste control form was specially adopted, based on CDTN experience, to record all the information related to the conditioned waste. These data made the inventory estimation possible. The block diagram (Fig. 1) shows the steps in decontamination and the responsibilities of the different groups involved in the process.

2.2. Packages

The characteristics and location of the repository are very important in establishing the criteria that must be adopted for conditioning the waste. The decision to use transitory storage implies having suitable products for transportation to final disposal.
FIG. 1. Block diagram for Goiânia's waste management strategy.
After consideration of the packages available in Brazil within the short term, three types were selected:

— 200-litre drums tested and qualified for low specific activity waste, having good corrosion resistance, and available commercially. All the drums were inspected according to procedures established for their acceptance.

— Metal containers tested, qualified and designed by FURNAS (utility owner of the Angra NPPs) for contaminated steel components. This design was adapted to Goiânia's waste conditions. Local industries manufactured the containers under the technical supervision of the waste treatment staff. One-way concrete containers (shielding), type A, from FURNAS. At the time these had been only preliminarily tested for qualification.

2.3. The use of Brazilian bentonite

To improve the quality of solidified wastes, Brazilian additives have been investigated since 1980 at the Waste Treatment Division. Experimental work using bentonites for cementation of simulated wastes containing $^{137}$Cs had produced good results. These data provided some cementation parameters for caesium retention using bentonites. Under Goiânan conditions, the products obtained presented a low leaching rate without jeopardizing their mechanical resistance.

2.4. Conditioning of contaminated animal carcasses

During the accident, there were a lot of contaminated animals which had to be sacrificed and conditioned as waste. Veterinary professors suggested injecting a formalin solution into the killed animals to avoid the generation and release of gases during biological degradation. Afterwards all these animal carcasses had to be immobilized using lime and charcoal.

2.5. Treatment of contaminated urine

It was planned to treat contaminated urine as waste. Several tests were performed at the CDTN in order to establish a decontamination method for such material. Chemical precipitation was the chosen process, based on the Division's experience in this field, its simplicity, and its relatively low cost. By this means, the slurry obtained could easily be cemented. This process is also applicable to other generated liquid wastes. The method was tested with nitric acid and nickel ferrocyanate according to predetermined operational conditions. The decontamination factor attained was about 97%.
2.6. Treatment systems

2.6.1. Compaction system

A great amount of solid waste was generated during the decontamination work and its volume could be reduced by compaction. A CDTN compactor press was transported to Goiânia for this purpose. It was installed at Estadio Olímpico and a filtration system for contaminated aerosol retention was coupled to it.

2.6.2. Cementation system

On account of the unusual conditions of the accident it was impossible to install a radioactive waste cementation system in Goiânia. A simple one was tested using a conventional concrete mixer that could easily be adapted to the situation. Parameters such as mixing speed and residence time associated with the setting time were determined. The mixer capacity, performance, and its installation under radiological and operational conditions were evaluated.

3. DECONTAMINATION TASKS

The CDTN staff worked on the decontamination of all critical areas and in the storage facility. The names of the places mentioned below were given by the local people.

3.1. Estadio Olímpico (Olympic Stadium)

The first contaminated persons were isolated in the Estadio Olímpico. This place was also used to monitor the population. Prior to the establishment of the interim storage, collected wastes, including part of the source shielding, were sent to the Stadium. The compactable solid wastes were baled in 200-litre drums and the non-compactable wastes were conditioned.

3.2. Ferro Velho I (Junkyard I)

The Ferro Velho I, located on 26-A Street, was the name given to an old scrapyard. The area was about 800 m², with some houses and a workshop. The contamination was caused by the disassembling of a piece of the capsule of the caesium source. Ferro Velho I was the first place where all the infrastructure was set up, determining the layout of the control area, specification of the equipment and conditioning systems, and the logistics needed. All the measures applied were useful for
the decontamination of other areas. The waste of Ferro Velho I consisted of compacted waste paper, rubbish, equipment, soil, domestic utensils, furniture, clothes and some left over food. The first decontamination step was the removal of wastes with a high exposure rate, which were conditioned in concrete containers.

3.3. Ferro Velho III (Junkyard III)

The Ferro Velho III was a covered junkyard for non-ferrous metals. It was located on P-19 Street and had an area of about 400 m². It received all the scrap from other junkyards. The contamination had spread because of the manipulation of a piece of source shielding. It was verified that the main operational problem was related to the incorporation of contaminated dust but not to the exposure of the workers. The wastes consisted of cables, rubbish, metals, batteries, tools, a balance and two trucks. The waste volume reduction was done by using an acetylene cutting torch and an electric blade saw, followed by conditioning.

3.4. Rua 57 (57th Street)

The source was broken in the back yard of a house on 57th Street and the radioactive material spread out, contaminating persons, personal objects, furniture, domestic utensils, animals, houses, soil, trees and plants. During the emergency phase the more active waste had been collected in packages that later needed new conditioning in order to make their transportation to interim storage safe. A great amount of soil was removed and conditioned in metal packages.

3.5. COPEL — COMERCIO DE APARAS DE PAPEL LTDA

COPEL is a company located in Goiânia that deals with waste paper for recycling which is distributed to different regions of Brazil. Waste paper comes from industries, banks, junkyards, etc. and is compacted and tied into 200-kg bales. COPEL has many storage places in the city and the biggest one is located in the district of Santa Genoveva on 21st Street. Probably the main contamination came from waste paper handling at Ferro Velho I and was spread throughout the whole storage area, stackers, back yard, and trucks. The compacted waste paper bales were wrapped in plastic and conditioned in shipping containers, while rubbish, soil and scrap were conditioned in drums and metal containers. After the decontamination, the storage floor was covered with a new concrete layer, and the ground of the back yard was filled with uncontaminated soil.
3.6. Casa da Fossa

"Casa da Fossa" was the name given to the house located on 17-A Street, where some source fragments were thrown out into the sewage. The contamination spread from the sewage pipe into the soil. The soil was removed and conditioned in metal containers and the area was refilled with stone and sand.

3.7. Residences

Residences were contaminated properties such as houses, workshops, bars, etc. detected by the group called "Denuncias e Buscas" (Renunciation and Search). The contamination was disseminated by persons and animals. For the decontamination work, the area was checked over and, if necessary, residents were removed from the area. The procedure consisted of decontamination of the structural parts and removal of dust from walls, floors, furniture, etc. Contaminated objects were conditioned in plastic bags and then in drums or metal containers. Surface chemical decontamination was applied whenever possible. In some places contaminated animals were found, such as dogs, pigs, rabbits and birds. They were sacrificed and conditioned in drums with charcoal and lime.

4. WASTE IMMOBILIZATION

In Goiânia, the most active wastes were collected in drums and conditioned in concrete packages using mortar and concrete with bentonite. Drums containing organic material and/or with corrosion pits were conditioned in metal containers previously prepared with a concrete layer at the bottom. Afterwards, the voids were filled with the same mixture. The liquid generated during the decontamination work was treated by a chemical precipitation process and the resulting slurry was solidified with cement and bentonite.

5. TRANSPORTATION AND STORAGE

Before the trucks were loaded for transportation, the waste packages were decontaminated, monitored, and identified. A special control form was filled up with all available information. Each transport was accompanied by radiation protection and waste treatment supervisors.
6. WASTE STORAGE FACILITY IN ABADIA DE GOIÂNIA

The interim storage was specially constructed to receive waste generated during the decontamination work. It is located in Abadia, about 30 km from downtown Goiânia. The facility, which is highly protected and guarded, consists of a waste storage area, a control and utilities house and a natural dam. The storage area itself has six concrete platforms in a modular arrangement, and each one has 64 bases and six effluent control systems. Each base has a capacity of up to 32 drums or eight metal containers (storage in two decks). The packages were arranged in such a way that those with higher exposure rates were placed in the centre of the base. The position of each one was recorded on a special form; through this, it is possible to get information about any package at any time.

7. GENERAL COMMENTS

The Waste Treatment R&D programme developed at CDTN and the experience of the Waste Treatment Division staff were very helpful and strongly contributed to the different tasks performed in Goiânia. This Division is responsible for the waste management of the Centre and for the development of waste treatment processes, with the purpose of obtaining a waste product suitable for transportation and storage. In this context, package projects were developed and qualification tests were performed.

Considering the emergency situation caused by the accident, some parts of established procedures were not applied, especially those related to waste segregation and conditioning. As a consequence, a substantial amount of uncontaminated material was collected and conditioned.

Another aspect that has to be considered is the quality of the final product: the collected material was very heterogeneous and the quality control of the manufacturing of the packages was not efficient enough. This aspect is important for the future projects that should be developed for Abadia storage, with regard to the final disposal of the waste.

It was learned from the accident that the population must be aware of the risks involved in handling radioactive material. It is also important to have an adequate infrastructure and a well trained staff ready to act and to intervene efficiently in cases of emergency.

REFERENCE

OPERATIONAL FAILURE AND PLAN FOR THE RECOVERY OF A COBALT-60 SOURCE

F.A. KHAN, M.A. RAB MOLLA, K.O. AWAL, S.M.F. KARIM
Nuclear Safety and Radiation Protection Division,
Bangladesh Atomic Energy Commission,
Dhaka

M.A. MANNAN
Bangladesh Atomic Energy Commission,
Dhaka

M.A.T. ALI
Bangladesh University of Engineering and Technology,
Dhaka

Bangladesh

Abstract

OPERATIONAL FAILURE AND PLAN FOR THE RECOVERY OF A COBALT-60 SOURCE.

Radioactive materials and ionizing radiations including X rays are being widely used in Bangladesh, both in the private and the public sector. A significant number of sealed radiation sources of varying strengths are being used in industry, medicine, food, agriculture, etc. A $^{60}$Co Gammabeam 150-B having a nominal activity of 222 TBq (6000 Ci), installed at Bangladesh Institute of Nuclear Agriculture (BINA), Mymensingh in early April, 1979 has been used since then for the development of a genetically improved variety of seeds, pest control and for other research and development activities. In June, 1985 this irradiator became stuck in the exposed condition. The emergency cable provided for manual operation was used repeatedly in attempts to lower the source into the shielded position but without any success. Finally, the cable snapped inside the source room, leaving hardly any possibility for operation from outside. A salvage attempt was undertaken in August 1987, seeking to push down the source. Accordingly a hole in the roof of the source room was made above the indicator rod, which was subjected to repeated impact from the top but the source drawer did not go down; instead, the indicator rod bent, making the subsequent salvage action more complicated. It is now planned to salvage the source by locally developing remote control handling devices capable of operating from above the roof. A device is being designed and constructed to hold the indicator rod firmly and apply repeated push-pull and vibration to the source drawer, possibly using appropriate lubricants. The device would also be able to cut the bent portion and give sufficient thrust to it, if needed. These actions may free the source and possibly bring it down into the shielded position.
1. INTRODUCTION

Radioactive materials and ionizing radiations including X rays are being widely used in Bangladesh, both in the private and the public sector, for improving public health and the socio-economic development of the country. A significant number of sealed radiation sources of varying strengths ranging from 18.5 TBq (500 Ci) to 1850 TBq (50 000 Ci) of $^{60}$Co and $^{137}$Cs and an unknown number of sealed $^{192}$Ir sources of around 3.7 TBq (100 Ci) strength are being widely used in industry, medicine, food, agriculture, etc. for non-destructive testing, research and development activities and other institutional applications.

The $^{60}$Co Gammabeam 150-B in question, consisting of 20 slugs (each of diameter 0.63 cm and length 2.54 cm) having a nominal activity of 222 TBq (6000 Ci) at the time of installation, supplied by Atomic Energy of Canada Limited (AECL), was installed at Bangladesh Institute of Nuclear Agriculture (BINA), located on the Agriculture University Campus, Mymensingh in early April 1979. This unit consists of a source drawer, which moves vertically through the centre of a cylindrical main shield [1]. Source drawer elevation is conducted with the help of a simple motorized pulley-cable drive and the exposure time can be precisely controlled by means of the digital timer. An emergency cable is provided for lowering the source drawer into the safe position in the event of a mechanical disorder or electrical power failure. This irradiator was housed in an appropriately designed shielded room and was being used for the development of a genetically improved variety of seeds, pest control and for other research development activities, particularly in the field of agriculture.

2. THE INCIDENT

On 12 June 1985, eight small bags of lentils were made ready for irradiation by the operator as a matter of routine and the irradiator was switched on. At this stage, the operator did not notice or could not anticipate any sort of irregularity regarding the source except that the digital timer had not functioned from the very beginning of installation. After the desired period of irradiation monitored by a wrist watch the operator pushed the ‘off’ button to bring the source down. But unfortunately the source did not come down and on the control panel both red and green indicators were giving an ‘on’ signal simultaneously, indicating that the irradiator was stuck in the exposed condition (Fig. 1). In this situation the operator waited not for 7 seconds but for about 2–3 minutes and then tried to bring the source back to the shielded position by repeatedly using the emergency cable provided for manual operation, but without any success. Finally, the cable snapped inside the source room, leaving hardly any possibility for operation from outside. It may be noted that the length of the snapped off part of the emergency cable was only about 8 cm.
Nevertheless, the remaining part could not be seen from outside. In the ‘on’ condition the drive motor can be heard to run freely.

By November 1989 the activity of the source was reduced to approximately 46 TBq (1245 Ci). When an attempt was made to enter the source room, the dose rate near the lead shielded door in open condition was found to exceed the maximum range of the survey meter used (100 mR/h), thus restricting any direct on-site repair attempt. Because the source room was designed and constructed with proper shielding (Fig. 2), the dose rates outside the shielded room were within the operational limits. As far as the geographical extent of the mishap is concerned, it can be classified as level 1 [2], i.e. within the source room.

3. ANALYSIS OF THE INCIDENT

The present incident is not the only one encountered with this source. An analysis of the operational history of the source revealed that on one occasion, some-
time in December 1979, within eight months after installation the source stuck for the first time, in the down shielded position, and was freed by hammering the side of the shield pot with a padded hammer. On three other occasions the emergency cable was used to lower the source, but in every attempt the cable had to be pulled with full might implying jamming due to possible wrong construction of the cable run or some other, unknown reasons. Tracing back the cause of the present incident, first the cable was not laid correctly as per the instruction procedure, and secondly an inherent jamming tendency of the whole drawer assembly may be attributed to the maladjustment of the motor-clutch system.
4. ATTEMPTED SALVAGE ACTIONS

One of the IAEA specialists while visiting BINA suggested drilling a hole in the roof of the source room vertically above the indicator rod and driving down the source drawer with a rod striking the indicator rod. Alternatively, he suggested the use of a jack system to push it down, reacting against the roof. Should the proposed methods fail, his suggestion was to fill the labyrinth with rubble and entomb the source till the dose level reduces to acceptable limits. Based on the first proposal, a salvage attempt was undertaken by IAEA and BAEC experts in August 1987 to push down the source. Accordingly, a hole (around 12.5 cm in diameter) was made in the roof of the source room (Fig. 3) above the indicator rod and it was subjected to repeated blows from the top but the source drawer did not descend; instead the indicator rod bent, making the subsequent salvage action more complicated.

5. PROPOSED ACTIONS AND RECOVERY MEASURES

In view of the above facts, the Nuclear Safety Committee appointed by the Government formed a Sub-Committee to make the necessary arrangements to salvage the source by locally developing remote control handling devices capable of operating from above the roof. Such a mechanism, consisting of several items of equipment, should be in a position to cut the bent indicator rod and hold it firmly.
The holding process can be performed by chuck jaw or collet mechanism which can be controlled from above the roof. The diameter of the existing hole referred to in Fig. 3 has been widened to about 23 cm in order to accommodate the easy passage of the remote control handling system and to increase the incident light intensity within the source room from the roof top. Repeated push as well as pull at different frequencies and amplitudes will be applied to the source through the indicator rod. The appropriate lubricating/anti-rust fluid will also be used if necessary. It is anticipated that the vibration in the presence of anti-rusting fluid would free the source and it would be possible to lower it to the shielded position. If needed the normal and safe reuse of the source following recovery would be considered after examining all aspects of radiation protection, otherwise the source would be replaced or replenished.

REFERENCES


ON-SITE RECOVERY OPERATIONS
(NUCLEAR FACILITIES)

(Session II)

Chairman (Part 1)

C.A. NEGIN
United States of America

Chairman (Part 2)

V.I. ZABRODIN
Union of Soviet Socialist Republics
Invited Paper

THE THREE MILE ISLAND UNIT 2 RECOVERY: A DECADE OF CHALLENGE

J.E. HILDEBRAND
GPU Nuclear Corporation,
Parsippany, New Jersey,
United States of America

Abstract

THE THREE MILE ISLAND UNIT 2 RECOVERY: A DECADE OF CHALLENGE.

The cleanup of Three Mile Island Unit 2 (TMI-2) has encompassed a decade of unprecedented decontamination and defuelling operations. The scope, complexity and cost of the project has far exceeded even the most pessimistic predictions made in 1979. At every phase of the project, the unexpected has occurred and the obstacles encountered have necessitated innovative and imaginative solutions. Similarly, the radiation protection programme required to ensure worker protection has been remarkable for its degree of planning and scrutiny. The hostile radiological conditions resulting from the accident were extraordinary, with gross surface contamination throughout the plant causing prohibitively high radiation levels. Worker doses, however, have been successfully managed and maintained 'as low as reasonably achievable' (ALARA) through the application of effective radiation protection principles and practices. It is now projected that the total collective dose up to the end of the cleanup (12 years through 1990) will be about 65 man-sieverts. When comparing this total to operating US plants over the same time period, it is clear that the health risk to TMI-2 cleanup workers has been no greater than to other nuclear plant workers. The paper reviews the cleanup of TMI-2 from a radiation protection perspective and addresses the problems encountered and the experiences gained in ensuring a safe and reliable work environment.

1. INTRODUCTION

A decade has passed since that ill-fated morning in 1979 when the public's attention became transfixed on a tranquil farming community in the eastern USA. As the drama of the event played out, Three Mile Island, depicted by those ominous looking cooling towers, became a symbol for fear, failed technology and misguided trust.

In reflecting on the experience of Three Mile Island Unit 2 (TMI-2) one can only be humbled by the devastating nature of the accident, which resulted in a ruined nuclear power plant, a billion dollar cleanup and a nearly bankrupt owner. Viewing the accident philosophically, what emerges from the experience is the quintessential
paradox. To the extent that the core was severely damaged and the plant grossly contami­nated, the technology failed. However, to the extent that the radioactive releases from core damage were contained, and no member of the public was harmed, the technology proved to be an enormous success. I stress this latter point because not enough interest and attention has been applied to the negligible health effects to the local population.

The truth is that most of the radioactivity was contained during the accident. The average dose to a member of the public was about 80 microsieverts, or about the dose I received in flying to this symposium. The consensus among the scientific experts is that it is extremely unlikely that any person will suffer discernible health effects during their lifetime from radiation associated with the TMI-2 accident. These were the projections made ten years ago and, to date, no new technically credible evidence has surfaced which would cause those conclusions to be changed. And yet the issue of health effects around TMI continues to receive considerable interest, thus perpetuating the myth that the TMI-2 accident caused, or will cause, health effects in the local population.

The recovery at TMI-2 presented the US nuclear industry with its greatest challenge, and one that bore heavily on its very survival. But the industry did coalesce to recover TMI-2 safely and to address the lessons learned from the accident. The changes that occurred in engineering, design, training, operations and regulations have been great. A detailed accounting of ten years of cleanup experience would consume the entire symposium. The American Nuclear Society Topical Meeting in Washington in 1988 did just that in a sort of "everything you ever wanted to know about TMI-2 but didn't have time to listen" meeting where well over 1200 technical papers were presented on the TMI-2 accident and recovery. The objective of this paper is to take a reflective overview of the cleanup, focusing on the radiological challenges that have been encountered during the past decade.

2. AN OVERVIEW OF THE TECHNICAL CHALLENGE

If there is one overriding tenet realized from the cleanup, it is that the operation has not simply been a defuelling and decontamination effort, but rather a grand research project of a complexity and scope far exceeding even the most pessimistic predictions made in 1979. It is apparent that the pioneering successes we have achieved have been largely due to our ability to improvise, adapt, modify and redesign to achieve practical solutions to onerous technical problems. At every phase of the project the unexpected has occurred and the obstacles encountered have neces-

---

1 The President's Commission on the Accident at Three Mile Island, Staff Reports of the Public Health and Safety Task Force to the President's Commission on the Accident at Three Mile Island US Govt Printing Office (1979).
situated the best ingenuity, imagination and engineering know-how available. Similarly, the radiation protection programme required to ensure worker safety has been remarkable for its degree of planning and scrutiny.

Much has been written about the technical lessons learned from the accident and cleanup. Similarly, from a radiation protection perspective, a large amount of information has been gained and applied throughout the industry. Of particular interest are the lessons learned in the areas of exposure management, decontamination techniques, robotics, respiratory protection, beta monitoring and dosimetry and heat stress management. The most clear lesson that has emerged from TMI-2, however, is that even with onerous radiological environments, decontamination work can be accomplished with low dose and minimal risk to workers through the application of effective radiation protection principles and practices.

Probably the most difficult radiation protection problem experienced during the cleanup has been the unknowns involved in performing high radiation jobs for the first time. Whether it was the first post-accident entry into the containment building, the first remote inspection of the core, or lifting of the reactor vessel head, the approach has been to obtain as much radiological data as possible, develop detailed work procedures and contingency plans, perform ALARA reviews and establish administrative dose controls.

My involvement with the cleanup dates back to early 1980. At that time, the radiological conditions were unknown in the containment building since manned entries had not been made. Furthermore, core damage predictions had been made but were later shown to be grossly underestimated. Throughout 1979, recovery efforts had focused on stabilizing the damaged core and decontaminating the auxiliary and fuel handling buildings. It was not until mid-1980, with the first manned entry into the containment building, that the extent of the accident began to be recognized. That first entry made by two volunteer engineers was a benchmark ALARA case study in preparing for entry into a hostile environment. Given the unknowns and uncertainties of what was to be found in the containment building, every possible precaution was taken for that first entry. Although some radiation measurements had been obtained through building penetrations, the true magnitude of the radiological conditions in the building was not known. The two engineers were equipped with nearly every protective garment and monitoring device available. Each man wore about 35 kg of equipment and several layers of protective clothing, including a deep-sea diving suit and self-contained breathing apparatus. Also, each man wore 46 dosimeters in nearly every location one could be placed. The comparison to a moon walk was certainly not inappropriate. Although the re-entry to the containment building presented an extreme case in protecting workers against unknown radiological hazards, the same approach utilizing proven radiological principles has been used throughout the cleanup for work in radiological environments. That first entry lasted only 20 minutes; however, as entries continued, the radiological portrait of the containment building demonstrated the extent of the severe radiological conditions and
the difficult work considerations to be encountered. The upper levels of the contain­
ment building were grossly contaminated with average dose rates in the 4 to 5 mil-
lisieverts per hour range. In the basement, 2.5 million litres of coolant had been
discharged from the reactor vessel to the building basement, creating a pool of water
3 metres deep.

During 1982–1983 the entries into the containment building were made mainly
to obtain technical information and radiological data for the purpose of planning
cleanup operations. It was not until 1982 that the extent of core damage was known.
During ‘Operation Quick-Look’, a small television camera discovered a two metre
void in the top of the core with a bed of rubble in the middle of fractured fuel assem­
blies. As it turned out, this revelation was the portent of a constancy of the cleanup;
namely, the element of discovery and surprise. At every new phase of the project
a new surprise has been encountered and with it, new obstacles and hurdles to over­
come. Mention of a few of the surprises will help to make the point.

— During the first manned entry into the containment building, an unexpected
event occurred which belied the belief that every possible consideration had
been planned for. Months of planning and training had readied the two man
crew for the historic event, which had received extraordinary media coverage.
As the world awaited the moment, the two men approached the air-locked door
but could not open it; it was stuck. After several futile attempts the mission
was aborted. A month later the entry was made successfully.

— In 1980, during an entry into a valve alley in the auxiliary building, highly
energetic beta radiation was unexpectedly encountered that penetrated protec­
tive clothing, causing high skin doses. The beta radiation was from
strontium/yttrium-90 fission products that had not generally been previously
experienced by health physicists in the nuclear industry. The extent of the beta
radiation resulted in a complete redesign of the personnel dosimetry system
used to measure worker doses in mixed beta–gamma fields.

— The second major in-vessel video inspection took place in early 1985 when a
small television camera was threaded through the lower plenum area, where
20 to 30 tonnes of molten debris were discovered. It was apparent with this
revelation that the defuelling operations would be faced with yet another level
of difficulty in cutting through layers of structural support plates in the lower
head to extract the large chunks of congealed fuel and structural debris.

— Another surprise came when the 2.5 million litres of water were pumped from
the containment building basement. As thermoluminescent dosimeters (TLDs)
were lowered into the basement to obtain radiation levels, it was discovered
that radioactivity had soaked into the unpainted concrete and cinder block
walls. In particular, the cinder block walls surrounding the elevator shaft and
the enclosed stairwell had acted like a sponge, leaving dose rates up to ten
sieverts per hour on the walls. Although considerable flushing of the walls had
been done using robotics, the hostile radiological environment has precluded manned entries into the basement.

— Another surprise occurred following the start of defuelling in the autumn of 1985 when microscopic organisms made the reactor vessel water so cloudy that the defuellers could not see the debris chunks with underwater cameras. Once again the unexpected had occurred, the organisms were thriving on the warm water and hydraulic fluid leaked from the defuelling tools. After months of testing with various physical and chemical methods, the critters were finally destroyed with hydrogen peroxide and removed with coagulant and filtration processes.

These are just a few of the obstacles that have confounded and exacerbated the cleanup schedule and cost. The solutions to these and other problems have required skill, imagination and perseverance. Clearly, it has been these characteristics manifested by dedicated workers and the management staff that has resulted in a successful outcome with minimal health risk to workers.

Much has been written about the techniques and processes during the cleanup. Any attempt to provide a detailed litany of achievements would be foolhardy and a great bore, but I do wish to review a few of the major activities that have been used during the cleanup and briefly discuss our experiences.

3. TASK MANAGEMENT

From the early days of work in the containment building, work activities have been monitored and directed from a centralized co-ordination centre. From this location, task co-ordinators observed work in progress and communicated with workers through audio and video systems in the building. To a large extent, TMI-2 has pioneered the use of video camera technology for surveillance and inspection in the nuclear plant environment.

The most dramatic use of video cameras has been surveillance of the damaged core and the reactor systems. But equally important has been the use of cameras in hostile radiological environments to direct and monitor work. For example, because of the high radiation levels in the containment building, it was recognized early on that camera surveillance would be essential to ensuring worker safety and monitoring work evolutions. The cameras can survey 75% of the containment building with remote pan, tilt and zoom capabilities. The cameras have proved extremely valuable to task management and personnel safety by allowing supervisory work guidance without the supervisor being in the area. This resulted in significant personnel dose savings in the early years when radiation levels were considerably higher than they are today. As the containment building was decontaminated and radiation levels were lowered, the importance of continuous visual contact lessened, but the co-ordination centre continues to be used as an effective task management tool.
In conjunction with videocameras, numerous types of two-way communication systems have been extensively tested and used throughout the cleanup with varying degrees of success. The lessons learned from extensive use of video technology and communication equipment certainly have direct applicability at nuclear plants in hostile work areas for observing, monitoring and evaluating work activities. These systems can pay real dividends in dose savings, improved industrial safety and work direction.

It was recognized early in the cleanup that because of the large amount of protective clothing and equipment required by cleanup workers, and the substantial number of workers entering the containment building, a dedicated facility was needed to expedite the entries and assist workers in suiting up. Some workers require multiple layers of protective clothing, respiratory protection, welding gear, radios, airborne samplers and radiation detection equipment. The facility, known as the Personnel Access Facility or PAF, provides for ready access to all equipment and assistance to workers in suiting up. The PAF has proved to be a substantial benefit in assuring that workers are properly and consistently suited up to minimize skin contami­nations and have all the equipment required for their tasks.

In addition, the PAF serves as a staging area for personnel awaiting their authorization to enter the building. Clearance to enter the building is given by the co-ordination centre only when the job is ready to be performed. The PAF serves as a tool in managing and in essence, 'orchestrating' the containment entry process, thus enhancing productivity and reducing personnel dose.

4. DECONTAMINATION

The most successful decontamination techniques utilized throughout the cleanup have been high pressure flushing and scabbling. Pressure flushing is first performed to remove excessive amounts of smearable contamination in most areas of the auxiliary and containment buildings. This method utilizes remote nozzles and hand held lancers. A typical flush is performed with pressures ranging from 200 to 900 kg/cm² and flow rates ranging from 22–68 litres per minute.

The typical second step performed is to remove surface coatings and concrete with the use of scabblers. Scabblers are air powered tools which have piston and bit assemblies designed for concrete and surface coating removal. Scabbling is the method that produced the greatest general area dose rate reductions. After an area has been scabbled, it is scrubbed and wiped to reduce levels of smearable contamination prior to new coatings being applied.

The typical third step in decontamination is to perform a steam and hot water flush of the area. This type of flush has been performed to reduce smearable contamination levels. If it was desired to release an area as radiologically clean, the steam and hot water flush would be followed by hands-on wiping.
These techniques have been substantially effective in removing large amounts of surface contamination and contamination that had leached into concrete. To date, nearly 90% of the auxiliary and fuel handling buildings has been decontaminated to previously defined end-point criteria. Most of this area has been released for general access without any protective clothing being required.

The upper levels of the containment building have been decontaminated to levels which allow workers to remain in the building for hours with minimal dose. The exposure levels have been reduced from 4–5 millisieverts per hour in 1980 to about 0.3 millisieverts per hour currently. Furthermore, dose rates on the defuelling platform where the defuelling work is performed have been maintained below 0.1 millisieverts per hour.

5. CONTAMINATION CONTROL

Throughout the cleanup, extensive measures have been taken to provide effective control of radioactive contamination to prevent spread outside the plant and to minimize the dose to workers from skin contamination and inhalation of airborne radioactive dust. Based on the nature of the job and the contamination levels involved, varying levels of protective clothing and respiratory protective devices are prescribed. The protective clothing requirements may range from a single pair of coveralls in low contamination zones, to coveralls, impermeable plastic suits and respirator protection for contamination work and where there are high levels of airborne activity.

Our approach in prescribing protective clothing in hostile environments has been to consider the overall risk to the worker, including heat stress, the risk to visual and hearing acuity, and cardiopulmonary stress. Using this approach, we have minimized skin contaminations to workers while at the same time maximizing worker productivity and efficiency, thereby minimizing worker doses and overall stress to the worker.

To improve the efficiency of the worker, extensive mock-up training has been used for jobs that involve work in high radiation environments. The mock-up training is performed in clean areas on replicas of the workstation and in the actual protective clothing required for the job. The two steam generators, for example, were inspected for residual fuel. The job took place in the highly contaminated D-rings of the containment building. Prior to the job being performed, it was fully rehearsed on a scale model of the upper head of the generator. Largely because of the training, the inspections were performed without incident with minimal worker exposure.

Defuelling operations were being performed with coveralls, plastic head and shoulder coverings, multiple layers of gloves, radio earphones and plastic face shields to protect against liquids being splashed in the face.
Because of the unusual nature of many of the cleanup operations, practical and innovative solutions to worker comfort problems have had to be developed. We realized early in the cleanup that it does no good to prescribe extensive protective clothing and expect the worker to perform his job effectively if the comfort level is poor.

6. OCCUPATIONAL DOSE CONTROL

For any nuclear plant today, the key performance indicator of an effective radiation protection programme is the collective dose to workers; this has been no different at TMI-2. Throughout the cleanup, predictions of high worker doses have been the subject of considerable interest. In fact, in the early stages of the cleanup, there were those who were calling for the establishment of a separate worker registry for cleanup workers to allow long term monitoring of possible worker health effects. It

<table>
<thead>
<tr>
<th>Year</th>
<th>Total worker dose (man-sievert)</th>
<th>Maximum worker dose (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1979</td>
<td>4.2</td>
<td>45</td>
</tr>
<tr>
<td>1980</td>
<td>1.9</td>
<td>21</td>
</tr>
<tr>
<td>1981</td>
<td>1.4</td>
<td>20</td>
</tr>
<tr>
<td>1982</td>
<td>3.8</td>
<td>30</td>
</tr>
<tr>
<td>1983</td>
<td>3.7</td>
<td>27</td>
</tr>
<tr>
<td>1984</td>
<td>5.1</td>
<td>37</td>
</tr>
<tr>
<td>1985</td>
<td>7.2</td>
<td>35</td>
</tr>
<tr>
<td>1986</td>
<td>9.0</td>
<td>34</td>
</tr>
<tr>
<td>1987</td>
<td>9.7</td>
<td>35</td>
</tr>
<tr>
<td>1988</td>
<td>9.2</td>
<td>36</td>
</tr>
<tr>
<td>1989</td>
<td>4.7</td>
<td>29</td>
</tr>
<tr>
<td>1979-1989</td>
<td>59.9</td>
<td></td>
</tr>
</tbody>
</table>

\(^a\) = 28 March 1979 to 31 December 1979.

\(^b\) = Up to 31 August 1989.
was expected that TMI-2 workers would be a highly exposed population and should be treated differently from other nuclear plant employees. However, through an effective radiation protection programme, worker doses have been minimized and are far lower than predicted.

As shown in Table I, the total TMI-2 cleanup collective dose to date is about 60 man-sieverts, over the 10.5 year period of the project. The projected estimate for the total cleanup is about 65 man-sieverts compared to the original NRC estimate of 130-460 man-sieverts, made in the Programmatic Environmental Impact Statement (Rev. 1) in 1980. Furthermore, no worker has been exposed to whole body radiation levels in excess of Nuclear Regulatory Commission standards and since 1979 no worker has exceeded 40 millisieverts per year. The important point is that the cleanup has not proved to be the high dose project originally expected. Through effective radiation optimization practices, comprehensive dose reduction measures, ongoing job planning and a strong management commitment to ALARA and optimization principles, the cleanup has been performed with minimal health risk to the TMI-2 work force. In fact, the risk to our workers has been no higher than that of typical workers at operating nuclear plants in the USA.

7. WORKING IN HIGH HEAT STRESS ENVIRONMENTS

High thermal work environments are common in nuclear plants. This was the case in the first few years of the decontamination work in the containment building. Because of the high contamination and airborne radioactivity levels in the building, workers wore layers of protective clothing including respirators, and often impermeable plastic suits as well. The result was that the potential for heat stress, rather than radiological conditions, became the limiting stay-time factor. In response to the problem, our industrial safety people, in conjunction with the Electric Power Research Institute and Pennsylvania State University, developed a comprehensive heat stress control programme consisting of employee training, administrative controls and personal cooling devices.

The Heat Stress Index utilized at TMI is a computer program based on input of environmental parameters and metabolic work rate estimates. Given this input, a safe stay-time is calculated for the specific task to be performed in a work area. The most notable feature of the program is its ability to determine the insulating effect of different protective clothing. In addition to providing the evaluation of specific jobs, a table of stay-times has been generated based on work rate, air temperature and clothing type. These guidelines are used by supervisory personnel in planning and conducting work. While training and administrative controls provide an increased margin of safety for the worker, they do not solve a more basic problem — how to provide protection while increasing productivity. As a result, personal cooling devices in the form of ice vests were introduced. The ice serves as a heat
sink for metabolic heat produced by the body and dissipates this heat as it melts. The ice vest has been very successful in extending work times in hot environments, in some cases by a factor of two to three.

In the first entries into the reactor building in 1980, respiratory protection consisted of self-contained breathing equipment. This type of respirator was required initially because of the high airborne radioactivity levels. However, these respirators limited workers’ stay-times to about 20 minutes. Negative pressure respirators could not be used extensively because of the low protection factor and the limited time a worker can tolerate the respirator. A search, therefore, was made for a respirator that had a high protection factor, high degree of comfort, allowed mobility, and did not limit stay-times. The device selected was the powered air purifying respirator (PAPR). This battery operated device draws ambient air through a filter and blows it into the face piece, thus providing a positive air flow to the face. Our experience is that the PAPRs have proved extremely beneficial at TMI-2 to the point that we use negative pressure respirators sparingly. We have found that employees work more efficiently with PAPRs than with the conventional respirators. I am convinced that the PAPR is the greatest improvement in respiratory protective devices in recent years and will have real benefits in nuclear plant operations.

8. ROBOTICS TECHNOLOGY

Perhaps the most well publicized technology of the TMI-2 cleanup has been robotics. With the large quantities of fission products discharged into the containment building basement, extremely high radiation levels prevented manned entries and remotely operated vehicles have been the only means of inspecting the area, obtaining radiation readings and core samples, as well as performing decontamination operations.

The development of robotics at TMI-2 has been an evolutionary one with each generation of technology building on the experience gained from the previous robot. A number of robots have been tested over the years with such names as ‘SISI’ and ‘Fred’, but the most noteworthy remotely operated vehicle has been the Remote Reconnaissance Vehicle (RRV) known as ‘ROVER’. This vehicle was constructed specifically for use in the containment building basement where dose rates were as high as 10 sieverts per hour. Perhaps the most important element of the vehicle is its modular design, which allows the attachment of different tools to accomplish various jobs. ROVER was developed by Carnegie Mellon University under EPRI contract. ROVER is a rugged, six-wheeled vehicle with a payload of cameras and radiation monitoring equipment. ROVER has successfully performed core boring of walls, flushed walls and equipment and vacuumed and removed sediment remaining on the basement floor.
A less complex robot than ROVER is ‘LOUIE’, which has been used in highly contaminated areas in the auxiliary building. Two models of LOUIE have been used to perform radiation monitoring, surveillance, and floor scabbling in cubicles where high radiation levels precluded manual decontamination operations.

TMI-2 has been in the forefront of remotely controlled mobile vehicles in the nuclear industry. Much has been learned with each generation of robot. A number of uses for remotely controlled vehicles in operating plants can be envisaged, including radiation monitoring and inspections in high dose rate areas, spill cleanup, decontamination, and reactor cavity cleaning as well as some routine plant maintenance. The major disadvantages of the vehicular robots are cost, plant accessibility and reliability. Consequently, it is uncertain at this time as to the role future generations of robots will play in nuclear plant maintenance operations.

9. DEFUELLING OPERATIONS

By far the most tedious and painstaking recovery task has been the protracted defuelling operation. The gruelling task of removing the 140 tonnes of debris began in 1985. The workers, located on a rotating shield platform, manipulated long-handled tools through 12 metres of water to extract the rubble from the vessel and place the pieces in fuel canisters. Throughout the defuelling, the major radiological problems were to minimize the defuelling doses on the platform and to maintain the contamination levels on the platform at acceptable levels. This was particularly difficult, given the continuous movement of defuelling equipment into and out of the reactor vessel.

By the end of 1987, the core region had been cleared. What followed was possibly the single most difficult and arduous task of the cleanup — the cutting of the lower core support assembly to gain access to the fuel debris that had flowed to the lower head and congealed on the bottom of the vessel. To obtain access to the lower head, the massive lower core support assembly consisting of five steel plates had to be cut in pieces. The plates range in thickness from 2.5 to 34 centimetres, with the thickest structure being the lower grid forging. Liquefied material had flowed through the periphery of the structures and had come to rest on the lower head, accumulating to a depth of about 1 metre. The cutting of the core support assembly took over a year, with hundreds of cuts required.

The cutting was accomplished with a plasma arc cutting torch working under twelve metres of water. The torch is an especially modified plasma arc torch operating at temperatures around 50 000° Fahrenheit (27 000°C). The torch's intense heat comes from a plasma of nitrogen gas that is ionized to conduct an electric arc. The metal being cut is melted by the arc and blown away by a secondary gas stream.
Although plasma arc welding and cutting is not new, the technology was utilized under conditions never before encountered. The torch was positioned by computers and guided remotely from the defuelling command centre in the turbine building. The layer-by-layer cutting was painstaking and fraught with mechanical and electrical problems.

The unconventional and unprecedented use of the plasma arc torch is typical of the technological innovations that have been necessitated by the complex nature of the TMI-2 cleanup and defuelling task. This onerous and often frustrating task was finally completed in May 1989 and the pieces removed, providing a 2 metre diameter opening through which to access the remaining 30 tonnes of fuel and structural debris in the lower head.

One of the major radiological problems encountered throughout the dismantlement of the core support assembly has been the increase in discreet radioactive particles, better known as 'hot particles'. Special precautions have been taken to ensure that particles remained in the containment building. This was done by frequent wipe-downs of workers, ventilation controls in the desuiting area and ongoing surveillance. This programme has been largely successful in controlling the spread of hot particles.

The worker dose from the defuelling operation has been about 14 man-sieverts or about 23% of the total recovery collective dose.

The defuelling has been completed and all the fuel is scheduled to be shipped to the repository by the end of 1989.

10. WATER EVAPORATION

One of the few remaining major cleanup tasks is that of disposal of the water generated during the accident. Since 1982, the 2.5 million litres of processed accident generated water have been stored at TMI. The principal radionuclide remaining in the water is tritium. After carefully evaluating the alternatives for disposal, GPU Nuclear chose evaporation as the option which would be most acceptable to the general public. Although of equally innocuous environmental impact, discharge to the river was not chosen because of public perceptions of adverse health effects.

The water evaporation process had to be approved by the NRC with public hearings held to hear the technical arguments in favour of evaporation with opposing reasons by members of the public. In the end, the NRC approved the evaporation process based on the clear evidence of insignificant off-site dose consequences. The estimated dose to a member of the public will be in the order of 1 microsievert over the two year period of operations.

General public acceptance of the water disposal plan was an essential element in obtaining approval of the evaporation process. Although evaporation was not the most expedient or the least costly alternative, the perception of the public regarding
the hazards of radiation necessitated the choosing of the water disposal methodology. The water evaporation process is scheduled to begin in November 1989.

11. WHAT REMAINS

With the completion of defuelling, the cleanup is nearing its conclusion. Several tasks remain to be accomplished before the plant is placed in monitored storage. Beginning later this year, samples will be taken in the lower head of the reactor vessel to assess the structural integrity of the reactor vessel. This study, a joint US and international effort has taken on more significance due to the recent findings of several small 'cracks'. The cracks were not accessible until the fuel rubble in the lower head was removed. Close inspection, with a colour video camera indicates that the cracks are most probably tears in the stainless steel cladding of the vessel and are not likely to have penetrated the vessel itself.

Following some additional decontamination in 1990, the plant will be placed in a stable, secure and monitored condition. There are currently no plans ever to operate the plant again. Full decommissioning is expected to occur at the same time that TMI-1 is decommissioned at the end of its life expectancy.

12. CONCLUSION

With the placement of TMI-2 into monitored storage next year, the cleanup of TMI-2 will be complete. It has been a remarkable journey replete with challenges and frustrations beyond anything that could have been imagined in 1979. Obstacle after obstacle was faced with a sense of dedication and insistence on assuring the health and safety of the workers and the public.

From a radiological protection perspective, basic principles of job planning, training, monitoring, adherence to procedures and supervision were the elements contributing to worker safety and low occupational dose.

All of us in the business of operating nuclear power plants have in some way been touched by TMI. In the area of my professional responsibility, health physics, there have been enormous changes brought about by TMI with significant improvements made in radiation protection performance in the USA.

What remains to be answered is, what is the legacy of TMI. It was David Brinkley, the popular US journalist, who is quoted as saying, ‘News is the first draft of history’. If this be true, TMI is destined to be forever etched in the annals of history. I suspect, however, that the historians will show TMI as an anecdotal blemish in the evolution of nuclear power that caused a temporary setback to the expansion of nuclear power. Regardless of its legacy, TMI has been a critical and painful lesson to the nuclear industry and one that will forever stand as a beacon for caution to those entrusted with its technology.
UNITED STATES NUCLEAR REGULATORY COMMISSION'S REGULATORY OVERSIGHT OF CLEANUP OPERATIONS AT THE THREE MILE ISLAND UNIT 2 STATION (1979-1989)

W.D. TRAVERS
United States Nuclear Regulatory Commission,
Washington, D.C.,
United States of America

Abstract

As a result of the Three Mile Island Unit 2 (TMI-2) accident in March 1979, the United States Nuclear Regulatory Commission (USNRC) established a programme to carry out its regulatory oversight responsibilities during cleanup of the damaged facility. In order to achieve its primary objectives of safety assurance and environmental impact minimization, a dedicated organization was formed to conduct USNRC reviews of the utility’s cleanup planning and implementation. A discussion of USNRC’s regulatory approach regarding specific cleanup activities is presented. Additionally, USNRC activities designed to facilitate the disposal of radioactive wastes, and the dissemination of information to the public are discussed.

1. INTRODUCTION

The recovery of the Three Mile Island Unit 2 (TMI-2) facility began almost immediately following the accident on 28 March 1979. In the more than ten years since the accident, significant progress has been made in completing the cleanup and eliminating the residual threat posed by the post-accident condition of the plant. During this period the USNRC has been faced with unique regulatory challenges which have necessitated innovative approaches towards carrying out the agency’s fundamental responsibility of assuring the protection of public health and safety. As the most important aspect of the cleanup, reactor vessel and reactor coolant system defuelling, is nearing completion the USNRC continues to be actively involved in monitoring the utility’s safe conduct of the ongoing cleanup activities.
For a period of several weeks following the accident, activities at TMI-2 were conducted nearly continuously in an emergency response mode. Resources, including personnel and equipment, were made available from a wide spectrum of sources, including nuclear utilities, nuclear suppliers, the Federal Government, the Commonwealth of Pennsylvania, and the international community. During this period, the USNRC provided regulatory oversight and technical assistance through senior technical management and selected staff who were dispatched to the TMI site from USNRC Headquarters (Washington, D.C.) and several USNRC regional offices across the USA. The USNRC role in these early post-accident weeks involved close monitoring of the utility's efforts to assure the safe shutdown of the facility, information dissemination to the public and government officials, and the co-ordination of activities of all involved Federal agencies. USNRC technical staff at the TMI site also became closely involved in the detailed planning to assure safe shutdown and to initiate cleanup actions. Within a few months of the accident, the USNRC role returned to the more traditional one of evaluating the safety and environmental aspects of the utility's efforts to plan for and carry out the cleanup. Nearly all of the recovery-related activities conducted during the early weeks following the accident were reviewed for approval by USNRC technical staff.

After control of the plant was regained, the initial assessments of plant conditions provided the first indication of the tremendous scope of the cleanup required. The nuclear fuel had been severely damaged, but the degree of damage was unknown. The reactor building (RB) was inaccessible because of the presence of radioactive gases (e.g. 1.6 PBq [43 000 Ci] of 85Kr) in the building atmosphere, and radioactive particulates (e.g. 137Cs and 90Sr) on building/equipment surfaces and in 2600 m³ (700 000 US gallons) of water that had flooded the reactor building basement. Additionally, most areas of the auxiliary building (AB) were highly contaminated as a result of the transfer of contaminated liquids from the RB to the AB and the subsequent overflow of tanks during the accident. The highest exposure rates in the AB were measured in the seal injection valve room, with rates as high as 70 μC kg⁻¹·s⁻¹ (1000 R/h).

In order to provide for effective regulatory oversight of the cleanup, the USNRC implemented a unique programme with three principal objectives: (1) to maintain reactor safety and control of radioactivity, (2) to minimize environmental impacts and assure that radiation exposures to workers and the public were within regulatory limits and were maintained as low as is reasonably achievable (ALARA), and (3) to assure the safe storage and/or disposal of radioactive wastes from cleanup operations. An additional primary objective of this programme was to assure the expeditious cleanup of TMI-2 in order to minimize the risks, to both the public and workers, associated with the damaged facility. In order to meet these goals, a dedicated organization comprised of nuclear engineers and health physicists was
established to review the technical adequacy of detailed cleanup activities and to conduct on-site inspections to assure compliance with USNRC regulations. Organizationally, the TMI Program Office (TMIPo), included senior USNRC management and enjoyed a significant degree of independence to allow for expeditious regulatory decisions. The Director of the TMIPo reported directly to the Director of the Office of Nuclear Reactor Regulation rather than through intermediate levels of agency management. This organization was also unique because it combined in one unit the management and professional staff necessary to carry out safety and environmental reviews as well as the direct inspection of nearly all aspects of cleanup implementation. Within the first three years following the accident, 30 to 40 USNRC staff members, divided equally between the site and Washington, D.C., were assigned to the TMI-2 cleanup. This level of staffing allowed for a high level of USNRC scrutiny of the unique cleanup activities, including the review and approval of the detailed procedures for implementing the cleanup. As the project has progressed the number of USNRC staff and the level of USNRC staff review of detailed cleanup activities has been gradually reduced. Since early 1988, following a major reduction in the number of on-site USNRC staff, the cleanup has been monitored by four on-site personnel who are supported by technical specialists located in Washington, D.C., and King of Prussia, Pennsylvania.

As a consequence of the accident the facility was placed into a condition which had not been specifically analysed at the time TMI-2 was licensed by the USNRC. Additionally, the activities which would need to be performed to decontaminate and defuel the facility had not previously been evaluated by the utility or the USNRC. As a result, and in order to comply with applicable US law and USNRC regulations, a principal responsibility of the USNRC staff during the cleanup has been the conduct of environmental and safety evaluations of major cleanup activities. These evaluations, which include consideration of both worker and environmental safety and which have examined various alternatives to performing cleanup activities, have been carried out to bound conservatively cleanup impacts and to assure conformance with applicable Federal regulations before USNRC approval.

The earliest major USNRC technical evaluations involved utility proposals related to the decontamination of (1) the water transferred from the RB to the AB during the accident [1500 m$^3$ (400 000 gallons); 1.5 Bq/m$^3$ (40 µCi/c$^3$) of $^{137}$Cs, $^{134}$Cs and $^3$H], and (2) the RB atmosphere [1.6 PBq (43 000 Ci) of $^{85}$Kr]. These activities were fundamental to initiating the first major steps of the cleanup and were required to permit the safe conduct of subsequent activities (e.g. defuelling). During USNRC evaluation of both of these activities public interest was substantial, particularly regarding the utility’s proposal to vent the RB atmosphere to the environment, and a number of public meetings were held to provide information and to allow for public comment. In accordance with agency regulations, documented USNRC evaluations were issued in draft and all comments received from Federal and State agencies, and the public were addressed in the final document. Subsequently, both
utility proposals, with some minor modifications, were determined to be safe and environmentally acceptable by the USNRC. In October 1979, decontamination of the AB water began, using a specialized ion exchange system, and in July 1980 the RB atmosphere was vented to the atmosphere to remove the atmospheric contamination.

In 1980, the USNRC staff began its most extensive assessment. The Programmatic Environmental Impact Statement (PEIS) (NUREG-0683) was published in March 1981 [1]. This evaluation, however, was not carried out in response to a specific utility proposal for completing the cleanup. The PEIS was conducted as a USNRC staff initiative in order to (1) provide an assessment of the safety and environmental impacts associated with the entire cleanup and (2) to complete to the extent possible USNRC-required reviews in advance of specific utility proposals, thus expediting the cleanup progress. The activities examined included building and equipment decontamination, reactor and primary system defuelling and decontamination, liquid and solid waste packaging and handling, and transportation and disposal of cleanup wastes. In order to estimate the sensitivity of projected impacts (e.g. radioactive releases, environmental dose, and worker radiation doses), a number of alternatives for each activity were evaluated. The PEIS concluded that the entire cleanup could be completed without significant environmental impacts and within acceptable regulatory limits. Specifically, the major impact was estimated to be incurred by cleanup workers, i.e. 20 to 80 man-Sv, but less than one additional cancer death in the worker population was projected. Although the PEIS has been supplemented three times since its original publication, it has continued to provide a conservative estimate of cleanup impacts and has continued to be relied upon during real time USNRC reviews of cleanup activities. The completion of this bounding document in 1981 facilitated USNRC staff review of specific utility proposals for major cleanup activities (e.g. defuelling) which were submitted subsequently. These specific reviews have been able to reference the environmental evaluations published in the PEIS and have not been subject to the potential delays associated with individual environmental reviews. As a result, USNRC reviews of specific utility proposals for those activities evaluated in the PEIS have not delayed cleanup progress.

The worker radiation protection programme at TMI-2 has been closely scrutinized by the USNRC and, subsequent to weaknesses early in the cleanup, has developed into a highly sophisticated and effective programme. Immediately following the accident, the USNRC became concerned regarding the utility’s ability to manage worker radiation safety. Several unplanned exposures in excess of USNRC regulatory limits for workers reinforced this concern. In September 1979, a special panel of health physics experts from within and without the USNRC was constituted to evaluate the existing radiation safety programme. The panel’s recommendations [2] in December 1979 focused on the need for an upgraded programme with a strong emphasis on training, technical depth, improved dosimetry, and management control. The programme that has developed since then has been successful in
maintaining individual doses below 4 cSv and the average annual worker dose has been comparable to the US industry average for doses at operating plants. Internal doses have been kept to a small fraction of the external dose and are indicative of thorough planning and engineering controls. The radiation protection programme has continued to be a prime focus of USNRC regulatory oversight. Frequent direct inspections by USNRC radiation protection specialists and periodic meetings between utility and USNRC management have been utilized to assure the continuing effectiveness of these programmes.

A unique aspect of the USNRC's regulatory role at TMI-2 during the cleanup has been the large degree of interaction with the public which has been required. As a result of the world attention drawn to the accident and the continuing interest in the cleanup, the USNRC has had a relatively large degree of interaction with the public throughout the cleanup. This interaction has included a great deal of public correspondence and many telephone inquiries, participation in public meetings to discuss specific cleanup activities, response to media inquiries, and briefings for elected Federal, State, and local officials. In recognition of the interest of local citizens in safe cleanup operations, the USNRC established the Advisory Panel for the Decontamination of Three Mile Island Unit 2 in 1980. The panel, which is composed of local citizens, scientists, and representatives of the Commonwealth of Pennsylvania and local governments, has held over 60 public meetings with the utility, the USNRC staff, and the Commission. It has provided recommendations to the Commissioners on the conduct of cleanup, including worker safety and the ongoing defuelling activities. The panel has also provided an important forum for public understanding and information exchange on the details of the cleanup activities. The Panel, with the USNRC staff's support for its activities, has been successful in providing the public with a continuing source of accurate information about the safety of the project.

The management and disposal of radioactive wastes have also posed a significant challenge at TMI-2. As a result of the accident, a large volume of the wastes generated by the cleanup contains relatively high concentrations of fission products and/or transuranics. The decontamination of the approximately 2600 m³ (700 000 US gallons) of water that spilled into the RB basement (7.4 Bq/m³ (200 μCi/c³), for example, was carried out via ion exchange using inorganic zeolite resins. This processing resulted in a number of zeolite containers (0.3 m³ (8 ft³)), loaded with up to 4 PBq (100 000 Ci) of primarily 137Cs, 134Cs, and 90Sr, which were not suitable for disposal in USNRC-licensed low level radioactive waste disposal facilities. In order to facilitate the removal and storage/disposal of these wastes and the TMI-2 nuclear fuel, the USNRC and the United States Department of Energy (USDOE) entered into an agreement [3] that allowed for the USDOE to accept the wastes for research and development purposes, and storage or disposal. Some of the zeolite resins have subsequently been stabilized via vitrification in a USDOE demonstration project and the remainder are part of a Monitored Retrieval
Burial Demonstration Program. Within the agreement between USDOE and USNRC, the damaged TMI-2 core is being transferred by rail to the USDOE’s Idaho National Engineering Laboratory for examination and storage pending ultimate disposal in the future US high level waste repository. In addition to removing a significant amount of non-commercial radiological waste from TMI, the implementation of the USDOE/USNRC agreement has resulted in the successful demonstration of waste management technology for handling, packaging, transporting, and in most cases disposing, of these wastes.

The current focus of the ongoing TMI-2 cleanup is the removal of the damaged nuclear fuel. Since October 1985, nearly 94% of the estimated 140 Mg (300 000 lb) of fuel and structural material has been removed using specialized tools designed specifically for this effort. The tools have included hydraulically operated shovels and grasping devices, a drilling machine used to break apart fused core materials, and a remotely operated plasma arc cutter used to section intact structural assemblies in the lower region of the reactor vessel. All of the core material is being placed in canisters designed to ensure against potential recriticality during its shipment and storage. The safety issues associated with this defuelling approach are unique and have required the utility and the USNRC staff to consider each issue carefully. Specific issues associated with these reviews have included: subcriticality in the reactor vessel and in the canisters used to transport the damaged fuel, the potential pyrophoricity hazard associated with unoxidized Zircaloy if exposed to air, the generation of hydrogen and oxygen by radiolytic decomposition of water during transportation and storage of the damaged fuel, and the potential impact of heavy load drop accidents in the reactor vessel and spent fuel pool. Prior to USNRC approval, and following extensive technical discussion between USNRC and the utility, each of these issues was judged to be satisfactorily resolved. Since initiation of defuelling in October 1985, USNRC staff have reviewed several supplemental defuelling proposals and have continued to monitor the conduct of defuelling at the TMI site. Thus far, the dose incurred by defuelling workers working on a shielded platform atop the reactor vessel has been maintained relatively low. As a result of early decontamination efforts in the reactor building and the design of the defuelling platform and tools, radiation dose rates for defuellers have been maintained at approximately 30 nSv/s (10 mrem/h) on the platform.

The USNRC staff is currently involved with a review of the utility’s proposal to place the TMI-2 facility into storage at the projected completion of defuelling (December 1989). As with most activities associated with the cleanup, this proposal is unique and essentially without precedent. Although similar to the storage (SAFSTOR) approach associated with decommissioning in the USA, the TMI proposal involves storage of a facility significantly more contaminated than undamaged facilities at the end of their operational usefulness [4]. If approved by the USNRC, the facility would be stored (10 to 30 years) before completion of several cleanup activities principally related to the decontamination of large areas in
the RB basement and the removal of residual nuclear fuel from the reactor and the primary system (approximately 1% of the fuel). A primary advantage of the storage approach versus continuing the cleanup to completion is a USNRC-projected worker dose saving of 3000 to 900 person-cSv resulting from radioactive decay and anticipated advancement in robotic decontamination technology. Because of the high dose rates, the decontamination effort carried out to date in the RB basement has involved robotics developed specifically for the TMI-2 cleanup.

3. CONCLUSION

The TMI-2 cleanup, although not yet complete, has accomplished a great deal towards achieving the goal of removing any potential threat to the public, the environment, and the TMI workforce. Most importantly, the cleanup activities to date have been conducted safely with minimal impact on and off the site. USNRC programmes developed specifically for the unique aspects of the cleanup have been successful in helping to assure this record of safety.

REFERENCES


ОБЕСПЕЧЕНИЕ РАДИАЦИОННОЙ
БЕЗОПАСНОСТИ ПРИ СООРУЖЕНИИ ОБЪЕКТА
“УКРЫТИЕ” НА ЧЕРНОБЫЛЬСКОЙ АЭС

А.П. ПАНФИЛОВ, Л.Ф. БЕЛОВОДСКИЙ,
В.И. ГРИШМАНОВСКИЙ
Государственный комитет по использованию
атомной энергии СССР,
Москва,
Союз Советских Социалистических Республик

Abstract–Аннотация

RADIATION SAFETY DURING CONSTRUCTION OF THE ENCAPSULATION AT
THE CHERNOBYL NUCLEAR POWER PLANT.

A review is given of the main radiation safety problems which were solved during
design and construction of the encapsulation for Unit 4 of the Chernobyl nuclear power plant
which was destroyed in the accident of 26 April 1986. Information is given on the conditions
under which large scale restoration work was performed, and on the design stipulations laid
down for construction of the encapsulation for the destroyed unit. The paper discusses the
technical, organizational and health measures which were used to ensure that radiation safety
regulations and standards were observed during construction. The problems of organizing a
radiation safety service inside the construction and assembly organization which built the
encapsulation are discussed. Finally, conclusions are drawn with regard to the experience
which has been gained in the area of radiation safety implementation during large scale post-
accident restoration work under problematic radiation conditions such as those at the
Chernobyl nuclear power plant site.

ОБЕСПЕЧЕНИЕ РАДИАЦИОННОЙ БЕЗОПАСНОСТИ ПРИ СООРУЖЕНИИ
ОБЪЕКТА “УКРЫТИЕ” НА ЧЕРНОБЫЛЬСКОЙ АЭС

В докладе рассматриваются основные вопросы обеспечения радиационной
безопасности, которые были решены при проектировании и в процессе строительства
объекта “Укрытие” на IV энергоблоке Чернобыльской АЭС (ЧАЭС), разрушенном в результате аварии 26 апреля 1986 г. Представлены сведения об условиях проведения крупномасштабных восстановительных работ и о требованиях, предъявляемых к проектным решениям при сооружении “Укрытия” аварийного блока
ЧАЭС. Рассматриваются комплексы организационно-технических и санитарно-
технических мероприятий, обеспечивающих создание “Укрытия” при соблюдении
требований норм и правил радиационной безопасности. Обсуждаются вопросы
организации службы радиационной безопасности в строительно-монтажной орга-
nизации, осуществлявшей сооружение объекта “Укрытие”. Сделаны выводы о
накопленном опыте решения проблемы обеспечения радиационной безопасности в процессе проведения крупномасштабных аварийно-восстановительных работ в тяжелых радиационных условиях на площадке ЧАЭС.

1. ВВЕДЕНИЕ

26 апреля 1986 г. авария на IV энергоблоке Чернобыльской АЭС (ЧАЭС) привела к разрушению активной зоны реакторной установки и части здания энергоблока, в результате чего в окружающую среду было выброшено значительное количество радиоактивных веществ, накопившихся в реакторе за время его эксплуатации, а на площадке ЧАЭС вблизи аварийного блока также были разбросаны высокоактивные осколки топливных элементов и кусков графита из реактора.

Информация об аварии на ЧАЭС, ее причинах и последствиях была рассмотрена на специальном совещании, проведенном МАГАТЭ 25–29 августа 1986 г. в Вене [1]. В итоговом докладе Международной консультативной группы по ядерной безопасности (МКГЯБ), подготовленном по результатам работы совещания, были рассмотрены основные этапы ликвидации последствий аварии (ЛПА) непосредственно на площадке станции по данным, полученным до 1 августа 1986 г. [2].

В 1987 г. советскими авторами были опубликованы материалы о последующем периоде ЛПА, завершившимся созданием объекта "Укрытие" на IV энергоблоке ЧАЭС, который вначале назывался "Саркофагом" [3, 4]. Представляется весьма важным более подробно рассмотреть вопросы, связанные с обеспечением радиационной безопасности при разработке проектных решений объекта "Укрытие" и в процессе его сооружения.

2. НАЗНАЧЕНИЕ ОБЪЕКТА "УКРЫТИЕ" И ОСНОВНЫЕ ПРОБЛЕМЫ ЕГО СОЗДАНИЯ

После решения первоочередных задач на начальных этапах ЛПА, а именно: ликвидации пожара сразу после аварии на IV блоке ЧАЭС и предотвращения горения графита в шахте разрушенного реактора, в середине мая 1986 г. Правительственной комиссией, возглавлявшей работы по ЛПА, было принято решение о долговременной консервации аварийного блока. Это было обусловлено необходимостью предотвращения выхода в окружающую среду радиоактивных веществ из разрушенного блока и защиты территории АЭС от проникающего излучения. Кроме того, необ-
В соответствии с заданием на захоронение разрушенного энергоблока необходимо было обеспечить контроль поведения топливной массы разрушенного реактора и иметь возможность воздействовать на нежелательные процессы.

Задача захоронения разрушенного энергоблока была сложна и уникальна, так как не имела аналогов в мировой инженерной практике. Кроме того, ее предстояло решить в весьма сжатые сроки. Сложность создания подобного сооружения кроме значительных разрушений здания IV энергоблока существенно усугублялась тяжелой радиационной обстановкой в зоне разрушения блока, что делало его труднодоступным и крайне ограничивало использование обычных инженерных решений.

Радиационные разведки, проведенные в подготовительный период (в конце мая — начале июня 1986 г.) в районе разрушенного блока показали, что основным фактором, определяющим радиационную обстановку при сооружении "Укрытия", являлось γ-излучение, мощность дозы которого составляла от сотен мР/ч до сотен Р/ч, а в отдельных точках, где находились обломки элементов активной зоны реактора — до тысяч Р/ч. Загрязнение поверхностей территории и зданий АЭС составляло от $10^4$ до $10^8$ β-частиц (см²·мин) и от $10^5$ до $10^9$ α-частиц (см²·мин), загрязнение воздушной среды радиоактивными аэрозолями в местах проведения работ...
достигало 1–10 допустимых концентраций для различных радионуклидов, установленных "Нормами радиационной безопасности НРБ-76" [5].

Основные проблемы, которые предстояло решить при выборе проектных решений объекта "Укрытие":

1. Объект по своему назначению не являлся ни могильником радиоактивных отходов, ни хранилищем ядерного топлива, а должен быть обслуживаемым объектом, где контролируются и исключаются:
   — возникновение цепной реакции деления;
   — нарушение теплосъема с остатков топлива, которое привело бы к их плавлению;
   — образование взрывоопасной концентрации радиолизного водорода.

2. Необходимо было свести к минимуму время строительства объекта для уменьшения влияния разрушенного реактора на окружающую среду и на площадку АЭС, что позволяло бы в короткие сроки возобновить эксплуатацию остановленных энергоблоков.

3. Сооружение объекта должно производиться с минимальным пребыванием строительного и монтажного персонала в радиационно опасных условиях, с целью уменьшения индивидуальных и коллективных доз облучения.

4. Предстояло обеспечить достаточную прочность и надежность строительных конструкций при использовании, по возможности, уцелевших элементов здания энергоблока и сохранение функций объекта при воздействии различных природных явлений (атмосферные осадки, ураганы и т. п.).

3. ОРГАНИЗАЦИЯ РАБОТ ПО СООРУЖЕНИЮ ОБЪЕКТА "УКРЫТИЕ"

Для решения такой сложной задачи 20 мая 1986 г. была образована специализированная организация — Управление строительства № 605 (УС-605). Эта мощная строительная организация состояла из нескольких строительных районов, возводивших различные элементы "Укрытия", монтажного района, бетонных заводов и управлений; механизации и автотранспорта, энергоснабжения, производственно-технической комплектации. В состав УС-605 входили отдел дозиметрического контроля (ОДК), службы рабочего снабжения, включая столовые, материально-технического и санитарно-бытового обеспечения, а также базы проживания персонала [6]. Подразделения УС-605 дислоцировались непосредственно на территории ЧАЭС, в г. Чернобыле, в г. Иванкове и на ж/д
станции Тетерев Киевской области. Базы проживания и вспомогательные службы размещались на расстоянии 50–100 км от места проведения работ. С учетом сложной радиационной обстановки и необходимости соблюдения требований норм и правил радиационной безопасности был установлен вахтовый метод работы персонала с продолжительностью одной вахты — 2 месяца. Численность одной вахты достигала 10 000 чел., персонал на территории ЧАЭС работал круглосуточно в 4 смены. Все это обусловливало определенные бытовые трудности с размещением, организацией питания и доставкой персонала.

При УС-605 были сформированы бригады научных консультантов, осуществлявших авторский надзор, и проектантов для решения вопросов оперативного проектирования при сооружении объекта «Укрытие».

4. ОСНОВНЫЕ КОНСТРУКТИВНЫЕ РЕШЕНИЯ И ОРГАНИЗАЦИОНАЛЬНО-ТЕХНИЧЕСКИЕ МЕРОПРИЯТИЯ, ОБЕСПЕЧИВШИЕ СОЗДАНИЕ ОБЪЕКТА "УКРЫТИЕ"

С учетом перечисленных выше проблем проектантами было проработано 18 вариантов проекта. В окончательном варианте, реализованном в дальнейшем, максимально использовались в качестве опор объекта «Укрытие» сохранившиеся и частично разрушенные конструкции энергоблока. Это решение позволило существенно сократить сроки строительства и уменьшить расход строительных материалов.

Выбранный проект представлял собой оригинальную объемно-пространственную структуру, образованную со стороны основного завала (северной) тремя каскадно поднимающимися блоками (высотой до 12 м), с уцелевшей (западной) стороны — монолитной, бетонной стеной, усиленной металлическим каркасом с контрфорсами, с внутренних сторон (восточной и южной) — использовались уцелевшие конструкции с дополнительными опорами на завалах и перекрытие металлическими балками (длиной 70 м) (рис.2). Наибольшую сложность представляло создание перекрытия центрального зала IV реактора, оно было обеспечено установкой специальной конструкции в виде моста и двух параллельных металлических балок. Остальная часть объекта «Укрытие» состояла из разделительных стенок: бетонной — между третьим и четвертым энергоблоками (при этом максимально использовались имеющиеся стены) и герметичной металлической — между вторым и третьим энергоблоками. Дополнительная защитная стена была предусмотрена с южной стороны машинного зала, а также дополнительное перекрытие его крыши.

Существенными особенностями организации всех работ по ЛПА были зональность и принцип поэтапного выполнения работ от периферии
к центру, что было вызвано необходимостью уменьшения распространения радиоактивных веществ и снижения облучаемости персонала.

Реализация проектных решений при строительстве объекта "Укрытие" стала возможной в такой сложной радиационной обстановке благодаря комплексу специально разработанных, с учетом мер радиационной защиты персонала, организационно-технических мероприятий, к которым относятся следующие.

1. Широкое использование строительной техники и машин с дистанционным управлением, в том числе по радио и телевидению. Для управления процессом монтажа был создан центральный оперативный пост с телекамерами, соединенный системой связи с выносными подвижными телекамерами, смонтированными непосредственно на стрелах кранов и специальных вышках, установленных в точках максимального обзора. Аналогичным образом с помощью телемониторов и двухсторонней громкоговорящей связи была организована работа на местах с повышенными уровнями излучения (более 10 Р/ч).

2. Разработка специальных технологий производства бетонных работ с дистанционным применением бетононасосной техники и различными методами задержания бетонной смеси с использованием металличес-
Рис. 3. Автосамосвал КРАЗ с защитной кабиной для транспортирования радиоактивных материалов.

...ских и капроновых сетей, мешков с бетоном или щебнем и т. п., а также дистанционного усиления сохранившихся конструкций.

3. Разработка и использование различных радиационно-защитных кабин машин и механизмов (рис. 3) и экранов для проведения работ в полях излучения от единиц до сотен рентген в час, которые имели коэффициенты защиты от излучения 5–3000 [7]. Для выполнения работ и разведок состояния конструкций в особо сложных условиях (в полях, где излучение было свыше 100 Р/ч) были созданы специальные транспортабельные бронекабины, получившие название "батискафы", которые имели коэффициенты защиты до 2000.

4. Разработка и внедрение технологий и технических средств механической деактивации территории и сооружений ЧАЭС. Основная часть территории вокруг разрушенного блока деактивировалась путем удаления разбросанных активных элементов и снятия зараженного поверхностного слоя грунта, в отдельных местах производился пылеотсос, подавление локальных источников осуществлялось путем засыпки щебнем и бетонирования. Общая толщина покрытия составляла около 0,5 м, в некоторых местах до 1,5 м. Большая часть высокоактивных элементов при очистке территории загружалась в контейнеры и сбрасывалась в развал реактора, т. е. во внутренний объем возводимого "Укрытия", снятый ...
ричного оборудования радиационно-защитными кабинами с грейферными захватами на выдвижной стреле и ножом бульдозера, радиоуправляемые бульдозеры DT-250 (СССР), "Камацу" (Япония), фронтальные погрузчики производства ПНР и фирмы "Торо" (Финляндия) и другая строительно-дорожная техника, оснащенная биозащитой рабочего места оператора, установками фильтрации воздуха и аппаратурой теленаблюдения и радиосвязью.

Для дезактивации загрязненных кровель применялись роботизированные дистанционно управляемые механизмы PP, TP, CTP, "Мобот-4" (СССР) и MF-2, MF-3 (ФРГ) (рис.4), а также радиационно защищенные минитракторы, оснащенные либо бульдозерными ножами, либо фрезами, либо грейферными захватами.

5. Применение в процессе монтажных работ кранов большой грузоподъемности фирм "Демаг" и "Либхерр" (ФРГ), оснащенных телекамерами и позволявших монтировать элементы конструкций массой до 160 т на вылетах стрел до 50 м. Внедрены новые методы строповки и расстроповки конструкций без участия человека, а также специальные кондукторы для точной "посадки" конструкций на место.
5. КОМПЛЕКС САНИТАРНО-ТЕХНИЧЕСКИХ МЕРОПРИЯТИЙ
ПО ОБЕСПЕЧЕНИЮ РАДИАЦИОННОЙ БЕЗОПАСНОСТИ ПРИ
ЛИКВИДАЦИИ ПОСЛЕДСТВИЙ АВАРИИ НА ЧАЭС И СООРУЖЕНИИ “УКРЫТИЯ”

Кроме перечисленных выше организационно-технических мероприятий и инженерно-строительных решений, обеспечивающих быстрое создание сооружения “Укрытие” в условиях очень сложной радиационной обстановки в районе проведения работ, в самый начальный период был специально разработан и согласован с органами государственного санитарно-гигиенического надзора Министерства здравоохранения СССР комплекс санитарно-технических мероприятий по обеспечению радиационной безопасности, основные из которых заключались в следующем.

1. На период работ по ЛПА на ЧАЭС на основании положений “Норм радиационной безопасности НРБ-76” и “Основных санитарных правил работы с радиоактивными веществами и другими источниками ионизирующих излучений ОСП-72/80” [5] была установлена суммарная предельная индивидуальная доза внешнего облучения, равная 25 рентген, и контрольные уровни основных факторов радиационной опасности для производственных и жилых зон УС-605, приведенные в табл. 1. Порядок деления промышленных зон рассмотрен в п. 3 данного раздела. Численные значения контрольных уровней были определены с учетом реальной радиационной обстановки, преобладающей опасности воздействия гамма-излучения, необходимости уменьшения распространения радиоактивных веществ из зоны аварии.

   Было установлено, что при достижении предельной индивидуальной дозы внешнего облучения работник освобождался от работ в зоне ЧАЭС и направлялся на медицинское обследование в медсвяч.

   Кроме приведенных в табл. 1 при ЛПА были приняты следующие ограничения радиационного воздействия: спецодежда, белье и обувь направлялись на дезактивацию в спецпрачечную, если мощность дозы гамма-излучения на расстоянии 3-10 см от отдельных предметов не превышала 30 мР/ч или 100 мР/ч от мешка со спецодеждой; при загрязнении вышеуказанных величин эти предметы рассматривались как радиоактивные отходы и направлялись на пункты захоронения; радиоактивное загрязнение столовой посуды не допускалось.

2. Организована и укомплектована специалистами, дозиметрическими приборами, материалами и оборудованием служба радиационной безопасности УС-605, которая обеспечивала в полном объеме радиационный контроль в производственных и жилых зонах УС-605 и контроль облучаемости, а также совместно с проектантами и строителями разраба-
### Таблица 1. Контрольные уровни основных факторов радиационной опасности для работников УС-605 при ликвидации последствий аварии на Чернобыльской АЭС

<table>
<thead>
<tr>
<th>№</th>
<th>Наименование контролируемой величины и объекта контроля</th>
<th>Единицы измерений</th>
<th>Промышленные зоны</th>
<th>За пределами</th>
<th>Жилая зона 30-ти км зоны</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>I</td>
<td>II</td>
<td>III</td>
</tr>
<tr>
<td>1.</td>
<td>Эквивалентная доза внешнего облучения</td>
<td>бэр</td>
<td>25</td>
<td>25</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>за все время работы</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>за день работы</td>
<td></td>
<td>1,0</td>
<td>1,0</td>
<td>-</td>
</tr>
<tr>
<td>2.</td>
<td>Radioактивное загрязнение</td>
<td>мР/ч</td>
<td>2,0</td>
<td>1,0</td>
<td>0,5</td>
</tr>
<tr>
<td></td>
<td>Кожные покровы, нательное белье</td>
<td>(β-част./см²·мин)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Постельные принадлежности, личная одежда, обувь, внутренние поверхности жилых помещений и столовых</td>
<td>mР/ч</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>Специодежда индивидуальной защиты</td>
<td>мР/ч</td>
<td>5,0</td>
<td>3,0</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>Рабочая одежда и обувь</td>
<td>(β-част./см²·мин)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Поверхности помещений постоянного пребывания персонала, внутренние поверхности транспортных средств, кабин машин и механизмов</td>
<td>mР/ч</td>
<td>5,0</td>
<td>3,0</td>
<td>1,0</td>
</tr>
<tr>
<td></td>
<td>Наружные поверхности транспортных средств и механизмов, покрытия дорог</td>
<td>mР/ч</td>
<td>20,0</td>
<td>5,0</td>
<td>1,5</td>
</tr>
</tbody>
</table>
тывала мероприятия по обеспечению и совершенствованию мер радиационной защиты на работах по ЛПА, проводимых УС-605.

6. В целях сокращения разноса радиоактивных веществ транспортом и людьми район отчуждения вокруг ЧАЭС (30-км зона) был разделен на 3 зоны: I — территория, ограниченная границей 30-километровой зоной, в которую входил г. Чернобыль, где размещались центры управления работами по ЛПА, вспомогательные службы, столовые; II — территория с границей на расстоянии около 15 км от ЧАЭС, где размещались основные производственные подразделения, в которых готовились строительные конструкции, материалы, включая бетон и механизмы перед производством работ непосредственно на ЧАЭС; III — промплощадка ЧАЭС с радиусом 3–4 км от станции.

Весь транспорт на выездах из "грязных" в "чистые" зоны подвергался обязательному контролю за уровнем загрязнения радиоактивными веществами и если он превышал установленный контрольный, то направлялся на специальные пункты санитарной обработки. Автобусы, осуществлявшие перевозку людей, были закреплены за соответствующими зонами, пересадка людей производилась на специальных площадках около пунктов санобработки транспорта.

На границе II и III зон осуществлялась перегрузка бетона на специальной эстакаде из "чистых" бетоновозов–миксеров в "грязные", осуществлявшие перевозку бетона на территорию станции.

4. Все работы в районе IV энергоблока ЧАЭС проводились только после контроля радиационной обстановки, определения основных источников излучения и безопасных регламентов работ. На наиболее опасных участках, где мощность дозы гамма-излучения превышала 10 Р/ч, работы осуществлялись по допускам и с по-операционным дозиметрическим контролем.

Радиационная защита персонала осуществлялась: ограничением времени пребывания в радиационно опасных условиях, дистанционным выполнением технологических операций и использованием защитных экранов и укрытий (убежищ). По возможности сокращалось время доставки персонала к месту производства работ путем использования лифтов и подъемников, для провоза работников УС-605 по территории ЧАЭС использовались автобусы, имеющие свинцовые экраны (с коэффициентом защиты, равным 3) и с очисткой поступающего в салон воздуха через фильтры, часть машин имела кондиционеры. Кроме перечисленных в разделе 3 инженерно-технических мероприятий принцип защиты расстоянием реализовывался также путем использования инструмента и приспособлений, отдаляющих человека от локальных источников излучения (захваты, манипуляторы, удлиненные рукава). В качестве экранов, кроме названных выше, широко применялись стенки из мешков с песком,
железобетонных плит, свинцовых кирпичей и временных штор из листового свинца на переносных каркасах.

5. На работах, связанных с образованием пыли, принимались меры по пылеподавлению (полив водой дорог, орошение территории участка работ, увлажнение грунта и т. п.).

6. Все работники, принимавшие участие в ЛПА на ЧАЭС, в обязательном порядке проходили медицинское освидетельствование и инструктаж по вопросам радиационной безопасности, мерам личной гигиены, способам защиты и правилам использования СИЗ в целях снижения радиационного воздействия на организм.

7. В зоне отчуждения ЧАЭС был установлен санитарно-пропускной режим, предусматривавший переодевание, обмыв и принудительный радиационный контроль персонала при выходе из "грязных" зон в "чистые".

8. Все работники, занятые на ЛПА, обеспечивались комплектом основной спецодежды, обувью и индивидуальными средствами защиты органов дыхания — респираторами "Лепесток-200". Кроме того, в зависимости от характера и условий работы они оснащались дополнительными средствами защиты: освинцованными фартуками и поясами и очками для глаз, а также пленочными фартуками, нарукавниками, перчатками, бахилами, фильтрующими противогазами, изолирующими дыхательными аппаратами и костюмами.

Загрязненные спецодежда и спецобувь собирались в санпропускниках и направлялись на дезактивацию в спецпрачечную, а в случаях невозможности дезактивации — на захоронение в специально выделенные места.

9. При входе в столовые и жилые зоны работники УС-605 проходили через дозиметрические посты, где контролировалось загрязнение рук, одежды, обуви и были установлены умывальники и устройства для обмыва обуви. Проход в столовые и жилые помещения при наличии загрязнений свыше установленных контрольных уровней был запрещен.

6. ОРГАНИЗАЦИЯ ДЕЯТЕЛЬНОСТИ СЛУЖБЫ РАДИАЦИОННОЙ БЕЗОПАСНОСТИ УС-605

В целях контроля радиационной обстановки в местах проведения работ и проживания персонала УС-605, контроля его облучаемости и разработки мероприятий по радиационной защите работающих на ЛПА в составе Управления строительства № 605 был образован отдел дозиметрического контроля, который подчинялся заместителю главного инженера УС-605 по радиационной безопасности.
Структурная схема УС-605 представлена на рис.5. В состав ОДК входило три лаборатории: оперативного контроля, бытовой дозиметрии, радиометрии и ремонта приборов и группа обеспечения. Отдел комплексовался квалифицированными специалистами по дозиметрии и защите от ионизирующих излучений командированных в УС-605 из различных предприятий и организаций.

Общая численность ОДК составляла 150-270 человек в различные периоды деятельности УС-605, из которых 40-50% были инженерно-технические работники, остальные — лаборанты-дозиметристы.

Лаборатория оперативного контроля состояла из групп: оперативной радиационной разведки, дозиметрического контроля районов и подразделений УС-605 (в каждом районе и подразделении была закреплена отдельная группа) и индивидуального дозиметрического контроля, основными задачами лаборатории являлись:

— проведение радиационных разведок и составление картограмм полей излучения на территории ЧАЭС и в местах производства работ при сооружении "Укрытия" с целью разработки конкретных проектных решений, планов производства строительно-монтажных работ и принятия мер по радиационной защите персонала;
— контроль индивидуальных доз облучения персонала, включая оперативный (за смену, за операцию) и суммарный (за отдельные периоды работы и за всю вахту в целом), а также обработка результатов контроля на ЭВМ, оформление и выдача справок и отчетов о дозовых нагрузках;
— определение реальной эффективности принятых мер по защите персонала при использовании радиационно-защитных кабин и оборудования, в результате удаления и подавления источников излучения и проведения дезактивации территории, помещений и техники.

Лаборатория бытовой дозиметрии состояла из групп: санпропускников, контроля столовых и мест проживания. Ее основными задачами являлись: контроль уровней гамма-излучения и загрязнения спецодежды, обуви, средств индивидуальной защиты и кожных покровов персонала в санпропускниках, столовых, гостиницах и административно-бытовых помещениях УС-605, а также поверхностей мебели и имущества, контроль загрязнения транспортных средств, оборудования и имущества, отправляемого за пределы 30-км зоны.

Лаборатория радиометрии и ремонта приборов состояла из групп радиометрии и ремонта приборов. Ее основными задачами являлись:

— пробоотбор и радиометрический и спектрографический анализ проб воздуха, воды, почвы, растительности и "мазков", взятых в местах проведения строительно-монтажных работ, в помещениях административно-бытовых, приема пищи и проживания персонала УС-605;

IAEA-SM-316/40 99
РИС. 5. Структурная схема отдела дозиметрического контроля.
— ремонт, проверка и градуировка дозиметрической и радиометрической аппаратуры, используемой ОДК, а также установка и наладка многоканальных систем дистанционного контроля радиационной обстановки на отдельных участках работ при сооружении "Укрытия".

В группу обеспечения входили специалисты по средствам индивидуальной защиты, снабжения, кладовщик, уборщики-дезактиваторщики, секретарь-машинастка. На эту группу были возложены задачи: оснащение ОДК необходимыми приборами, материалами, оборудованием, спецодеждой, спецобувью и средствами индивидуальной защиты; оказание помощи и консультаций управлению материально-технического обеспечения и производственным подразделениям УС-605 по вопросам применения спецодежды и СИЗ.

Деятельность ОДК была регламентирована соответствующим положением об отделе, планами и графиками радиационного контроля. Структура, приборное и методическое оснащение позволяли осуществлять в необходимом объеме радиационный контроль согласно требованиям "Основных санитарных правил... ОСП-72/80" [5] с учетом реальной радиационной обстановки на территории ЧАЭС и в других зонах. Вопросам организации и результатам радиационного контроля при сооружении "Укрытия" будет посвящен отдельный доклад на данном симпозиуме.

В целом результаты радиационного контроля при сооружении "Укрытия" свидетельствуют о том, что принятые комплексные меры по обеспечению радиационной защиты персонала УС-605 позволили свести к минимуму его дозы облучения, несмотря на сложную радиационную обстановку, крупные масштабы и высокие темпы аварийно-восстановительных работ. В основном установленные контрольные уровни факторов радиационной опасности соблюдались. Средняя доза облучения персонала УС-605 составила 8,6 Р, доля работников, получивших дозы свыше установленного предела, равного 25 Р, составила 0,7% от общего числа работников УС-605, причем у большинства из этой группы (численность которой равна 155 человек) превышение допустимого уровня было на 1-3 Р, максимальная доза — 49,2 Р.

7. ЗАКЛЮЧЕНИЕ

В сжатые сроки в условиях сложной радиационной обстановки и связанных с ней организационных, технических и бытовых трудностей задача создания объекта "Укрытие" была успешно решена (рис.6). В ноябре 1986 г. объект был принят государственной комиссией и передан для долговременного обслуживания. Разрушенный четвертый энергоблок ЧАЭС надежно законсервирован с необходимой защитой помещений.
смежного третьего энергоблока и прилегающей территории, предотвра­щен неконтролируемый выход радиоактивных веществ, предусмотрено аварийное гашение процесса в случае возникновения цепной реакции деления, контролируется взрывобезопасность объекта. Тем самым созданы условия для технического обслуживания аварийного блока и вводы в эксплуатацию первых трех энергоблоков АЭС.

Создание объекта стало возможным благодаря успешному решению проектных и организационно-технических проблем, выполнению крупномасштабных строительно-монтажных работ и реализации комплекса мероприятий, обеспечивших соблюдение требований норм и правил радиационной безопасности в процессе аварийно-восстановительных работ.

Опыт работы по ЛПА на ЧАЭС позволил сделать выводы о возможности выполнения крупномасштабных аварийных работ в очаге аварий с тяжелыми радиационными последствиями и о необходимости совершенствования технического оснащения и степени подготовки сил к действиям в аварийных и послеаварийных условиях. В результате в настоящее время в рамках национальных программ, а также совместно со странами-членами СЭВ намечены и проводятся работы по повышению готовности к ведению аварийно-восстановительных работ в случае ядерных аварий. К их числу относятся следующие:
— совершенствование автоматизированных систем контроля радиационной обстановки, приборов радиационной разведки с дистанционным и широкодиапазонным измерением параметров, в том числе бортовых, а также технических средств для индивидуального дозиметрического контроля;

— создание типовых робототехнических устройств и биозащищенной техники, пригодных для работы в зоне радиационной аварии;

— совершенствование методического и программного обеспечения радиационного контроля для аварийных ситуаций и ликвидации их последствий;

— совершенствование средств и методов дезактивации спецодежды и средств индивидуальной защиты применительно к условиям крупной радиационной аварии.

Кроме того, разрабатываются и осуществляются дополнительные организационные меры по повышению готовности объектов атомной энергетики и промышленности к действиям в чрезвычайных ситуациях, ограниченным и ликвидации последствий радиационных аварий, в том числе по следующим направлениям:

— образование специальных аварийно-технических центров и формирований, подготовленных и оснащенных к действиям в чрезвычайных ситуациях, включая аварии на радиационно-опасных объектах;

— установление систем связи и порядка взаимодействия в аварийных ситуациях сил и формирований различных организаций и ведомств, в том числе гражданской обороны, инженерных и химических войск Министерства обороны;

— повышение уровня готовности объектовых и местных сил к действиям в аварийных и экстремальных ситуациях, включая проведение аварийных тренировок на АЭС и других потенциально опасных производствах.

ЛИТЕРАТУРА


RADIATION MONITORING DURING CONSTRUCTION OF THE ENCAPSULATION FOR UNIT 4 OF THE CHERNOBYL NUCLEAR POWER PLANT.

The accident at the Chernobyl nuclear power plant caused high levels of surface contamination by radionuclides (up to $10^8$ dis/min per cm$^2$ for beta radiation, and $10^5$ dis/min per cm$^2$ for alpha radiation), and gamma radiation exposure dose levels in excess of 400 R/h. Moreover, the radiation fields were uneven and inhomogeneous, amongst other things with regard to their spectral characteristics. This is due to the fact that, in addition to dispersed fuel, fragments of the reactor core were also ejected into the buildings and the area surrounding the Chernobyl nuclear power plant. In a situation of this kind, radiation situation monitoring data are highly important and serve as a basis for design decisions, planning of construction and assembly work, and protection of personnel to minimize dose commitments while maintaining maximum speed of work during construction of the encapsulation. The dosimetric monitoring section monitored the radiation situation (gamma radiation dose levels, beta particle fluxes, contamination of surfaces and the air, exposure of personnel, etc.). Both traditional and specially developed methods were used to monitor the radiation situation, enabling the measurement of radiation risk factors, the determination of space-angular distribution of gamma radiation, and the detection of local contamination sources. Radiation situation monitoring results showed that 75–80% of the gamma radiation was coming from nuclear fuel in the plant compound and not “streamings” from the wreck of the Unit 4 reactor.

The area has been covered with a protective layer (~0.3 m crushed stone and ~0.3 m concrete) thus reducing the gamma radiation levels by 7–20 times. After the encapsulation had been erected, gamma radiation levels in the vicinity of Unit 4 decreased by a factor of approximately 100. The concentration of radioactive aerosols at the work sites while the encapsulation was under construction was, at most, ten times the permitted concentration (PCA), and only when certain operations were being performed which raised a lot of dust did it reach 100–300 PCA. Owing to the high levels of gamma radiation, the danger of external irradiation of personnel was significantly greater than the danger from internal irradiation. Therefore staff were monitored individually for gamma radiation. A permissible dose level of 25 R for the whole period of work (1–2 months) was implemented for the purpose of individual dosimetric monitoring, and a control level of 1 R per shift. The mean exposure dose received by personnel directly involved in the construction of the encapsulation was 8.6 R,
106 БЕЛОВОДСКИЙ и др.

and 50.6% of staff did not receive a dose of more than 5.0 R; radionuclide content in the body (zirconium-90, niobium-95, ruthenium-103, caesium-134 and 137) did not exceed 0.3 of the permissible level.

РАДИАЦИОННЫЙ КОНТРОЛЬ ПРИ СООРУЖЕНИИ "УКРЫТИЯ" ЧЕТВЕРТОГО ЭНЕРГОБЛОКА ЧЕРНОБЫЛЬСКОЙ АЭС.

Авария на Чернобыльской АЭС характеризовалась высокими уровнями загрязнения поверхностей радионуклидами (до $10^6$ расп/мин·см² по бета-излучению и в $10^4$ расп/мин·см² по альфа-излучению) и мощностей экспозиционной дозы, гамма-излучения свыше 400 Р/ч. При этом отмечалась неравномерность и неоднородность полей излучения, в том числе и по спектральным характеристикам. Это обусловлено тем, что на территорию в помещении ЧАЭС были выброшены, помимо диспергированного топлива, фрагменты активной зоны реактора. В этих условиях важное значение приобретают данные контроля радиационной обстановки, которые служат основой проектных решений, планов производства строительно-монтажных работ и способов защиты персонала с целью обеспечения минимальных дозовых затрат при максимальных темпах сооружения "Укрытия" IV энергоблока. Контроль радиационной обстановки (мощностей доз гамма-излучения, потоков beta-частиц, загрязнение поверхностей и воздуха, доз облучения персонала и др.) осуществлялся отделом дозиметрического контроля. При контроле радиационной обстановки использовались традиционные и специально разработанные методики, позволяющие измерять факторы радиационной опасности, а также определять пространственно-угловое распределение гамма-излучения и локальные источники загрязнения. Анализ результатов контроля радиационной обстановки показал, что 75...80% мощности дозы гамма-излучения обусловлено ядерным топливом на территории ЧАЭС, а не "прострелами" из развалов четвертого реактора. На территории ЧАЭС создан защитный слой (~ 0,3 м щебень и ~ 0,3 м бетон), что позволило снизить уровни гамма-излучения в 7...20 раз. После сооружения "Укрытия" уровни гамма-излучения на территории четвертого энергоблока снизились примерно в 100 раз. Концентрация радиоактивных аэрозолей в зонах работ при сооружении "Укрытия" составляла до 10 допустимых концентрации ДКД и лишь при выполнении отдельных пылящих операций достигала 100-300 ДКД. При высоких уровнях гамма-излучения опасность внешнего облучения персонала была значительно выше по сравнению с внутренним облучением. Поэтому индивидуальный контроль персонала проводился по гамма-излучению. При проведении индивидуального дозиметрического контроля руководствовались значениями допустимой дозы 25 Р за весь период работы (1...2 месяца) и контрольного уровня 1 Р за смену. Средняя доза облучения персонала непосредственно занятого сооружением "Укрытия" составила 8,6 Р, при этом у 50,6% персонала дозы облучения не превышали 5,0 Р. Содержание радионуклидов (циркония-95, ниобия-95, рутения-103, цезия-134 и 137) в организме работников не превышало 0,3 допустимого содержания.

1. ВВЕДЕНИЕ

Авария на Чернобыльской АЭС характеризовалась высокими уровнями загрязнения поверхностей радионуклидами и мощностей экспозиционной до-
зы (МЭД) гамма-излучения свыше 400 В/ч. При этом отмечалась неопределенность основных радиационных параметров: защитных свойств зданий (конфигурация, мощность, спектр излучения), характеристик источников излучения, поскольку на территорию станции и в помещения III и IV блоков были выброшены значительные количества ядерного топлива и фрагменты активной зоны реактора [1, 2, 3]. Поэтому практически была невозможна расчетная оценка МЭД и парциальных вкладов в дозы от различных источников излучения.

В этих условиях важное значение приобретают результаты радиационной разведки и контроля, так как они являются основой для принятия проектных решений, разработки планов производства строительно-монтажных работ и способов защиты персонала с целью обеспечения минимальных дозовых затрат.

Учитывая важность результатов радиационной разведки, отдел дозиметрического контроля УС-605 (ОДК) в начальный период сосредоточил основное внимание на обследовании территории и помещений вокруг разрушенного IV блока. Были составлены подробные (с указанием локальных источников) картограммы полей излучения в целом по АЭС и в районах проведения работ и проживания персонала. На основании полученных данных разработаны способы защиты персонала, выбраны безопасные места для проведения вспомогательных работ (монтаж грузоподъемной техники, укрупненная сборка металлоконструкций, районы выжидания для бетоновозов и др.), найдены и обозначены знаками безопасные проходы и подъезды к местам производства работ.

В аварийных условиях ЧАЭС выполнение основных задач ОДК существенно осложнялось высокими уровнями (более 400 Р/ч) проникающего излучения, наличием нерадиационных факторов опасности (обрушение строительных конструкций и инженерных коммуникаций), большими объемами работ по дозиметрическому контролю, практическим отсутствием штатных методик и серийно выпускаемых приборов радиационной разведки применительно к аварийным ситуациям.

Указанные обстоятельства потребовали от ОДК дополнительных методических исследований и принятия ряда принципиально новых решений по регламенту контроля радиационной обстановки (РО).

2. АППАРАТУРА И МЕТОДЫ РАДИАЦИОННОГО КОНТРОЛЯ

Для выполнения возложенных задач и функций отдел ДК был оснащен аппаратурой, характеристики которой приведены в табл. I. Радиационный
<table>
<thead>
<tr>
<th>Назначение прибора</th>
<th>Тип прибора</th>
<th>Диапазон измерений</th>
<th>Энергетический диапазон, МэВ, МэВ</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Индивидуальный контроль дозы гамма-облучения</td>
<td>1. ДК-02</td>
<td>0,02...0,20 Р</td>
<td>0,20...2,00</td>
</tr>
<tr>
<td></td>
<td>2. (Д-2Р) с пультом</td>
<td>0,20...2,00</td>
<td></td>
</tr>
<tr>
<td></td>
<td>УИ-27'</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>ЛКП-50А (комплект)</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>ДП-22В</td>
<td>2,00...50,00 Р</td>
<td>0,20...2,00</td>
</tr>
<tr>
<td></td>
<td>ИКС-А</td>
<td>0,20...10^3 Р</td>
<td>0,05...6,00</td>
</tr>
<tr>
<td></td>
<td>ДПГ-03 с пультом УШ-02</td>
<td>0,10...10^4 Р</td>
<td>0,06...1,25</td>
</tr>
<tr>
<td>2. Индивидуальные пороговые сигнализаторы гамма-излучения</td>
<td>ДРГС-01</td>
<td>6 порогов в диапазоне</td>
<td>0,05...3,00</td>
</tr>
<tr>
<td></td>
<td>ЛКС-04 &quot;Стриж&quot;</td>
<td>3×10^-3, 3,3×10^-1 Р/ч</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>9 порогов в диапазоне</td>
<td>0,05...3,0</td>
</tr>
<tr>
<td></td>
<td></td>
<td>10^-4...1,5×10^-1 Р/ч</td>
<td></td>
</tr>
<tr>
<td>3. Радиационная разведка и оперативный контроль</td>
<td>СРГ-68</td>
<td>0...3×10^-1 Р/ч</td>
<td>0,03</td>
</tr>
<tr>
<td>(измерение мощности дозы гамма-излучения)</td>
<td>ДРГЗ-03</td>
<td>3,6×10^-4...3,6 Р/ч</td>
<td>0,02...3,00</td>
</tr>
<tr>
<td></td>
<td>ДРГ-05</td>
<td>3,6×10^-4...36 Р/ч</td>
<td>0,04...10,0</td>
</tr>
<tr>
<td></td>
<td>ДРГ-01Т</td>
<td>10^-4...10^3 Р/ч</td>
<td>0,10...3,00</td>
</tr>
<tr>
<td></td>
<td>ДП-2В</td>
<td>5×10^-4...2×10^3 Р/ч</td>
<td>0,08...1,25</td>
</tr>
<tr>
<td></td>
<td>КРБГ-1</td>
<td>2,5×10^-4...3×10^3 Р/ч</td>
<td>0,10...1,25</td>
</tr>
<tr>
<td></td>
<td>КЛГ-1</td>
<td>10^-4...10^3 Р/ч</td>
<td>0,08...1,25</td>
</tr>
<tr>
<td>4. Дистанционный (до 300 м) контроль мощности дозы (гамма-излучения)</td>
<td>УИМ-2еM с датчиками УСИТ и БДМГ</td>
<td>0,10...2×10^3 Р/ч</td>
<td>0,06...1,25</td>
</tr>
</tbody>
</table>
5. Контроль нейтронного излучения

| Образец | Источник | Норма 
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>КРАН-1</td>
<td>$10^{10}$ н/с•см$^2$</td>
<td>0,10</td>
</tr>
<tr>
<td>КДН-2</td>
<td>$5 \times 10^{-9} \ldots 5 \times 10^{-3}$ бэр/с</td>
<td>2,5 $\times 10^{-8}$ \ldots 20,0</td>
</tr>
<tr>
<td>ДЭДНИ</td>
<td>0,1 мбэр/ч</td>
<td></td>
</tr>
</tbody>
</table>

6. Оперативный контроль загрязнения поверхностей бета-активными нуклидами

| Образец | Источник | Норма 
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>РУП-1</td>
<td>$10^{5} \ldots 5 \times 10^{4}$ част/мин•см$^2$</td>
<td>0,07</td>
</tr>
<tr>
<td>ЭП-5</td>
<td>$5 \times 10^3$ част/мин•см$^2$</td>
<td>-</td>
</tr>
<tr>
<td>КРВГ-1</td>
<td>$2.5 \ldots 2.5 \times 10^4$ част/мин•см$^2$</td>
<td>0,10</td>
</tr>
<tr>
<td>КРБ-1</td>
<td>$5 \ldots 5 \times 10^6$ част/мин•см$^2$</td>
<td>0,07 \ldots 0,08</td>
</tr>
<tr>
<td>СПАР</td>
<td>$1 \ldots 1,5 \times 10^2$ част/мин•см$^2$</td>
<td>4,2</td>
</tr>
<tr>
<td>РУП-1</td>
<td>$1 \ldots 2 \times 10^4$</td>
<td>то же</td>
</tr>
<tr>
<td>КРА-1</td>
<td>$1 \ldots 10^4$ расп/мин•см$^2$</td>
<td>-</td>
</tr>
</tbody>
</table>

7. Самоконтроль бета-загрязнения (пороговые сигнализаторы)

| Образец | Норма 
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>СЗБ-03, СЗБ-04, СЗБ-05</td>
<td>$10 \ldots 2000$ част/мин•см$^2$</td>
</tr>
</tbody>
</table>

8. Контроль концентрации альфа- и бета-активных нуклидов в воздухе

<table>
<thead>
<tr>
<th>Образец</th>
<th>Норма</th>
</tr>
</thead>
<tbody>
<tr>
<td>КРК-1 с пробозаборным устройством</td>
<td>Чувствительность $2 \times 10^{-5}$ част/с•л</td>
</tr>
</tbody>
</table>

9. Спектрометрические измерения для определения радиоизотопного состава в различных пробах (воздух, вода, почва, растительность)

<table>
<thead>
<tr>
<th>Образец</th>
<th>Чувствительность</th>
</tr>
</thead>
<tbody>
<tr>
<td>Гамма-спектрометр Nokia</td>
<td>$2 \times 10^{-2}$ Бк, разрешение $3,5$ кэВ по $^{135}$Cs (0,66 МэВ)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Образец</th>
<th>Загрузка 5 $\times 10^3$ имп/с</th>
</tr>
</thead>
<tbody>
<tr>
<td>Альфа-спектрометр СЭА-01 с детектором ДКПед-25-1</td>
<td>4,5 \ldots 8,0</td>
</tr>
</tbody>
</table>

1 Сбор, хранение и обработка информации с помощью ЭВМ.
контроль проводили по методикам, изложенным в работах \[4, 5\], а также по методикам, разработанным применительно к специфике обстановки на ЧАЭС: определения пространственно-углового распределения гамма-излучения, отыскания локальных источников излучения и др. Эти измерения проводили с использованием стандартной аппаратуры (ДП-5В, СРП-68) или дозиметров (ИКС-А, ДПГ-03) с набором различных коллиматоров, разработанных в ОДК. Одна из методик, предназначенная для получения информации о доминирующих источниках в пространстве, применялась в июне—июле 1986 г. на территории ЧАЭС. В качестве "коллиматора" использовали бронетранспортер, на гранях которого (перед—зад, левый—правый борт, верх—низ) укрепляли дозиметры ИКС-А. Стальной корпус и свинцовая защита бронетранспортера обеспечивают со стороны каждой из шести граней измерение интегральной дозы от соответствующего полупространства. Это позволяет с угловым разрешением \(2\pi\) стерадиан выделять направления на основные группы источников с оценкой их относительного вклада в мощность дозы.

Измерения показали, что 75-80% МЭД излучения обусловлено топливом, разбросанным по поверхности территории, а не "прострелами" из разрушенного реактора (такое мнение утвердили на ЧАЭС с первых дней аварии). В связи с этим ОДК было предложено создать защитный экран путем засыпки щебнем (~ 30 см) с последующей заливкой бетоном (~ 30 см) территории, примыкающей к IV блоку. Реализация этих предложений привела к снижению МЭД в 7-20 раз (от 180-430 Р/ч до 10-25 Р/ч).

Проведение разведки в опасных и труднодоступных местах требовало доработки приборов: удлинили (до 15-20 м) кабели, соединяющие детекторы с измерительным пультом, уменьшили постоянную времени установления стрелки прибора (для сокращения времени измерения в полях с высокими уровнями МЭД), оборудовали детекторы штангами, позволяющими выносить детектор на 3-4 м. После доработки приборы градуировали с помощью эталон-источников. Дорабатывали также аппаратуру, используемую для дистанционного (до 300 м) контроля МЭД гамма-излучения (табл. I).

Концентрацию радионуклидов в воздухе (с точки пробыотбора приведены на рис. 1) определяли аспирационным методом, путем отбора проб воздуха на фильтры типа АФА-РМП-20 с последующим радиометрическим и спектрометрическим анализом проб на фильтрах.

Индивидуальный дозиметрический контроль (ИДК) проводили с помощью дозиметров-накопителей (ИКС-А или ДПГ-03), которыми были обеспечены все работники УС-605. Для оперативного контроля облучения персонал обеспечивался дозиметрами Д-2 (Д-2Р) на рабочем месте при выдаче
РИС. 1. Фиксированные точки радиационного контроля на территории Чернобыльской АЭС.

задания на работу (рис. 2). Учет доз по Д-2 осуществлялся ежесменно или пооперационно, то есть несколько раз за смену. Сведения об облучаемости персонала систематизировались и хранились в памяти ЭВМ ДВК-3.

Уже на начальном этапе ИДК было отмечено расхождение показаний дозиметров ИКС-А (ДПГ-03) и Д-2 (Д-2Р). Анализ результатов контроля показал, что в идентичных условиях облучения на ЧАЭС показания дозиметров Д-2 и ИКС-А различаются в 1,81 ± 0,48 раза. Очевидно, что причину подобных расхождений следовало искать в энергетической зависимости дозовой чувствительности ("ходе с жесткостью") используемых дозиметров. Поэтому определяли спектральные характеристики излучения в различных местах проведения работ. Результаты измерений (рис. 3) показали, что реальные спектры гамма-излучения существенно "мягче" (~ 100 кэВ), чем следовало ожидать от излучения рассеянного топлива (0,55-0,70 МэВ).

На основании предварительной информации были условно определены три характерных типа радиационных полей, различающихся по жесткости спектра гамма-излучения:
— "жесткие" ($E_\gamma \sim 0.5$ МэВ), типичные для рабочих мест, на которых основной вклад в МЭД дает излучение "прямой видимости" (пол и крыша машинного зала, помещения с загрязненными поверхностями, локальные источники);
— "средние" ($E_\gamma \sim 0.2$ МэВ), типичные для открытых площадок на грунте с объемно распределенными радиоактивными веществами;
— "мягкие" ($E_\gamma \sim 0.1$ МэВ), типичные для рабочих мест, на которых основной вклад дают источники, находящиеся за защитой из легких материалов (щебень, грунт, бетон).

Проводили сличение дозиметров Д-2 (Д-2Р), ИКС-А и ДПГ-03 в полях излучений указанных трех типов, при этом МЭД измеряли приборами ДП-5В и ДРГЗ-03. В качестве эталонного прибора использовался ДРГЗ-03 с ткань-эквивалентным детектором NaJ (Т1) и низким энергетическим порогом регистрации $E_{пор.} = 20$ кэВ).

Полученные результаты (табл. II) свидетельствуют о том, что для контроля дозы облучения по Д-2 следует вводить корректирующие коэффициенты. По мере сооружения "Укрытия" изменялись характеристики полей гамма-излучения, влияющие на показания дозиметров. Поэтому проводилось периодическое сличение дозиметров с целью уточнения коэффициентов коррекции.
РИС. 3. Спектры гамма-излучения при сооружении "Укрытия": а — открытый грунт (на 10.07.86 г.); б — вклад в МЭД энергетических компонент над перекрытием IV блока (на 04.10.86 г.)
ТАБЛИЦА II. ПОКАЗАНИЯ ПРИБОРОВ ОТНОСИТЕЛЬНО ДРГЗ-03 В ПОЛЯХ ИЗЛУЧЕНИЯ (0,6-1,5 Р/ч) РАЗЛИЧНОЙ "ЖЕСТКОСТИ" СПЕКТРА

<table>
<thead>
<tr>
<th>Место измерений</th>
<th>Показания приборов, Р/ч</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Тип прибора</td>
</tr>
<tr>
<td></td>
<td>ДП-5П</td>
</tr>
<tr>
<td>1. На грунте со слоем бетона и шебня</td>
<td>(E_γ ~ 0,1 МэВ)</td>
</tr>
<tr>
<td></td>
<td>1,50 ± 0,40</td>
</tr>
<tr>
<td>2. На открытом грунте</td>
<td>(E_γ ~ 0,2 МэВ)</td>
</tr>
<tr>
<td></td>
<td>1,50 ± 0,40</td>
</tr>
<tr>
<td>3. В помещениях блока &quot;В&quot;</td>
<td>(E_γ ~ 0,5 МэВ)</td>
</tr>
<tr>
<td></td>
<td>1,13 ± 0,14</td>
</tr>
<tr>
<td>4. В машинном зале</td>
<td>(E_γ ~ 0,5 МэВ)</td>
</tr>
<tr>
<td></td>
<td>1,07 ± 0,14</td>
</tr>
</tbody>
</table>

Непрерывные измерения в машинном зале ЧАЭС (с 20.06.86. по 14.07.86 г.) показали, что при МЭД гамма-излучения 0,40-0,76 Р/ч уровни нейтронного излучения составляют 0,0004-0,008 бэр/ч, то есть вклад нейтронов в полную дозу не превышает 2%. Поэтому при проведении ИДК вклад нейтронов можно не учитывать.

Поступление радионуклидов в организм оценивали путем измерения активности респираторов-лепестков и мазков из пазух носа, после выполнения работ в атмосфере с известной концентрацией радионуклидов в течение фиксированного отрезка времени [5, 6]. Для контроля содержания радионуклидов в организме, часть персонала получивших дозу внешнего облучения 20 Р и выше, направлялась в лаборатории Минздрава СССР для обследования на счетчике излучения человека (СИЧ) или определения радионуклидов по результатам анализа биосубстратов [5].

Особенностью работ по сооружению "Укрытия", наряду с высокими уровнями излучения в зоне основных работ, являлась большая протяженность радиоактивного источника [7]. Это привело к необходимости контроля РО не только на самой станции, но и в жилой зоне, учитывать дозы излучения, получаемые персоналом в пути следования к станции и непосредственно к рабочему месту.
ТАБЛИЦА III. ИЗМЕНЕНИЕ МОЩНОСТЕЙ ДОЗ НА ТЕРРИТОРИИ ЧЕРНОБЫЛЬСКОЙ АЭС, Р/ч ВО ВРЕМЕНИ

Мощности доз, Р/ч

<table>
<thead>
<tr>
<th>№ точки на рис. 1</th>
<th>10.06.</th>
<th>25.07.</th>
<th>15.09</th>
<th>25.10</th>
<th>20.11.</th>
<th>01.02.</th>
<th>29.06.</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>5,6</td>
<td>0,7</td>
<td>0,2</td>
<td>0,07</td>
<td>0,03</td>
<td>0,03</td>
<td>0,015</td>
</tr>
<tr>
<td>2</td>
<td>5,6</td>
<td>2,8</td>
<td>0,2</td>
<td>0,07</td>
<td>0,03</td>
<td>0,03</td>
<td>0,02</td>
</tr>
<tr>
<td>3</td>
<td>7,0</td>
<td>1,5</td>
<td>0,2</td>
<td>0,07</td>
<td>0,03</td>
<td>0,03</td>
<td>0,018</td>
</tr>
<tr>
<td>4</td>
<td>4,8</td>
<td>1,7</td>
<td>0,4</td>
<td>0,20</td>
<td>0,15</td>
<td>0,07</td>
<td>0,03</td>
</tr>
<tr>
<td>5</td>
<td>50</td>
<td>2,0</td>
<td>1,2</td>
<td>0,80</td>
<td>0,30</td>
<td>0,26</td>
<td>0,10</td>
</tr>
<tr>
<td>6</td>
<td>20</td>
<td>5,0</td>
<td>2,0</td>
<td>0,80</td>
<td>0,30</td>
<td>0,15</td>
<td>0,10</td>
</tr>
<tr>
<td>7</td>
<td>30</td>
<td>4,0</td>
<td>2,0</td>
<td>0,80</td>
<td>0,30</td>
<td>0,19</td>
<td>0,06</td>
</tr>
<tr>
<td>8</td>
<td>4,0</td>
<td>2,5</td>
<td>1,5</td>
<td>0,10</td>
<td>0,10</td>
<td>0,10</td>
<td>0,03</td>
</tr>
<tr>
<td>9</td>
<td>25</td>
<td>10</td>
<td>2,0</td>
<td>1,00</td>
<td>0,30</td>
<td>0,15</td>
<td>0,10</td>
</tr>
<tr>
<td>10</td>
<td>40</td>
<td>15</td>
<td>4,0</td>
<td>1,00</td>
<td>0,30</td>
<td>0,21</td>
<td>0,15</td>
</tr>
<tr>
<td>11</td>
<td>300</td>
<td>40</td>
<td>5,0</td>
<td>2,00</td>
<td>1,50</td>
<td>0,24</td>
<td>0,40</td>
</tr>
<tr>
<td>12</td>
<td>200</td>
<td>55</td>
<td>2,5</td>
<td>1,50</td>
<td>0,80</td>
<td>0,38</td>
<td>0,18</td>
</tr>
<tr>
<td>13</td>
<td>400</td>
<td>180</td>
<td>3,0</td>
<td>1,50</td>
<td>0,80</td>
<td>0,50</td>
<td>0,30</td>
</tr>
<tr>
<td>14</td>
<td>150</td>
<td>80</td>
<td>3,5</td>
<td>1,50</td>
<td>1,50</td>
<td>0,80</td>
<td>0,25</td>
</tr>
<tr>
<td>15</td>
<td>400</td>
<td>70</td>
<td>3,0</td>
<td>1,50</td>
<td>1,50</td>
<td>0,90</td>
<td>0,40</td>
</tr>
<tr>
<td>16</td>
<td>180</td>
<td>6,0</td>
<td>3,0</td>
<td>1,50</td>
<td>1,50</td>
<td>1,00</td>
<td>0,20</td>
</tr>
<tr>
<td>17</td>
<td>20</td>
<td>3,0</td>
<td>1,7</td>
<td>1,50</td>
<td>1,50</td>
<td>0,60</td>
<td>0,15</td>
</tr>
<tr>
<td>18</td>
<td>4,5</td>
<td>4,0</td>
<td>1,5</td>
<td>0,40</td>
<td>0,30</td>
<td>0,10</td>
<td>0,03</td>
</tr>
</tbody>
</table>

3. РЕЗУЛЬТАТЫ РАДИАЦИОННОГО КОНТРОЛЯ

3.1. Параметры проникающей радиации

Полученные данные дозиметрического контроля свидетельствовали о преобладающей опасности для персонала внешнего облучения гамма-квантами. Параметры радиоактивного источника в процессе выполнения работ по сооружению "Укрытия" изменялись за счет выполнения технических мероприятий и естественного распада радионуклидов (табл. III). Оценки показывают, что за счет распада МЭД гамма-излучения на территории ЧАЭС за период с 01.06.86. по 01.12.86 г. снизилась в ~ 5 раз, а через год после аварии МЭД гамма-излучения уменьшилась в 55 раз [2, 8].
ТАБЛИЦА IV. МОЩНОСТИ ДОЗ ГАММА-ИЗЛУЧЕНИЯ НА УЧАСТКАХ РАБОТ IV БЛОКА В 1986 г., Р/ч

<table>
<thead>
<tr>
<th>Отметка, м</th>
<th>Июль</th>
<th></th>
<th>Сентябрь</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Среднее</td>
<td>Максимум</td>
<td>Среднее</td>
</tr>
<tr>
<td>+1,0</td>
<td>0,1-0,2</td>
<td>15</td>
<td>0,017-0,02</td>
</tr>
<tr>
<td>+5,8</td>
<td>0,2-2,0</td>
<td>100</td>
<td>0,01-0,035</td>
</tr>
<tr>
<td>+10,00</td>
<td>0,15-5,0</td>
<td>50</td>
<td>0,018-0,05</td>
</tr>
<tr>
<td>+16,40</td>
<td>0,10-3,5</td>
<td>50</td>
<td>0,1-0,3</td>
</tr>
<tr>
<td>+19,50</td>
<td>0,25-5,0</td>
<td>200</td>
<td>0,05-0,19</td>
</tr>
<tr>
<td>+24,27</td>
<td>0,25-3,0</td>
<td>50</td>
<td>0,025-0,2</td>
</tr>
<tr>
<td>+27,70</td>
<td>0,7-3,0</td>
<td>40</td>
<td>0,05-0,4</td>
</tr>
<tr>
<td>+31,50</td>
<td>2,5-5,0</td>
<td>52</td>
<td>0,5-1,1</td>
</tr>
<tr>
<td>+35,50</td>
<td>5,0-15,0</td>
<td>15</td>
<td>0,1-0,3</td>
</tr>
<tr>
<td>+43,00</td>
<td>-</td>
<td>-</td>
<td>1-10</td>
</tr>
</tbody>
</table>

Из табл. III видно улучшение РО на территории ЧАЭС, которое сопровождалось снижением доз облучения персонала за время следования к месту производства работ.

Вместе с тем непосредственно в зоне работ при возведении элементов "Укрытия" существенного снижения уровней излучения не происходило, а в ряде случаев наблюдался рост МЭД. Так при возведении разделительной стенки в машинном зале в начале работ на отметке — 4,20 МЭД гамма-излучения не превышала 0,5 Р/ч. При выходе на отметку 0,00 МЭД увеличилась до 2,5-5,0 Р/ч, затем, по мере подъема стенки и приближения ее к перекрытию машинного зала, уровни излучения возросли до 10-20 Р/ч. Аналогичная ситуация наблюдалась и при сооружении разделительной стенки в блоке В (табл. IV), а также при возведении каскадной стенки, поскольку происходило постепенное приближение рабочих мест к источнику излучения. После завершения строительства "Укрытия" (рис. 4), уровни излучения на каскадной стенке и вокруг нее также снизились.

3.2. Загрязнение воздушной среды

Опасность внутреннего облучения персонала была связана с загрязнением альфа-бета-активными нуклидами поверхностей и воздуха в зоне работ.
РИС. 4. Картограммы полей гамма-излучения после завершения строительства "Укрытия" (на 02.11.86 г.) в Р/ч:

а — на поверхности каскадной стенки;

б — на перекрытии блока.
### Таблица V.

Концентрация радионуклидов в воздухе при сооружении "Укрытия". В 1986 г.

<table>
<thead>
<tr>
<th>Точка контроля по рис. 1</th>
<th>Бета-активные</th>
<th>Альфа-активные</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>июль</td>
<td>сентябрь</td>
</tr>
<tr>
<td>Т-111</td>
<td>2.8 × 10^{-12}</td>
<td>2.0 × 10^{-12}</td>
</tr>
<tr>
<td>Т-112</td>
<td>2.2 × 10^{-11}</td>
<td>4.1 × 10^{-12}</td>
</tr>
<tr>
<td>Т-113</td>
<td>3.1 × 10^{-12}</td>
<td>4.5 × 10^{-14}</td>
</tr>
<tr>
<td>Т-121</td>
<td>6.5 × 10^{-13}</td>
<td>2.4 × 10^{-13}</td>
</tr>
<tr>
<td>Т-122</td>
<td>1.4 × 10^{-13}</td>
<td>4.0 × 10^{-14}</td>
</tr>
<tr>
<td>Т-123</td>
<td>5.0 × 10^{-14}</td>
<td>7.0 × 10^{-14}</td>
</tr>
<tr>
<td>Т-131</td>
<td>5.1 × 10^{-13}</td>
<td>4.5 × 10^{-14}</td>
</tr>
<tr>
<td>Т-132</td>
<td>2.9 × 10^{-12}</td>
<td>8.0 × 10^{-14}</td>
</tr>
<tr>
<td>Т-141</td>
<td>3.5 × 10^{-13}</td>
<td>3.3 × 10^{-14}</td>
</tr>
<tr>
<td>Т-142</td>
<td>1.4 × 10^{-13}</td>
<td>3.8 × 10^{-15}</td>
</tr>
<tr>
<td>Т-161 (блок В)</td>
<td>4.5 × 10^{-12}</td>
<td>-</td>
</tr>
<tr>
<td>Т-162 (блок В)</td>
<td>4.5 × 10^{-12}</td>
<td>-</td>
</tr>
<tr>
<td>Машинный зап</td>
<td>8.0 × 10^{-12}</td>
<td>-</td>
</tr>
</tbody>
</table>

Средние концентрации аэрозолей, Кйл/л.
**ТАБЛИЦА VI. СОСТАВ И ОТНОСИТЕЛЬНОЕ СОДЕРЖАНИЕ БЕТА-АКТИВНЫХ НУКЛИДОВ В ПРОБЕ ВОЗДУХА, ОТОБРАННОЙ В БЛОКЕ "В" В ОКТЯБРЕ 1986 Г.**

<table>
<thead>
<tr>
<th>Нуклид</th>
<th>Период полураспада</th>
<th>Концентрация (Сi), Ки/л $\times 10^{-12}$</th>
<th>Вклад ДКА, Ки/л</th>
<th>ДКаА, (Сi), Ки/л</th>
<th>Концентрация (Сi), ДКаА</th>
</tr>
</thead>
<tbody>
<tr>
<td>Церий-144</td>
<td>284,4 сут</td>
<td>8,24</td>
<td>38,3</td>
<td>6,4 $\times 10^{-12}$</td>
<td>1,288</td>
</tr>
<tr>
<td>Церий-141</td>
<td>32,5 сут</td>
<td>0,376</td>
<td>1,8</td>
<td>1,6 $\times 10^{-10}$</td>
<td>0,002</td>
</tr>
<tr>
<td>Рутений-103</td>
<td>39,4 сут</td>
<td>0,379</td>
<td>1,8</td>
<td>5,2 $\times 10^{-11}$</td>
<td>0,007</td>
</tr>
<tr>
<td>Рутений-106</td>
<td>367 сут</td>
<td>1,74</td>
<td>8,1</td>
<td>5,6 $\times 10^{-12}$</td>
<td>0,311</td>
</tr>
<tr>
<td>Цезий-137</td>
<td>30,2 лет</td>
<td>2,48</td>
<td>11,5</td>
<td>1,4 $\times 10^{-11}$</td>
<td>0,177</td>
</tr>
<tr>
<td>Цирконий-95</td>
<td>60 сут</td>
<td>2,64</td>
<td>12,3</td>
<td>3,2 $\times 10^{-11}$</td>
<td>0,083</td>
</tr>
<tr>
<td>Ниобий-95</td>
<td>35 сут</td>
<td>4,81</td>
<td>22,4</td>
<td>1,0 $\times 10^{-10}$</td>
<td>0,048</td>
</tr>
<tr>
<td>Цезий-134</td>
<td>2,07 лет</td>
<td>0,825</td>
<td>3,8</td>
<td>1,3 $\times 10^{-11}$</td>
<td>0,063</td>
</tr>
</tbody>
</table>

Сумма ------------ 21,5  -  -  1,98 ДКаА

---

2 \( \text{ДКаА} = 1,1 \times 10^{-11} \text{Ки/л.} \)

и проживания. Уровни загрязнения поверхностей строительных конструкций, оборудования, производственных помещений в начале сооружения "Укрытия" достигали \(10^8 \text{част/мин-см}^2\) по бета-активным нуклидам и \(10^5 \text{част/мин-см}^2\) по альфа-активным нуклидам.

Определение концентрации радионуклидов в воздухе осуществлялось как по суммарной активности фильтров, так и излучения активности составляющих нуклидов в отдельности (спектрометрическим методом). При этом использовали гамма-спектрометры (табл. I) для идентификации и измерения бета-активности нуклидов в пробе. У некоторых нуклидов (например, стронций-90), входящих в смесь продуктов деления, бета-распад не сопровождается испусканием гамма-квантов. Эти нуклиды не могут быть зарегистрированы гамма-спектрометром, однако вклад их, в общую активность продуктов деления, незначителен и его можно не учитывать. Гамма-спектрометрический метод позволяет, исходя из активностей нуклидов и установленных для них допустимых концентраций (ДКаА), выразить загрязнение воздуха в единицах ДКаА [9].

Пробы воздуха для анализа отбирались на территории и в помещениях ЧАЭС в заданных точках (рис.1). Кроме того, регулярно осуществлялся объезд на бронетранспортере по периметру станции с целью отбора проб воз-
духа в шести точках. Средняя концентрация бета-активных нуклидов в воздухе, усредненная по периметру станции за период с 11.06.86 по 11.07.86, составила 3 × 10⁻¹¹Ки/л (от 10⁻¹² до 2,5 × 10⁻¹⁰Ки/л). Уровни загрязнения воздушной среды в зоне работ приведены в табл. V. По мере сооружения "Укрытия" снижались и концентрации радионуклидов в воздухе (табл. V). Одновременно происходило ужесточение значений ДКд за счет уменьшения доли короткоживущих нуклидов. Так, для бета-активных нуклидов средневзвешенное значение ДКд на 20.06.86 составляло 1,6 × 10⁻¹¹Ки/л, а на октябрь 1986 г. — 1,1.10⁻¹¹Ки/л (табл. VI), а для альфа-активных нуклидов (плутоний-238, 239 и 240, кюрий-242) с июня по октябрь 1986 г. значение ДКд снизилось с 5,5 × 10⁻¹⁵Ки/л до 3,8 × 10⁻¹⁵Ки/л.

Анализ данных по состоянию воздушной среды в зоне работ по сооружению "Укрытия" свидетельствует о том, что концентрации бета-активных аэрозолей в 10²— 10³ раз превышали концентрации альфа-активных аэрозолей. В частности, в пробах воздуха, отобранных в помещениях блока "В" в июле 1986 г., средняя величина отношения бета-активности и альфа-активности составляла 600. Это отношение в выброшенных из реактора продуктах на 06.05.86 составляло 1100. Однако опасность внутреннего облучения, обусловленная альфа-активными аэрозолями, в 2…5 раз выше по сравнению с бета-активными в связи с меньшей ДКд (~ в 3.10³ раз) для альфа-активных аэрозолей.

Концентрации аэрозолей в зоне работ как правило не превышали допустимые более чем на порядок. Лишь при выполнении отдельных пылящих операций загрязнение воздуха достигало 100…300 ДКд. Поскольку уровни гамма-излучения лежали в пределах 10²…5 × 10⁴ДМДд (1 ДМД = 2,8 × 10⁻³Р/ч), то опасность внутреннего облучения персонала была существенно меньшей по сравнению с внешним облучением.

3.3. Облучение персонала

При проведении индивидуального контроля в 1986 г. ОДК использовал 12 пультов УИ-27 и свыше 10 тыс. дозиметров Д-2 (Д-2Р), а также 9 пультов УПФ-02 с 10 сериями дозиметров ДПГ-03 по 1250 дозиметров в серии и 4 пульта ИКС с 6 тыс. дозиметров ИКС-А. Проведено около 90 тыс. измерений суммарной дозы и свыше 700 тыс. измерений дозиметров Д-2. При проведении ИДК руководствовались значениями допустимой дозы 25 Р за период работы (~ 2 месяца) и контрольного уровня 1 Р за смену. Результаты контроля приведены в табл. VII.
Облучаемость персонала в период работ по сооружению "Укрытия" была неравномерной. Наибольшему облучению персонал подвергался в октябре-ноябре, когда проводились работы по монтажу перекрытия IV блока (средняя доза 10,4 Р). На этот период приходятся случаи повышенного (более 25 Р) облучения, при этом переоблучение в большинстве случаев не превышало 1-3 Р. Основной причиной повышенного облучения являлось привлечение добровольцев из числа высококвалифицированного персонала с накопленной дозой свыше 20 Р к выполнению аккордных разовых работ, при которых дозы облучения составляли 2...4 Р.

Внешнее облучение связано не только с гамма-, но и с бета-излучением. Для бета-излучения критическим органом является кожа тела человека. Подавляющая часть тела человека экранирована по отношению к бета-излучению средствами индивидуальной защиты (сапоги, перчатки, фартуки из просвинцованной резины, специальные очки или щитки). В связи с этим фактическая опасность бета-излучения существенно ниже по сравнению с гамма-излучением.

Выборочный контроль внутреннего содержания осуществлялся по бета-активным (гамма-излучающим) нуклидам на счетчике излучения человека (СИЧ). Было обследовано 350 сотрудников, получивших дозу внешнего облучения 20 Р и выше. У 90% обследованных содержание радионуклидов в организме составляло 0,01—0,10 ДСд, максимальное содержание не превышало 0,3 ДСд (цирконий-95, ниобий-95, рутений-103, цезий-134 и 137). Внутреннее облучение, рассчитанное на 50 лет, не превышает 1 бэр.

На основании полученных данных можно сделать вывод о том, что внутреннее облучение персонала было ~ в 10^2 раз ниже внешнего облучения.

4. ЗАКЛЮЧЕНИЕ

Сооружение объекта "Укрытие" по своим масштабам, сложности, ответственности за безопасность большого количества людей не имеет аналогов в отечественной и мировой практике [3]. Накопленный опыт работы в условиях ликвидации последствий аварии на Чернобыльской АЭС и полученные научно-технические результаты радиационного контроля позволяют сформулировать следующие рекомендации, которые целесообразно реализовать в рамках МАГАТЭ:

1. Разработать технические требования и создать комплекс аппаратуры для проведения радиационной разведки и контроля радиационной обстановки при ликвидации последствий возможных радиационных аварий.
<table>
<thead>
<tr>
<th>Количество сотрудников, занятых сооружением &quot;Укрытия&quot;</th>
<th>Средняя доза, Р</th>
<th>Коллективная доза, Р</th>
<th>Количество сотрудников, получивших дозу облучения в пределах, Р</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>0,5</td>
</tr>
<tr>
<td>21 511</td>
<td>8,57</td>
<td>184 451</td>
<td>10 890</td>
</tr>
<tr>
<td>(100%)</td>
<td></td>
<td></td>
<td>(50,6%)</td>
</tr>
</tbody>
</table>
2. Разработать штатные методики прогнозирования, разведки и контроля радиационной обстановки на всех стадиях и этапах радиационной аварии и ее последствий.

3. Создать учебно-методические и справочные пособия, регламентирующие действия служб радиационной безопасности в условиях возможных аварий применительно к различным производствам и АЭС.

ЛИТЕРАТУРА


Abstract—Аннотация

RADIATION RECONNAISSANCE AND PROTECTION METHODS DURING CONSTRUCTION OF THE ENCAPSULATION FOR UNIT 4 OF THE CHERNOBYL NUCLEAR POWER PLANT.

A summary is given of the results obtained and experience acquired by the radiation reconnaissance and operations radiation protection group during work on construction of the encapsulation for Unit 4 of the Chernobyl nuclear power plant. The group’s objectives included: experimental study of radiation fields in order to determine their structure; determination of which radiation sources were making the greatest contribution to the exposure dose received by construction personnel at the proposed work sites; issuing of recommendations on preliminary damping of the sources, or protection of work sites on those sides where the risk from radiation was at its greatest; and prediction of changes in the radiation situation brought about by implementing the proposed measures.

METODY RADIAZIONNOY RAZVEDEKI I ZAŠCHITY PRI SOORUŽENII "UKRYYTIJA" IV ENERGOBLOKA CHERNOBYL'SKOY AES.

В докладе представлено обобщение результатов и опыта, полученных группой радиационной разведки и оперативной противорадиационной защиты в ходе работ по сооружению "Укрытия" IV энергоблока ЧАЭС. В задачи группы входили: экспериментальное исследование радиационных полей с целью расшифровки их структуры; определение источников излучения, дающих основной вклад в мощность экспозиционной дозы на предполагаемых рабочих местах строительного персонала; выработка рекомендаций по предварительному подавлению этих источников или защите рабочих мест со стороны наиболее радиационно опасных направлений; прогнозирование изменения радиационной обстановки в результате выполнения предложенных мероприятий.
Основными отличиями радиационной обстановки на АЭС, сложившейся в результате крупной и особокрупной (типа Чернобыльской) аварий, от радиационной обстановки в нормальном режиме ее эксплуатации являются следующие:

1. Мягкий энергетический спектр гамма-излучения источников.
2. Заметный вклад бета-частиц в полную дозу внешнего излучения.
3. Высокая опасность внутреннего облучения за счет ингаляционного поступления альфа-, бета- и гамма-активных аэрозолей.
4. Очень большая степень неопределенности в пространственном распределении источников излучения и, как следствие, большое разнообразие видов угловых распределений мощности дозы.

Перечисленные выше отличия затрудняют использование в аварийных ситуациях штатной дозиметрической аппаратуры, разработанной для эксплуатации в условиях нормально действующей АЭС. Кроме того, штатные ситуации, с которыми обычно сталкиваются службы радиационной безопасности (РБ) при авариях на АЭС, в том числе и максимальных проектных авариях, характеризуется тем, что большинство параметров, определяющих радиационную обстановку (геометрия задачи, конфигурация и материал защитных стен, пространственно-угловое и энергетическое распределение, а также суммарная активность источников излучения) известны с достаточной степенью точности.

Рассматриваемая ситуация на Чернобыльской АЭС характеризуется очень большой неопределенностью во всех основных радиационных параметрах. В связи с этим известные практичеокие методики в таком случае не вполне приемлемы. Тем более невозможна какая-либо априорная расчетная оценка мощностей доз и парциальных вкладов в дозы от различных источников.

Следовательно, в условиях послеаварийной ситуации на Чернобыльской АЭС необходимо опираться исключительно на экспериментальные измерения и методики, причем, как правило, следует выявлять не отдельные источники, а целые классы этих источников (типа "верх-низ", "права-лево"). Расчетные методики могут быть использованы только для определения параметров защитных устройств, причем исходными данными для расчета должны быть непосредственно измеренные величины.

Учитывая изложенные выше отличительные особенности радиационной обстановки в районе аварийного IV энергоблока Чернобыльской АЭС в структуре отдела радиационной безопасности, была создана специальная группа радиационно-технической разведки (РТР) и противорадиационной защиты (ПРЗ).
Целью РТР и ПРЗ является выработка рекомендаций по оптимальной временной последовательности выполнения строительных работ и отдельных операций, а также по необходимым дополнительным противорадиационно-защитным мероприятиям на рабочих местах строительного персонала для минимизации дозовых затрат.

Задачей РТР является оперативное получение экспериментальной информации об отдельных характеристиках дозовых полей, позволяющей после соответствующего анализа и обработки с использованием априорной информации о физических закономерностях формирования этих

РИС.1. Коллимирующие устройства для определения пространственного распределения источников излучения (D — детекторы), где: а — свинцовый коллиматор 'еж'; б — бронированное транспортное средство.
полей, расшифровать их структуру, определить основные источники радиационной опасности и их относительные вклады в полную мощность дозы, а также дать количественный прогноз улучшения радиационной обстановки по мере реализации тех или иных строительных мероприятий до их выполнения.

Задачей ПРЗ является экспериментально-расчетное определение защитных свойств сохранившихся и проектируемых элементов строительных конструкций, а также определение параметров оперативных противорадиационных защит рабочих мест персонала: тип защиты, место ее расположения, конфигурация, материал, толщина и др., необходимых для снижения мощности дозы непосредственно на рабочих местах строителей до приемлемых пределов в случаях, когда РТР установила необходимость и возможность проведения дополнительных защитных мероприятий.

Ниже рассмотрены основные оперативные методики РТР, предназначенные для локального анализа структуры дозовых полей на рабочих местах персонала, позволяющие выявлять 1-2 доминирующих источника (или направления на источник) и выработать конкретные рекомендации по защитным мероприятиям.

Одна из разработанных методик предназначена для проведения РТР в условиях, когда отсутствует предварительная информация о доминирующих источниках излучения в пространстве. Для измерения уровней излучения использовали коллиматор “еж” (Рис. 1(a)), представляющий собой 8 отсеков, разделенных свинцовыми листами 100 х 100 и толщиной 10 мм. Во внутренних углах коллиматора расположены 8 дозиметров интегрального типа ИКС-А (ТЛД), каждый из которых измеряет излучение из 1/8 пространства. Коллиматор с детекторами устанавливали на штативе в точках, подлежащих контролю. После определенного времени экспонирования производили измерения накопленной дозы. Сопоставление данных, полученных в результате обработки дозиметров с учетом местоположения точки измерения и ориентации коллиматора, позволяет выделить основные источники излучения, дающие определяющий вклад в формирование радиационного поля в данной точке (с угловым разрешением π/2 стерадиан).

Другой модификацией этой методики является использование защитной техники (бронетранспортер, танк и др.) в качестве “коллиматора” (Рис. 1(b)). В этом случае ТЛД дозиметры крепятся по граням транспортного средства (перед-зад, левый-правый борт, верх-вниз). Защитные свойства транспортного средства (стальной корпус и свинцовая защита) обеспечивают со стороны каждой из шести граней измерение интегральной дозы от соответствующего полупространства и позволяет с угловым разрешением 2π стерадиан выделить направления на основные
группы источников, а также оценить их относительный вклад в мощность дозы излучения в данной точке.

Методику целесообразно использовать на больших площадях загрязненной территории с высокой неопределенностью распределения источников и высокими уровнями излучения (10–100 Р/ч и более).

К достоинствам методики следует отнести минимизацию дозовых нагрузок на персонал, осуществляющий РТР, так как установка и снятие коллиматора с детекторами (Рис. 1(a)) в полях с высоким уровнем радиации может быть осуществлена достаточно быстро и, в ряде случаев, с использованием защищенной техники, а применение модифицированного варианта (Рис. 1(b)) методики исключает возможность переоблучения персонала даже при работе в полях радиации в несколько сотен Р/ч. Кроме того, имеется возможность получения дополнительной информации путем суммирования показаний детекторов вплоть до угла $4\pi$ стерадиан.

Недостатками методики являются:
— большая, по сравнению с использованием прямопоказывающих дозиметров, задержка в получении результатов (за счет времени доставки и обработки дозиметров);
— недостаточная точность абсолютных измерений (особенно полной дозовой характеристики) из-за недостаточного ослабления излучения стенками коллиматора и из-за погрешностей, вносимых экспозицией во время их транспортировки.


Следует отметить, что до проведения отделом РБ радиационной разведки и выработки технических предложений, доминирующим являлось мнение о том, что основным источником излучения является разрушенный реактор и в качестве первоочередной задачи рассматривалось возведение защитных стен из бетона вдоль оси A и с западной и северной сторон IV энергоблока. Выполнение ее привело бы, без заметного положительного эффекта (снижения мощности дозы) к значительному увеличению сроков восстановительных работ и неоправданным дозовым
РИС.2. Устройство для определения локальных источников излучения (D — детектор), где: а — свинцовый коллиматор с торцевым отверстием; б — коллиматор со щелевым отверстием.

затратам (свыше 10 тыс. чел.-бэр). Для сравнения укажем, что дозовые затраты на проведение РТР по территории составили 10 чел.-бэр.

Для оперативного поиска наиболее мощных точечных или линейных источников ("горячих" точек) и определения вклада "прострельного" гамма-излучения из отверстий и щелей в стенах использовали коллиматоры, представленные на Рис. 2 (а, б). Свинцовый коллиматор толщиной 1 см (Рис. 2(а)) с торцевым отверстием обеспечивает лучшее угловое разрешение и позволяет выявлять мощные локальные источники. Этот коллиматор использовали с приборами ДРГЗ-03, СРП-68. Коллиматор со щелевым отверстием (Рис. 2(б)) имеет худшее угловое разрешение, но с его помощью удобнее выявлять протяженные источники (трубы, кабельные короба и др.)

К достоинствам методики следует отнести ее оперативность, так как источники излучения идентифицируются сразу во время проведения РТР.

Недостатками методики являются:
— относительность полученных данных;
— невозможность оценки абсолютного вклада исследуемого источника в формирование радиационного поля в данной точке.
Определение вкладов в мощность дозы излучения от различных источников (поверхности почвы, дорог, кровли, пола, стен и др.) осуществлялось путем трех последовательных измерений (Рис. 3) в одной точке: 1 — полная мощность дозы ($P_0$), 2 — мощность дозы за теневой защитой $t_1$ ($P_1$), 3 — мощность дозы за теневой защитой $t_1 + t_2$ ($P_2$). Теневая защита — свинцовый коллиматор, закрывающий детектор на 180 градусов. Толщина коллиматора $t_1 = t_2 = 3-5$ мм.

Результаты измерений могут быть использованы как непосредственно (при этом значения ($P_2$), ($P_2/P_0$) дают, соответственно, абсолютный и относительный вклады гамма-излучения от плоскости с незашитенной стороны коллиматора), так и с последующей расчетной обработкой, позволяющей определить эффективную толщину дополнительной защиты от излучения, приходящего со стороны коллиматора, при решении задач ПЗ.

Рис. 3. Определение вкладов в мощность дозы излучения от протяженных источников.
Пусть $P_s$ и $P_a$ — абсолютные вклады в мощность дозы от анализируемого источника и от всех остальных источников. Тогда данные измерений $P_0$, $P_1$, $P_2$ удовлетворяют системе уравнений (при $t_1 = t_2$):

\[
\begin{align*}
P_0 &= P_s + P_a, \\
P_1 &= P_s/K + P_a, \\
P_2 &= P_s \frac{1}{K^2} + P_a,
\end{align*}
\]

где $K$ — кратность ослабления от анализируемого источника излучения слоем свинца толщиной $t_1$. Решая систему уравнений, получим:

\[
\begin{align*}
P_s &= \frac{(P_0 - P_1)^2}{P_0 - 2P_1 + P_2}; \\
P_s &= \frac{P_0P_2 - P_1^2}{P_0 - 2P_1 + P_2}; \\
K &= \frac{P_0 - P_1}{P_1 - P_2}.
\end{align*}
\]

Использование соотношений (2) позволяет не только оценить абсолютный вклад анализируемого источника в формирование мощности дозы в рассматриваемой точке ($P_s$), но и определить (по величине кратности ослабления $K$, толщине защиты $t_1$ и справочным данным о кратности ослабления гамма-излучения материалами различных толщин в зависимости от его энергии (1) эффективную энергию излучения. Кроме того, использование результатов измерений позволяет определить целесообразность использования ПРЗ и ее оптимальные параметры: коэффициент ослабления, материал и требуемую при этом толщину следующим образом.

Пусть $P_n$ — новое значение мощности дозы в исследуемой точке после проведения противорадиационных мероприятий, то есть после установки ПРЗ от излучения из выделенного направления. Тогда

\[
P_n = P_a + P_s/K_s,
\]

где $K_s$ — кратность ослабления устанавливаемой защитой излучения, приходящего с выделенного направления. Очевидно, что при $P_a > 0$ уменьшение вклада от источника в новое значение мощности дозы $P_n$ до нуля не имеет смысла. Разумно снизить этот вклад до значений, малых по сравнению с $P_n$, например до величины порядка погрешности опреде-
ления $P_n$, то есть $P_n = P_a(1 + d)$, где $d$ — значение относительной погрешности измерений $0,1 < d < 0,3$. Тогда получаем:

$$K_s = \frac{P_s}{P_a d} = \frac{(P_0 - P_1)^2}{(P_0P_2 - P_1P_1) d}.$$  \hspace{1cm} (4)

Необходимая толщина защиты из бетона или какого-либо другого материала определяется по таблицам для расчета защит, связывающим между собой энергию гамма-квантов, материал и толщина защиты с коэффициентом ослабления $K_s$. Сначала по таблицам для свинца по значениям $t$ и $K$ (смотри выражение (2)) находим $E_{\text{eff}}$ — эффективную энергию спектра гамма-излучения от анализируемого источника, а затем по данным для бетона (или иного материала) и значениям $K_s$ и $E_{\text{eff}}$ определяем необходимую толщину защиты из этого материала.

Полученные в эксперименте и с использованием расчетной методики значения $P_0$, $P_s$ и $P_a$ были использованы для предварительной оценки кратности уменьшения дозы на рабочем месте после проведения защитных мероприятий $K_d$:

$$K_d = \frac{P_0}{P_n} = \frac{P_0}{(1 + d)} \cdot \frac{(P_0 - 2P_1 + P_2)}{P_0P_2 - P_1^2}.$$  \hspace{1cm} (5)

Достоинства методики:
— быстрота получения конечного результата при достаточной точности измерения;
— получение абсолютных значений полной дозы в точке измерения;
— возможность расчетным путем по исходным значениям $P_0$, $P_1$, $P_2$ определять абсолютный вклад рассматриваемого источника в полную мощность дозы $P_0$, кратность ослабления теневой защиты для условий измерения и необходимую толщину защитных сооружений из любого другого материала.

Недостаток методики — невысокое угловое разрешение, что не позволяет точно выявить мощные локальные (сосредоточенные) источники.

Разрабатывались и использовались другие методы радиационной разведки и контроля радиационной обстановки. Так, для исследования распределения уровней гамма-излучения по высоте (до 80 метров) использовали гелиевые шары-зонды с капровым шнуром, на который через фиксированные промежутки (около 1 м) крепили дозиметры ИКС-А. Таким образом определяли распределение уровней излучения по высоте
машинного зала, на кровлях третьего энергоблока и территории вокруг четвертого блока.

Распределение уровней излучения по горизонтали (на поверхностях оборудования, кровлях и др.) определялось путем экспонирования в течение определенного промежутка времени дозиметров ИКС-А, закрепленных на полиэтиленовых листах (простынях) или капроновых шнурах. Шнуры имели на концах свинцовые грузы, с помощью которых гирлянды дозиметров забрасывались в труднодоступные места.

Анализ и обработка данных, полученных по указанным методикам, позволили сформулировать требования к противорадиационной защите рабочих мест строителей. Было установлено, что на каждом этапе работ по сооружению "Укрытия" существовал нижний предел возможного снижения уровней воздействия на персонал, обусловленный вкладом в эти уровни рассматриваемой компоненты излучения. Указанное обстоятельство позволило выбирать оптимальные параметры радиационной защиты для различных видов восстановительных работ. Как правило, оптимальный коэффициент кратности ослабления мощности дозы направленного излучения теневой защитой (экранией) составлял 5-10.

**ЛИТЕРАТУРА**

ОПТИМИЗАЦИЯ ДОЗОВЫХ ЗАТРАТ ПРИ ЛИКВИДАЦИИ ПОСЛЕДСТВИЙ КРУПНЫХ АВАРИЙ НА АЭС

Л.Ф. БЕЛОВОДСКИЙ
Государственный комитет по использованию атомной энергии СССР

И.А. БЕЛЯЕВ, Л.А. ЛЕБЕДЕВ, С.Г. МИХЕЕНКО, А.А. СТРОГАНОВ
Московский инженерно-физический институт, Москва, Союз Советских Социалистических Республик

Abstract—Аннотация

ОПТИМИЗАЦИЯ ДОЗОВЫХ ЗАТРАТ ПРИ ЛИКВИДАЦИИ ПОСЛЕДСТВИЙ КРУПНЫХ АВАРИЙ НА АЭС.

Рассмотрены теоретические подходы к оптимизации дозовых затрат при ликвидации последствий крупных аварий на АЭС, а также их практическая реализация.

1. ТЕОРЕТИЧЕСКИЕ ПОДХОДЫ К ОПТИМИЗАЦИИ ДОЗОВЫХ ЗАТРАТ

Для решения задач оптимизации дозовых нагрузок при минимальных затратах на обеспечение приемлемого уровня radiационной безопасности в атомной технике и энергетике общепринятым является метод анализа соотношения "польза-ущерб", основанный на так называемой "концепции оправданного риска" [1].

Значение коллективной дозы персонала при условии непревышения отдельными лицами предельно допустимых значений дается в общем виде выражением:

\[
D(t_h, t_k) = \int_{t_h}^{t_k} \int_{r_0}^{r_1} H(r') f(t) n(r', t) F(r', t) \, dr \, dt ,
\]  

(1)

где \( r \) — пространственная переменная, определяющая положение рабочих мест персонала; \( H(r') \) — пространственное распределение мощности экспозиционной дозы на рабочих местах разрушенного энергоблока; \( f(t) \) — временная зависимость изменения радиационных характеристик облученного ядерного топлива за счет радиоактивного распада; \( n(r', t) \) — распределение численности основного строительного персонала по рабочим местам и во времени, определяемое полным объемом восстановительных работ \( W \) и его распределением по рабочим местам \( w(r') \) следующим образом:

\[
\int_{t_h}^{t_k} n(r', t) \, dt = w(r')/q; \quad \int w(r') \, dr' = W ,
\]  

(2)

где \( q \) — средняя производительность работ; \( F(r', t) \) — функция относительного уменьшения мощности дозы \( H(r') f(t) \) за счет уже выполненной части строительно-восстановительных работ (дезактивации, возведения защитных элементов сооружения и т. д.), \( t_h, t_k \) — время начала и окончания работ по сооружению объекта "Укрытие".

В отличии от традиционной формулировки задач оптимизации в проблеме обеспечения радиационной безопасности, значение \( D \) в нашем случае существенно зависит (через \( F(r', t) \)) не только от прямых затрат на мероприятия по противорадиационной защите персонала, но и от выполнения этим персоналом своей основной работы, точнее — от принятой последовательности выполнения отдельных ее этапов — \( n(r', t) \), которая может варьироваться в очень широких пределах. Характер зависимости функции \( F(r', t) \), названной авторами функцией "активного воздействия на радиационную обстановку", от распределения последовательности выполнения восстановительных работ \( n(r', t) \) рассмотрен ниже. Запишем выражение для функции \( F(r', t) \) через пространственное распределение выброшенного из аварийного энергоблока облученного топлива — источника излучения \( S(r') \) на момент \( t = 0 \) и через функцию \( G(r- > r'^{+}) \) — формирования мощности дозы излучения в точке \( r^{+} \) от единичного источника, находящегося в точке \( r' \) с учетом ослабления за счет рассеяния \( (r- > r') \) и
защитных свойств сооружения. Полагая, что выполнение намеченного объема восстановительных работ в районе \( \vec{r} - w(\vec{r}) \) обеспечивает безусловное подавление источников \( S(\vec{r}) \) в этой области (за счет дезактивации, либо противорадиационно-защитных свойств сооружаемого в \( \vec{r} \) элемента "Укрытия") получим:

\[
H(\vec{r}) = \int S(\vec{r}'') G(\vec{r}'' \rightarrow \vec{r}) d\vec{r}''
\]

(3)

\[
H(\vec{r}) F(\vec{r}, t) = \int S(\vec{r}'') G(\vec{r}'' \rightarrow \vec{r}) (1 - q/w(\vec{r}'')) \int_0^t n(\vec{r};t') dt' d\vec{r}''
\]

(4)

Поиск функции \( n(\vec{r}, t) \), удовлетворяющей условиям (2), на которой функционал \( D \), определяемый выражениями (1)–(4), достигает минимума, имеет очевидный физический смысл. Он состоит в определении оптимальной последовательности отдельных этапов работ, обеспечивающей наиболее быстрое активное снижение уровней радиационных полей на рабочих местах для выполнения последующих этапов. Решение этой задачи не представляет принципиальной сложности и может быть проведено стандартными численными методами при условии, что известны функции \( S(\vec{r}) \), \( G(\vec{r}'' \rightarrow \vec{r}) \), характеризующие выброс радиоактивных веществ, защитные свойства сохранившихся элементов сооружения, и функция \( w(\vec{r}) \), определяющая объемы конкретных работ по восстановлению сооружения. Однако это возможно лишь при относительно небольших авариях, но даже в этом случае данные по \( G(\vec{r}'' \rightarrow \vec{r}) \), полученные для любого сооружения АЭС заранее, в научной литературе отсутствуют. Исключение составляют лишь коэффициенты переноса излучений в приземном слое атмосферы (так называемый "skyshine") [3].

При крупных и особо крупных авариях на АЭС данные по распределению \( S(\vec{r}) \), состоянию сооружения \( G(\vec{r}'' \rightarrow \vec{r}) \), объему конкретных восстановительных работ \( w(\vec{r}) \) на момент начала восстановления практически полностью отсутствуют. Для этих условий предложена следующая, реализованная в основных чертах при сооружении "Укрытия", практическая схема минимизации дозовых нагрузок персонала, конкретизирующая концепцию оптимального активного воздействия на радиационную обстановку с целью ее нормализации.

1. Предстоящие строительно-восстановительные работы классифицируются на основные этапы с существенно различными уровнями облучения персонала, разной численностью и разными основными задачами.

2. Определяется стратегия проведения работ на первом этапе: общий уровень технической вооруженности, характерные противорадиа-
ционные защитные мероприятия, степень нормализации радиационной обстановки, позволяющая считать этап завершенным. Особую роль играет основанная на имеющихся в распоряжении экспериментальных данных, визуальных наблюдениях, фото- и видеоматериалах предварительная оценка мощности основных источников излучения (развал реактора, территория, кровли и т.д.) и, самое главное, расчетное предсказание степени влияния каждого из них на радиационную обстановку на рабочих местах текущего этапа работ. Временная очередность выполнения восстановительных работ на различных участках объекта должна соответствовать упорядоченной по убывающей относительных величин последовательности предполагаемых вкладов от доминирующих источников в облучение персонала при выполнении работ, запланированных на каждом этапе. Так как в процессе выполнения работ изменяется мощность источников, то периодически порядок этой последовательности должен уточняться.

3. Определяется регламент проведения строительных работ. Основными принципами при этом являются:

(a) проведение строительных работ в радиационно-опасных условиях допустимо только, если они ведут к уменьшению $S(F)$;

(b) при неразличимости вкладов в районе $F$ от источников, расположенных в непосредственной близости $S(F)$ и от удаленных источников $S(F')$, строительные работы в $F$ должны быть приостановлены. Очевидным следствием этих принципов является "дробность" по времени выполнения полного объема работ, многократное возвращение персонала на одни и те же рабочие места. Такая тактика обеспечивает лишь локальное выполнение концепции оптимального активного воздействия на радиационную обстановку с целью ее нормализации, но реально, при условии корректного выполнения принципов стратегии на данном этапе и с учетом использования защитных средств и специальной техники, фактически получающаяся последовательность выполнения работ обеспечит значение $D$, близкое к минимальному.

2. ПРАКТИЧЕСКАЯ РЕАЛИЗАЦИЯ

В соответствии с разработанными выше принципами в самом начале работ по сооружению "Укрытия" IV-го энергоблока ЧАЭС была принята предварительная классификация основных этапов работ по его захоронению:
ПЕРВЫЙ ЭТАП характеризовался:

(a) недостаточностью данных по мощности экспозиционной дозы на рабочих местах персонала, довольно высокими ее уровнями;
(b) полной неопределенностью структуры радиационных полей на рабочих местах.

Задачей этапа было выполнение работ по сооружению конструкционных узлов и элементов "Укрытия", защищающих рабочие места от основных источников излучения, которые дают основной вклад в мощность дозы, или дезактивация этих источников (подготовка пространства для развертывания основных строительных работ). Задачей специалистов по радиационной физике отдела дозиметрического контроля (ОДК), занимающихся оперативной радиационно-технической разведкой (РТР) и противорадиационно-защитным сопровождением строительных работ, было определение структуры мощности экспозиционной дозы на рабочих местах персонала (но не мощности отдельных источников, а только их вклад в структуру поля), выявление источников, определение направления на них, выработка рекомендаций по последовательности их подавления (предварительная дезактивация, сооружение элемента "Укрытия", установка временных теневых противорадиационных защит и т. д.). Использованные для этой цели методы анализа полей были почти чисто экспериментальными.

Трудозатраты строительного персонала, занятого на работах первого этапа, должны быть минимальны, а уровень средств индивидуальной защиты — достаточно высоким (использование, где это возможно, транспорта с защищенными кабинами, теневых подвесных защит, освинцованных капсул-батискаfov и т. д.). Первый этап завершается, когда наиболее сильные удаленные и близкорасположенные источники излучений подавлены, то есть на рабочих местах персонала невозможно выделить доминирующий вклад от какого-либо источника.

ВТОРОЙ ЭТАП характеризовался:

(a) множественностью источников излучения небольшой мощности, создающих на всех рабочих местах сравнительно невысокие уровни мощности дозы, причем распределение мощности дозы носит "диффузный" характер;
(b) возможностью для персонала широким фронтом выполнять операции по сооружению "Укрытия", изредка отвлекаясь на подавление отдельных источников излучения.

Задача второго этапа — выполнение основного объема строительных работ, кроме отделочных.
<table>
<thead>
<tr>
<th>Источник &gt; &gt;</th>
<th>Территория около IV-го энергоблока</th>
<th>Разрушенный реактор</th>
<th>Кровля</th>
<th>Внутреннее загрязнение помещений</th>
</tr>
</thead>
<tbody>
<tr>
<td>Территория около IV-го энергоблока</td>
<td>85</td>
<td>5-10</td>
<td>0-5</td>
<td>90-95</td>
</tr>
<tr>
<td>Каскадная стена (отметки 0-18 м)</td>
<td>90</td>
<td>10</td>
<td>0-5</td>
<td>80</td>
</tr>
<tr>
<td>Каскадная стена (отметки 18-35 м)</td>
<td>80</td>
<td>15</td>
<td>5-10</td>
<td>70</td>
</tr>
<tr>
<td>Каскадная стена выше 35 м</td>
<td>70-75</td>
<td>15-20</td>
<td>10-15</td>
<td>65</td>
</tr>
<tr>
<td>Разделительная стена в блоке &quot;В&quot; ниже 35 м</td>
<td>20</td>
<td>20</td>
<td>80</td>
<td>0-15</td>
</tr>
<tr>
<td>Разделительная стена в блоке &quot;В&quot; выше 35 м</td>
<td>25</td>
<td>5</td>
<td>70</td>
<td>0-5</td>
</tr>
<tr>
<td>Разделительная стена в МЗ-3 до отметки 18 м</td>
<td>5</td>
<td>5</td>
<td>80</td>
<td>10</td>
</tr>
<tr>
<td>Разделительная стена в МЗ-3 (отметки 18-28 м)</td>
<td>0</td>
<td>0</td>
<td>100</td>
<td>0-10</td>
</tr>
</tbody>
</table>

Примечание. Вверху — предварительная оценка и рекомендации группы ОДК; внизу — реальные значения, определенные прямым измерением после подавления источников; в скобках — последовательность их подавления, оптимизирующая дозовые затраты.
Задачей специалистов по радиационной физике является выявление всех источников излучения в пределах сооружения (а не только на рабочих местах строителей), для чего должно быть проведено экспериментальное исследование объемной структуры радиационных полей с последующей расчетной обработкой. Трудозатраты строителей на этом этапе максимальны, но относительные дозовые затраты должны быть на порядки меньше, чем на первом этапе.

ТРЕТИЙ ЭТАП характеризуется завершенностью сооружения и практически полным отсутствием вблизи него сколько-нибудь существенного количества радиоактивных веществ. В этих условиях определяющим становится вклад удаленных внешних источников большой мощности за счет переноса излучения в приземном слое атмосферы. На первый план при этом выходят задачи специалистов, состоящие в расчетном или расчетно-экспериментальном прогнозировании уменьшения мощности дозы за счет устранения влияния тех или иных источников. Поскольку технические мероприятия по такому устранению могут быть весьма дороги, необходима достаточно корректная расчетная оценка ожидаемого выигрыша до реализации этого технического решения. Необходима также оценка дозовых затрат на его реализацию. Задачей третьего этапа является завершение работ в целом.

Рассмотрим конкретные примеры реализации разработанных оптимизационных подходов при захоронении IV-го энергоблока ЧАЭС. В табл. I представлены оцененные по результатам анализа структуры дозовых полей относительные вклады в полную мощность дозы на основных строительных площадках первого этапа сооружения "Укрытия".

На третьем этапе работ (после замены кровли машинного зала в августе 1987 г.) были оценены значения относительных средних вкладов (%) от различных удаленных источников в полную мощность дозы на кровле машинного зала (М3) III-го энергоблока ЧАЭС, приведенные в табл. II.

Снижение дозовых затрат персонала, занятого на ликвидации последствий аварии, может быть достигнуто соответствующими строительными решениями, технологией проведения строительных работ, а также комплексом организационно-технических мероприятий. В конкретных условиях последствий аварии на ЧАЭС была применена совокупность подобных решений. В качестве основных следует указать:

— поэтапное освоение пространства для производства основных работ с последовательным увеличением числа работающих по мере снижения уровней радиации на рабочих местах;
— надвижка предварительно смонтированных в безопасных местах защитных стен по контуру аварийного реактора на сохраняв-
ТАБЛИЦА II. ОТНОСИТЕЛЬНЫЙ ВКЛАД РАЗЛИЧНЫХ ИСТОЧНИКОВ В МОЩНОСТЬ ДОЗЫ НА КРОВЛЕ МАШИННОГО ЗАЛА (МЗ) III-го ЭНЕРГОБЛОКА

<table>
<thead>
<tr>
<th>№ п/п</th>
<th>Источники излучения</th>
<th>Относительный вклад, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Кровля МЗ IV-го энергоблока:</td>
<td></td>
</tr>
<tr>
<td></td>
<td>в осях 36-46</td>
<td>5 — 5</td>
</tr>
<tr>
<td></td>
<td>в осях 46-54</td>
<td>15 — 20</td>
</tr>
<tr>
<td></td>
<td>в осях 54-62</td>
<td>25 — 30</td>
</tr>
<tr>
<td></td>
<td>в осях 62-68</td>
<td>5 — 10</td>
</tr>
<tr>
<td>2</td>
<td>Объект “Укрытие”</td>
<td>20 — 25</td>
</tr>
<tr>
<td>3</td>
<td>Собственное загрязнение кровли МЗ III-го энергоблока</td>
<td>10 — 15</td>
</tr>
<tr>
<td>4</td>
<td>Южная промышленная зона</td>
<td>0 — 5</td>
</tr>
</tbody>
</table>

Примечание. Первое число — предварительная оценка и рекомендации ОДК, второе — реальные значения, определенные прямым измерением после подавления источников.

шшихся и вновь проложенных железнодорожных путях, использование максимально укрупненных конструкций, устанавливаемых мощным подъемным краном;
— использование “безлюдной” технологии укладки бетона и монтажа металлоконструкций;
— использование теневых противорадиационных защит и специальной защищенной техники;

После завершения основных работ по ликвидации последствий аварии на ЧАЭС сделана ретроспективная оценка максимально возможного снижения дозовых затрат при сооружении "Укрытия", которое могло быть достигнуто. Полученное значение оказалось близким к реально достигнутой кратности снижения дозовых затрат. Общее снижение этих затрат составляет величину около 3-х раз по сравнению с дозовыми затратами, ожидавшимися при применении обычных техник и приемов работ. Это снижение колеблется в достаточно широких пределах: от десятков раз при дезактивации и бетонировании территории вокруг IV-го энергоблока и возведения разделительной стены на нижних отметках блока "В" и машинного зала до 1,5–2 раз при выполнении работ на верхних отметках и кровлях.
ЛИТЕРАТУРА


PREPARATION FOR CLEANUP ACTIVITIES AFTER SEVERE ACCIDENTS: A SWEDISH PERSPECTIVE

G. HULTQVIST
Swedish State Power Board,
Forsmark,
Sweden

Abstract

PREPARATION FOR CLEANUP ACTIVITIES AFTER SEVERE ACCIDENTS: A SWEDISH PERSPECTIVE.

Everyone is concerned to avoid the accidents which can occur in any nuclear power plant. Environmental consequences can be limited if designs use a containment which only allows very small leakages. At Forsmark an additional filtered ventilation system has been installed to reduce the contamination of the environment in situations where high pressure could challenge the integrity of the containment. There are no rules for cleanup system design or for operations to clean up and handle damaged fuel and radioactive material released from the core of a plant. In Sweden, the Inspectorate and the utilities have been devoting increased attention to this problem. Two analyses have been performed to examine the consequences of major damage to nuclear fuel; one investigated a 1% damaged fuel pin in the core and another considered 10% damaged fuel. Neither case involved melted fuel. The 1% case presents minor problems to the plant; the 10% case creates huge problems, especially as regards waste management. Because of radioactivity within the plant and waste handling, a lot of areas have to be restricted. The paper presents results from these studies and outlines the FRIPP study of core meltdown currently under way.

Three Mile Island, Chernobyl, Chalk River and Sellafield provide examples that severe accidents can and do occur in nuclear power plants.

For the nuclear industry worldwide it is of great importance that such accidents are avoided. If severe accidents of any kind occur the nuclear industry must be able to handle the situation so that the impact on the environment and the employees is negligible.

In Sweden a lot of studies have been done to analyse severe accident scenarios. As a consequence of the Three Mile Island accident the authorities in Sweden enforced rules concerning the ability of the nuclear plants to handle an event where the reactor core has gone into a melting condition. The rules state that such a situation may not affect the environment.
This new demand forced the Swedish State Power Board to make detailed analyses of core meltdown scenarios and scenarios involving extreme pressures in the containment.

These analyses studied the effects until a steady state for containment condition and the core configuration had been reached. This time period was often less than 24 hours.

Studies such as FILTRA, MITRA and RAMA resulted in the development and construction of our filtered ventilation concept and the upgrading of our Emergency Operating Procedures (EOP). These came into force on 1 January 1989 at all Swedish nuclear plants.

All these studies have focused on the containment integrity and the impact on the environment. In recent years many studies have been performed to analyse the long term effects within the plant, after minor and severe core accident scenarios. Possibilities of bringing the plant back to an operating condition are analysed in these studies. These analyses are performed in co-operation between different companies and the authorities in the Nordic Liaison Committee for Atomic Energy and in one case (the FRIPP Project) within the Swedish State Power Board.

The results of these studies are briefly presented in this paper.

Two case are finished and one (the FRIPP Project) will continue until 1991.

I. 1% core damage study: Management of Radioactive Waste Resulting from Nuclear Fuel Damage in a BWR Power Plant

   (a) LOCA with a reduced capacity of emergency core cooling system at end of cycle operation. This causes 1% of the fuel rods to be damaged (pinholes).
   (b) Blockage of a fuel channel at end of cycle operation. The isolation valves are closed manually after 30 min.

   The distribution of radioactive water/steam throughout the plant is being studied. The consequences of high dose rates and the handling of leakages from systems are evaluated. How to clean the water and how to handle waste products such as activated resins and solid waste are analysed.
The results of this study show that

1. The two cases (a) and (b) give almost the same gamma source term to the reactor water.
2. Total activity discharged to the waste plant after 12 weeks will be 880 TBq (24 kCi).
3. Total amount of fission products after 12 weeks to the waste plants is in case (a) 0.09 TBq (2.5 Ci) of alpha activity, in case (b) 1.8 TBq (50 Ci) of alpha activity.
4. The caesium isotopes dominate the water media in the waste plant after 12 weeks.
(5) The accessibility of the reactor building will be reduced. A system containing activated water will from a DN 200 pipe (8 inch diameter) give a dose rate, 1 m from surfaces, of 7 mSv/h (=700 mR/h) after 2 weeks (Fig. 1).

(6) The accessibility of the waste handling plants will be reduced. However, with a well planned strategy for using storage tanks and filters the situation will not cause any extreme problems.

(7) The accessibility to the turbine building and its condenser will be restricted for at least 2 weeks in case (b).

(8) Operations of organic filters for 30 minutes in the contaminated water in the first part of the sequence (before closing the isolation valves) will cause a dose in the filters of magnitudes >1 MGy after 12 weeks.

(9) Ventilation of containment to the environment can be performed earlier than 4 weeks after the event, remaining well inside acceptable release levels.

(10) In case (a) the containment is filled with water to the top edge of the core and this causes much contamination on the walls, floors and the components.

(11) In case (b) core debris will be spread into reactor systems in operation. The gamma doses from the pipes will increase slightly because of the debris. In certain positions in the pipelines extra precautions must be taken.

(12) The cleanup procedure can use ordinary organic filters. Approximately 1000 waste drums will be produced from cleaning the water to normal operating values.

(13) It will take three shifts four months to clean the water.

(14) If some proposed modifications are performed the ability to handle the situation could be strengthened.

II. 10% CORE DAMAGE STUDY

This study was performed by ABB-ATOM (sponsored by the Nordic Liaison Committee for Atomic Energy). The study analysed an event that starts with a LOCA followed by loss of electric power for 35 minutes. The event occurs at end of cycle operation. The fuel pin cladding is pinholed to the extent that 10% of noble gases are released. No degradation of the core structure will occur.

The 10% of noble gases are released by 3% in burst release and 7% by other mechanisms. This will, according to the MAAP analysis, cause

- 7% of Cs and I nuclides to be released
- 1.5% of Te and Ag nuclides to be released and
- 0.16 kg uranium.

In this event no cleanup activities are performed within three months after the event.

The containment is assumed to have a leak rate of 1 m³/day.

The analysis gave the following main results:
(1) The amount of gamma sources in this case is the same as in the Three Mile Island (TMI-2) event. The calculated inventory of $^{134}$Cs and $^{137}$Cs in the TMI-2 core was $7 \times 10^3$ and $3.3 \times 10^4$ TBq, respectively. Owing to the very short operating time this is much lower than in the reference reactor. In TMI-2 about 75% of the core inventory of caesium was released from the fuel and since the corresponding value of the reference reactor has been estimated at 7% the amount of these caesium isotopes released is about the same in the two reactors.

(2) Gamma source in containment
   - after 1 week — 800 000 TBq
   - after 3 months — 3 300 T bq.

(3) Actinide sources in solution after the event: $6.3 \times 10^{-3}$ TBq.

(4) Caesium dominates the gamma sources, accounting for 99%.

(5) Emergency core cooling pipe (DN 250 = 10 inch diameter) gives the following dose rates at 1 m distance
   - after 3 days — 180 mSv/h
   - after 14 days — 72 mSv/h (10 times more than in the 1% case)
   - after 90 days — 46 mSv/h.

(6) If a valve is opened on a sampling line when the event occurs it will be very difficult to close within a 14 day period.

(7) Inspections and minor services of components will be impossible for at least 3 weeks for systems contaminated or containing the contaminated water. Repair work will involve large shielding efforts if it has to be done within the first month on contaminated components.

(8) Organic resin cannot be used in the cleanup activities. Zeolite filters are recommended (cf. TMI-2). These have to be built in the plant after the event.

(9) It will be necessary to build a return line from the waste plants to the containment so as to be able to handle excess water in the waste handling plant.

(10) The capacity of the solidification process determines the time for cleanup.
   - with a waste matrix with a surface dose rate of 30 mSv/h the water cleaning will take about 4 years (= 3 500 concrete moulds)
   - with a waste matrix with a surface dose rate of 300 mSv/h the water cleaning will take about 2 years (= 800 concrete moulds).

(11) The decontamination of the containment will involve similar problems to those encountered by TMI-2.

(12) Filling up the containment will cause a pressure increase in the containment. This makes ventilation and releases to the environment necessary. Strategies to handle this must be developed so that it will be acceptable to the public.

To sum up the 1% and 10% cases, the studies show that the consequences of core damage without core meltdown or core structure degradation can challenge the plant, plant system operation, waste handling and radiation protection considerably.
All nuclear power plants ought to have studied a similar situation, so as to be prepared for handling such an emergency.

III. CORE MELTDOWN STUDY (FRIPP)

This study is in progress and will be finished in 1991. The work is being done by the Swedish State Power Board. The analysis assumes that a core meltdown has occurred and that part of the core has passed into the bottom of the containment. Results from MAAP calculations for the first 24 hours give the starting point for this study.

The project includes many subprojects, e.g.

— source term calculations
— dose rates in the containment
— hydrogen production
— chemistry balance
— cooling capacity
— system operation strategies for 5 years with core and most of the contaminated water in the containment
— waste handling strategies.

Very few results are ready for presentation as of today.

Figure 1 shows that active cooling is necessary for almost 5 years if the containment’s design limits are not to be exceeded and filtered ventilation has to be used several times.
ФОРМИРОВАНИЕ ДОЗ ОБЛУЧЕНИЯ В ПЕРИОД ЛИКВИДАЦИИ ПОСЛЕДСТВИЙ КРУПНОЙ РАДИАЦИОННОЙ АВАРИИ

О.А. КОЧЕТКОВ, В.П. КРЮЧКОВ, В.А. КУТЬКОВ, Л.Г. ЛАПА, Д.П. ОСАНОВ, В.И. ПОПОВ
Институт биофизики Минздрава СССР, Москва,
Союз Советских Социалистических Республик

Abstract—Аннотация

ACCUMULATION OF THE EXPOSURE DOSE DURING OPERATIONS TO DEAL WITH THE CONSEQUENCES OF A SERIOUS RADIATION ACCIDENT.

Intensive contamination of the environment follows, as a rule, any serious radiation accident, resulting from the release of radioactive products of nuclear fuel outside the reactor core. The Chernobyl accident, for example, has resulted in the dispersion of more than 50 MCI of radioactive products over a large area. However, a significant amount of them has fallen within the 30-km zone, which has led to the intensive source formation of gamma and beta radiation and aerosols as well. Air contamination with aerosols has resulted from the products of the initial explosion, radionuclide fragments having being carried out of the damaged reactor by a hot air flow arising from long term graphite burning. These aerosols contain particles of fuel assemblies and fragments of mononuclides. The physico-chemical properties of aerosols, their dispersion characteristics, their nuclide composition and concentrations are described in the paper. It also discusses investigations into 'hot particles'. Three groups of persons have been established, depending on exposure conditions. The first group embraces population evacuated from the 30-km zone, the second relates to the occupational personnel of the NPP involved in recovery from the accident and the last group comprises the personnel involved later in dealing with the accident's consequences. Data on internal exposure doses of the groups are given. It was found that the internal exposure of all organs and tissues (bone tissue and red bone marrow are excluded) reaches its full value within a year after the accident, the thyroid being the critical organ. In subsequent years transuranic elements determine almost all the internal exposure. When $\beta_g$, the average geometric ratio, was about 3, the whole body content was equal to 150 Bq and 70 Bq 120 and 460 days after the accident, respectively. The gamma radiation predominates in the exposure of personnel, although in a number of operations beta and low energy gamma radiation have made a considerable contribution to the dose. The paper analyses the dose distribution and identifies its character as log-normal for the personnel of Chernobyl NPP, while for the external personnel involved, the dose distribution differs from log-normal.
местности и пространства. Так, в результате аварии на Чернобыльской АЭС более 50 МКи радиоактивных веществ было рассеяно на огромной территории, основная масса которых, однако, выпала в пределах 30-км зоны. Это привело к образованию мощного источника гамма- и бета-излучения, а также интенсивного источника аэрозолей. Загрязнение воздушной среды аэрозолями формировалось за счет продуктов первого взрыва, истечения паров осколочного радионуклида из реактора в результате длительного горения графита. Поэтому аэрозоли содержали частицы топливной матрицы и частицы мононуклидного состава. В работе приведены физико-химические свойства аэрозолей, их дисперсные характеристики, нуклидный состав и концентрации. В том числе рассмотрены "горячие частицы". Показано, что по условиям облучения можно выделить три группы лиц, это - население, которое было эвакуировано из 30- км зоны, персонал АЭС, участвующий в ликвидации аварии, и персонал, привлекаемый в дальнейшем к ликвидации последствий аварии. Для указанных трех групп представлены данные о дозах внутреннего облучения. Показано, что внутреннее облучение всех органов и тканей (за исключением костной ткани и красного костного мозга) реализуется практически полностью за первый год после аварии, причем критическим органом является щитовидная железа. Внутреннее облучение в последующие годы почти целиком определяется трансурановыми элементами и системное содержание на 120 сутки после аварии составило (медиана) 150 Бк, на 460-е сутки - 70 Бк при среднем геометрическом отношении $\beta_y$ около 3. Ведущую роль в облучении персонала имело гамма-излучение, хотя при выполнении ряда операций значительный вклад в дозу вносил бета- и низкоэнергетическое гамма-излучение. В докладе проанализировано распределение доз и показано, что, если для персонала ЧАЭС имеет место лог-нормальное распределение, то для привлеченного персонала оно отличается от лог-нормального.

В результате крупной радиационной аварии, связанной с выходом радиоактивных продуктов и ядерного топлива из реактора, происходит интенсивное загрязнение местности и пространства. Так, в результате аварии на Чернобыльской АЭС [1, 2] более 50 МК радиоактивных веществ было рассеяно на огромной территории, значительная часть которых выпала в пределах 30-км зоны, где проводились основные восстановительные работы. Загрязнение местности привело к образованию мощного источника бета-гамма-излучения, а также интенсивного источника аэрозолей.

В формировании выброса главную роль играли два процесса:

- диспергирование ядерного топлива в результате взрыва 26.04.86 г., приведшее к образованию аэrozоля, состоящего из частиц топливной матрицы, содержащей весь спектр радионуклидов продуктов деления;
ТАБЛИЦА I. ВКЛАД (%) ИЗОТОПОВ В АКТИВНОСТЬ ТОПЛИВНОЙ МАТРИЦЫ КАК ФУНКЦИЯ ВРЕМЕНИ ПОСЛЕ АВАРИИ НА ЧАЭС

<table>
<thead>
<tr>
<th>Изотоп</th>
<th>Время, сут</th>
<th>1</th>
<th>10</th>
<th>120</th>
<th>365</th>
<th>730</th>
<th>1095</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sr-89</td>
<td></td>
<td>2.264</td>
<td>4.740</td>
<td>3.841</td>
<td>0.348</td>
<td>0.004</td>
<td>0.000</td>
</tr>
<tr>
<td>Sr-90</td>
<td></td>
<td>0.157</td>
<td>0.373</td>
<td>1.356</td>
<td>3.490</td>
<td>6.395</td>
<td>9.432</td>
</tr>
<tr>
<td>Y-90</td>
<td></td>
<td>0.167</td>
<td>0.375</td>
<td>1.357</td>
<td>3.491</td>
<td>6.397</td>
<td>9.434</td>
</tr>
<tr>
<td>Y-91</td>
<td></td>
<td>2.910</td>
<td>6.201</td>
<td>6.176</td>
<td>0.887</td>
<td>0.022</td>
<td>0.000</td>
</tr>
<tr>
<td>Zr-95</td>
<td></td>
<td>3.645</td>
<td>7.829</td>
<td>8.716</td>
<td>1.604</td>
<td>0.058</td>
<td>0.002</td>
</tr>
<tr>
<td>Nb-95</td>
<td></td>
<td>3.494</td>
<td>8.262</td>
<td>15.527</td>
<td>3.479</td>
<td>0.128</td>
<td>0.004</td>
</tr>
<tr>
<td>Ru-103</td>
<td></td>
<td>3.017</td>
<td>6.098</td>
<td>3.221</td>
<td>0.112</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>Ru-106</td>
<td></td>
<td>0.811</td>
<td>1.890</td>
<td>5.633</td>
<td>9.287</td>
<td>8.771</td>
<td>6.668</td>
</tr>
<tr>
<td>I-131</td>
<td></td>
<td>2.032</td>
<td>2.254</td>
<td>0.001</td>
<td>0.000</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>Cs-134</td>
<td></td>
<td>0.106</td>
<td>0.250</td>
<td>0.830</td>
<td>1.728</td>
<td>2.314</td>
<td>2.508</td>
</tr>
<tr>
<td>Cs-137</td>
<td></td>
<td>0.192</td>
<td>0.454</td>
<td>1.651</td>
<td>4.253</td>
<td>7.804</td>
<td>11.526</td>
</tr>
<tr>
<td>Ba-140</td>
<td></td>
<td>3.845</td>
<td>5.591</td>
<td>0.053</td>
<td>0.000</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>La-140</td>
<td></td>
<td>4.181</td>
<td>6.421</td>
<td>0.061</td>
<td>0.000</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>Ce-141</td>
<td></td>
<td>3.665</td>
<td>7.166</td>
<td>2.516</td>
<td>0.035</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>Ce-144</td>
<td></td>
<td>2.589</td>
<td>5.998</td>
<td>16.818</td>
<td>24.200</td>
<td>18.659</td>
<td>11.583</td>
</tr>
<tr>
<td>Pr-143</td>
<td></td>
<td>3.530</td>
<td>5.955</td>
<td>0.075</td>
<td>0.000</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>Pr-144</td>
<td></td>
<td>2.589</td>
<td>5.998</td>
<td>16.818</td>
<td>24.200</td>
<td>18.659</td>
<td>11.585</td>
</tr>
<tr>
<td>Np-239</td>
<td></td>
<td>27.022</td>
<td>4.481</td>
<td>0.000</td>
<td>0.000</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>Pu-241</td>
<td></td>
<td>0.238</td>
<td>0.562</td>
<td>2.032</td>
<td>5.266</td>
<td>9.211</td>
<td>13.273</td>
</tr>
</tbody>
</table>

АЛЬФА-ИЗЛУЧАТЕЛИ

<table>
<thead>
<tr>
<th>Изотоп</th>
<th>Время, сут</th>
<th>1</th>
<th>10</th>
<th>120</th>
<th>365</th>
<th>730</th>
<th>1095</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-238</td>
<td></td>
<td>0.004</td>
<td>0.010</td>
<td>0.039</td>
<td>0.106</td>
<td>0.201</td>
<td>0.303</td>
</tr>
<tr>
<td>Pu-239</td>
<td></td>
<td>0.001</td>
<td>0.002</td>
<td>0.007</td>
<td>0.017</td>
<td>0.032</td>
<td>0.049</td>
</tr>
<tr>
<td>Pu-240</td>
<td></td>
<td>0.001</td>
<td>0.001</td>
<td>0.005</td>
<td>0.012</td>
<td>0.023</td>
<td>0.034</td>
</tr>
<tr>
<td>Am-241</td>
<td></td>
<td>0.000</td>
<td>0.000</td>
<td>0.002</td>
<td>0.010</td>
<td>0.034</td>
<td>0.074</td>
</tr>
<tr>
<td>Cm-242</td>
<td></td>
<td>0.093</td>
<td>0.212</td>
<td>0.487</td>
<td>0.449</td>
<td>0.179</td>
<td>0.057</td>
</tr>
<tr>
<td>Cm-244</td>
<td></td>
<td>0.002</td>
<td>0.004</td>
<td>0.015</td>
<td>0.037</td>
<td>0.067</td>
<td>0.098</td>
</tr>
</tbody>
</table>

Таким образом формирование доз облучения персонала, участвующего в ликвидации последствий Чернобыльской аварии, происходило за счет гамма-излучения, интенсивного бета-излучения (особенно в первые дни после аварии), и за счет ингаляции аэрозолей. Пероральный путь поступления отсутствовал, так как питание персонала осуществлялось только привозными продуктами.

В данном сообщении не рассматриваются проблемы аварийного облучения, т. е. тех уровней облучения, которые формировались в период аварии, т. е. в...
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>0-4</td>
<td>2326</td>
<td>834</td>
<td>639</td>
<td></td>
</tr>
<tr>
<td>0.4-1.2</td>
<td>235</td>
<td>1753</td>
<td>2890</td>
<td></td>
</tr>
<tr>
<td>1.2-2.4</td>
<td>369</td>
<td>145</td>
<td>1126</td>
<td></td>
</tr>
<tr>
<td>2.4-4.0</td>
<td>424</td>
<td>428</td>
<td>193</td>
<td></td>
</tr>
<tr>
<td>4.0-5.0</td>
<td>145</td>
<td>82</td>
<td>19</td>
<td></td>
</tr>
<tr>
<td>5.0-10</td>
<td>547</td>
<td>19</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>10-25</td>
<td>565</td>
<td>3</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>25</td>
<td>142</td>
<td>3</td>
<td>0</td>
<td></td>
</tr>
</tbody>
</table>
течение первых нескольких дней аварии на ЧАЭС. Эти данные были доложены МАГАТЭ [3], так же как были опубликованы данные о медицинских последствиях аварии на ЧАЭС. Как известно [4], радиационная обстановка в зоне аварийного блока и вокруг него стабилизировалась в начале мая 1986 г. и характеризовалась высокими уровнями излучения. Изотопный состав выпадений в ближней зоне был относительно одинаков, как на грунте так и в воздухе. Эти данные приведены в табл. I.


![Гистограммы распределения $lg D$ для персонала ЧАЭС за 1986 и 1987 гг.](image)

В 1986 г. во многих случаях определяющим фактором облучения персонала являлось бета- и низкоэнергетическое гамма-излучение. В зависимости от места проведения работ, условий облучения и степени дезактивации отношение доз бета- и низкоэнергетического гамма-излучения к дозам жесткого гамма-излучения на поверхности детекторов изменялось от 10 до 240 и было обусловлено бета-излучением с энергией 0,9 — 3,5 МэВ.

Вклад в дозу гамма-излучения с эффективной энергией около 20 кэВ, не регистрируемого штатными индивидуальными дозиметрами, достигал 6%. Максимальные значения отношения доз в коже рук к дозам на все тело равные 100 — 300 наблюдались при поиске и уборке высокоактивных предметов на крышах III и IV энергоблоков. Там, где первичная дезактивация была завершена, как показали фантомные исследования, средний вклад бета- и низкоэнергетического гамма-излучения составлял от 1,1 до 3,0 на глубине 7 мг/см² по отношению к жесткому гамма-излучению. Наибольшее отношение наблюдалось в тех местах, где эффективность дезактивации была низкой и тогда доза бета-излучения до 10 раз превосходила дозу жесткого гамма-излучения. Полученные в 1987 г. экспериментальные данные представлены на рис. 2 и они однозначно свидетельствуют о снижении роли внешнего бета- и низкоэнергетического гамма-излучения в формировании радиационной обстановки. Вместе с тем, очевидно, что и теперь, через 2—3 года после аварии их роль достаточно заметна. Таким образом в системе аварийной ИДК необходимо иметь средства контроля уровней облучения кожи и хрусталиков глаз.

Дозы внутреннего облучения определяли расчетным путем на основе исследований физико-химических свойств аэрозоля и их концентрации в воздухе в течение 1986—1987 гг., а также по результатам фактических измерений поступления и содержания радионуклидов в организме персонала ЧАЭС и прикомандированных лиц за тот же период. Проводили раздельное определение для альфаизлучающих нуклидов трансурановой группы и для бета-излучающих нуклидов осколочного происхождения.

Характеристику воздушной среды, а именно, концентрацию, изотопный и дисперсный состав радиоактивных аэрозолей определяли методом отбора проб на аналитические аэрозольные фильтры с их последующим радиохимиче-
Рис. 2. Гистограмма экспериментально измеренного отношения мощностей доз β- и γ-излучений к мощности дозы жесткого γ-излучения для 1987 года $(P_{β+γ}/P_γ)$.

Ским анализом и измерением, а также методом инерционного разделения аэрозольных частиц с помощью каскадного импактора. После разделения частиц на фракции по размерам измеряли активность и изотопный состав каждой фракции. Биологическую транспортабельность частиц оценивали методом диализа по двухсуточному содержанию активности растворимой фракции изотопов в растворе Рингера.

Предельно допустимое годовое поступление (ПДП) для смеси радионуклидов рассчитывали, исходя из основных дозовых пределов согласно НРБ-76/87, но с учетом дисперсности и транспортабельности аэрозолей и рекомендаций МКРЗ [5]. Для описания распределения активности нуклидов по размерам аэрозольных частиц использовали логнормальную функцию.
Реальное поступление и содержание нуклидов в организме было определено на основании данных обследования в биофизической лаборатории посредством измерения содержания трансурановых элементов (ТУЭ) и изотопов стронция в выделениях (моча, кал), в пробах мазков из носа, а также по результатам измерения гамма-излучающих нуклидов на счетчике излучения человека.

Обследование в биофизической лаборатории проводили в период с мая 1986 г. по сентябрь 1987 г. Оценка поступления ТУЭ в организм привлеченных к ликвидации последствий аварии людей (группа К) осуществлена по результатам измерения активности альфа-излучения в пробах кала (содержание ТУЭ в моче было ниже минимально измеряемой активности). Значение суммарного поступления определяли как произведение измеренного содержания ТУЭ в суточном количестве кала на длительность контакта. Определяемое таким способом поступление сопоставляли с расчетным значением ПДП на момент измерения.

По результатам биофизического обследования лиц группы К в 1986 г. (127 чел.) максимальное суточное поступление (по критерию 3 б) составило 19 Бк, что в предположении длительности работы в зоне 900 часов может привести к максимальному суммарному поступлению около 1,7 кБк (46 нКи) или менее 0,5 ПДП.

Поступление было также оценено по методике альфа-радиометрии проб мазков из носа. По результатам обследования 96 чел. среднее суточное ингаляционное поступление не превышало двух допустимых за сутки, что для принятой продолжительности работ приводит к поступлению не более 1 ПДП.

Приведенные оценки характеризуют возможное предельное поступление, поскольку суммарная длительность работ для лиц группы К в большинстве случаев из-за ограничений по дозе от внешнего облучения находилась в пределах до одного месяца.

Средняя по двум методам оценка поступления ТУЭ составляет 0,8 ПДП и хорошо согласуется с расчетными данными (1 — 5 ПДП по измеренным характеристикам воздушной среды) за 1986 г.

Следует отметить, что обследованный в 1986 г. контингент группы К был занят на разнообразных работах с повышенной опасностью ингаляционного поступления (допусковые работы внутри станции в недезактивированных помещениях, на территории промплощадки вблизи III и IV блоков, по расчистке крыши III блока от обломков твэлов, на пунктах санитарной обработки автомобилей и т. п.), причем работы выполнялись с обязательным применением средств защиты органов дыхания.
В 1987 г. было обследовано 94 чел. и получено, что максимальное поступление не превышало 0,2 ПДП для лиц, работающих непосредственно на ЧАЭС, и менее 0,1 ПДП для лиц, работающих в других зонах.

Необходимо подчеркнуть, что с учетом реального времени работы медианное значение поступления по результатам обследования в БФЛ в 20—30 раз меньше полученной максимальной оценки (0,8 ПДП и 0,2 ПДП в 1986 г. и 1987 г., соответственно).

Приведенные данные свидетельствуют в целом об удовлетворительном совпадении результатов обследования в биофизической лаборатории (БФЛ) и расчетных оценок поступления вторичных аэрозолей ТУЭ в связи с работами по ликвидации последствий аварии летом и осенью 1986 г. и в 1987 г. Вместе с тем, эти данные свидетельствуют о необходимости выборочного контроля поступления ТУЭ при работах по ликвидации последствий аварии.

Значительно большее поступление аэрозолей ТУЭ наблюдалось для лиц, находившихся в момент аварии на ЧАЭС или вблизи нее (группа А). Для этих лиц характерно относительно кратковременное (в первые 1—2 недели после аварии) ингаляционное поступление первичных аэрозолей из реактора с последующим либо полным, либо частичным прекращением контакта на различные сроки (эвакуация, обследование в больнице, отпуск и т. п.). Для этой группы оценка поступления проведена на основании определения содержания ТУЭ в организме по результатам радиометрии альфа-активности в суточных пробах мочи.

Для определения содержания активности в организме использована функция суточной экскреции плутония с мочей, полученная Лангхемом [6] и соответствующая ей (с учетом также экскреции с калом) функция удержания активности в организме. Возможность применения последней подтверждена данными измерений проб секционного материала погибших от острой лучевой болезни. Сравнение суммарного системного содержания альфа-активности по данным измерения секционного материала с активностью, определенной из результатов измерений проб мочи при жизни показало, что оценка содержания по функции Лангхема дало завышение результата до двух раз.


Реалистичность полученной оценки подтверждается содержанием альфа-активности, измеренной в секционном материале для погибших от ОЛБ спустя 15—30 суток после аварии. Содержание ТУЭ в скелете, печени и почках для большинства из них (22 чел.) находилось в диапазоне от 1,5 до 1400 Бк. Еще
Таблица III. Распределение содержания трансурановых элементов в организме для лиц группы А по данным обследования в 1986—1987 гг.

<table>
<thead>
<tr>
<th>Содержание</th>
<th>1986 г.</th>
<th>1987 г.</th>
<th>740-2500</th>
<th>740-1300</th>
</tr>
</thead>
<tbody>
<tr>
<td>альфа излучателей в организме, Бк</td>
<td>1,5</td>
<td>1,5-370</td>
<td>370-740</td>
<td>740-1300</td>
</tr>
</tbody>
</table>

| Число лиц (в скобках-%) от общего числа | 1986 г. | 1987 г. | | |
|----------------------------------------|---------|---------| | |
| 28 (27) | 65 (68) | 56 (54) | 21 (22) | 5 (5,3) | 4 (4,2) | |

у трех человек активность была менее 1,5 Бк, а у одного измерено около 14 кБк. Даже учитывая последнее исключительно высокое содержание (объяснимое местонахождением в момент взрыва), диапазон изменения системного содержания близок к тому, что было найдено для лиц группы А. Если учесть, что погибшие находились, хотя и недолго (от 1 до 6 часов), в наиболее опасных с точки зрения ингаляции аэрозолей ТУЭ условиях, т. е. вблизи разрушенного реактора и во время первичного массированного выброса, то полученные результаты (табл. III) можно считать достаточно правдоподобными.

В соответствии с данными о транспортабельности аэрозолей ТУЭ можно допустить, что аэрозоли обладали транспортабельностью не хуже 2%, что позволяет отнести их к классу легочной модели МКРЗ. Это допущение подтверждается результатами анализа секционного материала, если принять, что поступление происходило исключительно через легкие.

Для восстановления поступления у лиц группы А воспользовались функциями удержания для соединений, относящихся к классу для всех нуклидов ТУЭ, а его значение определяли для среднемедианного диаметра (АМАД) = 1 и 10 мкм.

По измеренному системному (вне легких) содержанию активности (150 Бк на 120 сутки после аварии) получили искомое поступление ТУЭ для различной дисперсности. При расчетах поступления и оценках доз приняты нуклидный состав, измеренный в мае-июне 1986 г.

Для определения содержания в организме изотопов стронция использовали функцию удержания для однократного поступления стронция, рекомендованную МКРЗ [7].
Из результатов измерений содержания $^{90}\text{Sr}$, полученных в 1986 г., следует, что среднее содержание изотопов в организме у 105 обследованных (группа А) составляет около 1 кБк или 1% допустимого содержания ($\text{ДС}_\text{A} = 74 \text{ кБк}$) для растворимых соединений $^{90}\text{Sr}$, причем максимальное содержание равно 0,23 $\text{ДС}_\text{д}$. Эти данные свидетельствуют о том, что основное поступление происходило в первые недели после аварии, что подтверждается также незначительным содержанием стронция у лиц из группы К. Данные за 1987 г. для лиц из группы А (85 человек) также свидетельствуют о том, что заметного дополнительного поступления $^{90}\text{Sr}$ не наблюдалось, поскольку среднее содержание (0,6 кБк) с учетом выведения хорошо соответствует содержанию, обнаруженному в 1986 г. (1 кБк). Полученные данные хорошо согласуются с результатами измерения $^{90}\text{Sr}$ в секционном материале погибших (среднее содержание 0,6 кБк). Данные о содержании стронция соответствуют поступлению около 5 кБк или менее 0,1 ПДП.

Для группы К содержание стронция в организме в 1986 и 1987 гг. было приблизительно на одном уровне, т. е. около 0,5% $\text{ДС}_\text{A}$ по результатам обследования 144 чел., или в 20 раз меньше, чем для группы А.

Расчеты доз выполнены в рамках легочной модели МКРЗ для условного (стандартного) человека с помощью двух программ, одна из которых позволяет получить данные, связанные с биокинетикой радионуклида, а другая рассчитывать дозовые нагрузки на различные органы — мишени условного человека [8].

При расчетах дозовых нагрузок за счет ТУЭ входные данные были получены для 1 года, 10 и 50 лет и для аэрозолей с АМАД равным 1 и 10 мкм по результатам оценки ингаляционного поступления. Так как при измерениях секционного материала большая часть активности зарегистрирована в костной ткани (в соотношении 14 : 1), то при расчетах для курия и америция принято, что доли, перемещаемые из переходной камеры в костную ткань и печень, составляют 0,8 и 0,1, а не по 0,45, как это принято МКРЗ.

### Таблица IV. СРЕДНИЕ ОЖИДАЕМЫЕ ЭКВИВАЛЕНТНЫЕ ДОЗЫ ДЛЯ ПРОФЕССИОНАЛОВ ЗА СЧЕТ ИНГАЛЯЦИОННОГО ПОСТУПЛЕНИЯ ТРАНС-УРАНОВЫХ ЭЛЕМЕНТОВ, мЗв

<table>
<thead>
<tr>
<th>Время, лет</th>
<th>Красный костный мозг</th>
<th>Легкие</th>
<th>Костная ткань</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1 мкм</td>
<td>10 мкм</td>
<td>1 мкм</td>
</tr>
<tr>
<td>1</td>
<td>11</td>
<td>10</td>
<td>25</td>
</tr>
<tr>
<td>10</td>
<td>29</td>
<td>26</td>
<td>–</td>
</tr>
<tr>
<td>50</td>
<td>107</td>
<td>96</td>
<td>–</td>
</tr>
</tbody>
</table>
ТАБЛИЦА V. СРЕДНИЕ ОЖИДАЕМЫЕ ЭКВИВАЛЕНТНЫЕ ДОЗЫ ДЛЯ ПРОФЕССИОНАЛОВ ЗА СЧЕТ ИНГАЛЯЦИОННОГО ПОСТУПЛЕНИЯ ГАММА-ИЗЛУЧАЮЩИХ РАДИОНУКЛИДОВ, мЭв

<table>
<thead>
<tr>
<th>Время, лет</th>
<th>Красный костный мозг</th>
<th></th>
<th>Леткие</th>
<th></th>
<th>Костная ткань</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>АМАД = 1 мкм</td>
<td>АМАД = 10 мкм</td>
<td>АМАД = 1 мкм</td>
<td>АМАД = 10 мкм</td>
<td>АМАД = 1 мкм</td>
<td>АМАД = 10 мкм</td>
</tr>
<tr>
<td>1</td>
<td>1,2</td>
<td>2,4</td>
<td>80</td>
<td>60</td>
<td>6,0</td>
<td>15</td>
</tr>
<tr>
<td>50</td>
<td>1,3</td>
<td>2,5</td>
<td>100</td>
<td>75</td>
<td>6,3</td>
<td>16</td>
</tr>
</tbody>
</table>
Приведенные в табл. IV ожидаемые эквивалентные дозы рассчитаны по программе на ЭВМ с использованием коэффициентов качества согласно [12]. Представлены дозы для максимально облучаемых органов в каждой из трех групп критических органов. Для рассматриваемой смеси ТУЭ криторганом является костная ткань, средняя доза в которой составляет около 0,4 ПДП за первый год и по 0,1 ПДП за все последующие годы. Суммарное облучение за 50 лет соответствует поступлению порядка 4 ПДП.

В соответствии с данными обследованиями имеется группа лиц, содержание ТУЭ у которых заметно больше среднего, что приводит к дозам в костной ткани около или более 1 ПДП за первый год. Из числа обследованных (199 чл.) в эту группу входят 5 человек (персонал ЧАЭС), содержание ТУЭ у которых при усреднении результатов не менее 3—5 измерений превышает 370 Бк.

Доза для 90 Sr, рассчитанная для полученного ранее усредненного поступления, значительно меньше не только ПДД, но и предела дозы для населения.

Что касается 89 Sr, то этот нуклид регистрировали в выделениях до августа 1986 г. и среднее его содержание по результатам обследования 62 чел. соответственно поступлению около 1 кБк, что приводит к дозам в пределах 0,01-0,02 мЗв для красного костного мозга и костной ткани. Указанная доза полностью реализуется за один год после аварии.

Для оценки доз от других радионуклидов были использованы данные измерений на установке СИЧ (геометрия измерения: детектор расположен со стороны спины в нижней области лопаток лежащего на кушетке человека). Одновременно с помощью радиометра с сцинтилляционным счетчиком проводили измерения мощности дозы от области щитовидной железы, вплотную к ней. Для оценок доз использованы результаты измерений, выполненные с 17.05.86 г. по 13.06.86 г. для 247 чел., из числа которых 164 — персонал АЭС, а остальных можно отнести к категории населения. Большинство из этих лиц были обследованы на содержание ТУЭ и стронция в биофизической лаборатории.

Учитывая, что аэрозоли некоторых нуклидов были представлены в виде двух фракций, расчет поступления и доз был выполнен для АМАД = 1 и 10 мкм.

Полученные значения доз (табл. V) слабо зависят от дисперсности аэрозолей, что позволяет принять усредненные оценки. Критическим органом для рассмотренных нуклидов являются легкие, средняя доза для которых составляет около 0,5 ПДД за первый год, в течение которого реализуется 80% ожидаемой дозы за 50 лет. Основной вклад в дозу на легкие вносят 106 Ru (55,7%), 239 Np (20%) и 144 Ce (до 10%).
Эти данные получены только для фактически измеренных радионуклидов без учета воздействия многих короткоживущих. Анализ функции изменения интегрального энерговыделения в топливе реактора РБМК-1000 за время выдержки от 1 часа до 10 суток показывает, что оно уменьшается приблизительно в 5 раз (суммарно по бета-гамма-излучению). Вероятно, примерно такую же кратность можно принять для оценки максимальной дозы для легочной ткани за первый год после аварии.

Для этих же лиц рассчитана доза для щитовидной железы по результатам измерений мощности дозы. Для группы профессионалов (164 человека) средняя доза равна 240 мЗв (медиана 200 мЗв). Если учесть вклад других изотопов йода, то получаем ± 340 мЗв, т. е. около 2,7 ПДД. С учетом погрешности эти результаты хорошо согласуются с опубликованными данными и известными нам данными других измерений.

Суммарная доза от внутреннего облучения за счет ТУЭ, изотопов стронция и гамма-излучающих нуклидов в рассмотренной модели метаболизма нуклидов представлена в табл. VI.

Альтернативой рассмотренной выше модели формирования доз внутреннего облучения у участников ликвидации последствий аварии на ЧАЭС является модель, в которой принято, что осколочные и трансурановые радионуклиды (табл. I) прочно удерживаются матрицей двуокиси урана. Радионуклиды в такой матрице должны приобретать специфические свойства — в органах дыхания и пищеварения они ведут себя подобно оксидам урана, т. е. соединениям, принадлежащим к ингаляционному классу "У" и имеющим коэффициент резорбции в желудочно-кишечном тракте, равный 0,002. Тогда согласно этой модели основным органом-депонентом радиоактивного вещества являются легкие. Этот вывод также не противоречит имеющимся данным по распределению радиоактивности в организме погибших от острой лучевой болезни, если принять, что при значительном радиоактивном загрязнении кожных покровов в сочетании со значительными термическими и радиационными ожогами происходило непосредственное поступление радиоактивного вещества в кровь. Поступление радиоактивности в организм по двум путям могло привести к соотношениям легкие—остальные органы, характерным для транспортабельных соединений радионуклидов.

Для восстановления первоначального поступления аэrozоля топливной матрицы в органы дыхания были использованы результаты измерения содержания ${}^{95}\text{Zr}$ в теле лиц указанной категории, выполненных на установке СИЧ в период с 16.05.86 г. по 31.08.86 г. Показано, что среди обследованных имеется значительная часть людей, в организме которых содержатся частицы топливной матрицы. Максимальное первоначальное поступление радиоактивности для лиц
ТАБЛИЦА VI. СУММАРНЫЕ ДОЗЫ ОТ ВНУТРЕННЕГО ОБЛУЧЕНИЯ ЗА СЧЕТ ИНГАЛЯЦИОННОГО ПОСТУПЛЕНИЯ РАДИОНУКЛИДОВ В РЕЗУЛЬТАТЕ АВАРИИ НА ЧАЭС ДЛЯ ПРОФЕССИОНАЛОВ, мЗв
(в скобках по отношению к ПДД)

<table>
<thead>
<tr>
<th>Время, лет</th>
<th>Красный костный мозг</th>
<th>Легкие</th>
<th>Щитовидная железа</th>
<th>Костная ткань</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>13 (0,26)</td>
<td>84 (0,56)</td>
<td>410 (2,7)</td>
<td>120 (0,40)</td>
</tr>
<tr>
<td>10</td>
<td>31 (0,06)</td>
<td>100 (0,07)</td>
<td>-</td>
<td>350 (0,12)</td>
</tr>
<tr>
<td>50</td>
<td>110 (0,04)</td>
<td>100 (0,01)</td>
<td>-</td>
<td>1300 (0,09)</td>
</tr>
</tbody>
</table>

Таблица VI показывает, что суммарные дозы от внутреннего облучения за счет ингаляционного поступления радионуклидов в результате аварии на ЧАЭС для профессионалов, выраженные в мЗв, в пределах от 13 до 110 мЗв. Величина дозы зависит от времени после аварии, с наибольшей дозой за первый год (до 410 мЗв). Доза для красного костного мозга составляет около 120 мЗв, а для легких — около 84 мЗв. Щитовидная железа также подвергается значительной дозе (до 410 мЗв). Костная ткань получает меньшую дозу по сравнению с другими органами.

Эффективная эквивалентная доза для этой категории колеблется в пределах от 0,12 до 0,7 Зв, что значительно меньше ПДД для профессионалов (5 ПДД). Сопоставление с результатами определения доз от внешнего облучения показывает, что добавка за счет внутреннего облучения не приводит в среднем по любому из органов и для всего тела к превышению допустимого предела (5 ПДД для случая аварийной ситуации с последующей компенсацией). Внутреннее облучение можно охарактеризовать одной величиной эффективной эквивалентной дозы (ЭЭД), медианное значение которой для группы А за первый год составляет 30 мЗв, а за 50 лет — 85 мЗв (геометрическое стандартное отклонение составляет 3).

Таким образом, проведенный анализ формирования доз облучения персонала, участвующего в ликвидации последствий крупной радиационной аварии на ЧАЭС, показывает, что так же как и в случае аварийного облучения основную роль играет проникающее гамма-излучение. Однако, вклад бета-излучения и низкозергетического гамма-излучения, особенно в первые дни после аварии, может быть значительным.

Эти выводы согласуются с результатами определения доз от внешнего облучения и показывают, что добавка за счет внутреннего облучения не приводит в среднем по любому из органов и для всего тела к превышению допустимого предела (5 ПДД для случая аварийной ситуации с последующей компенсацией). Внутреннее облучение можно охарактеризовать одной величиной эффективной эквивалентной дозы (ЭЭД), медианное значение которой для группы А за первый год составляет 30 мЗв, а за 50 лет — 85 мЗв (геометрическое стандартное отклонение составляет 3).

Таким образом, проведенный анализ формирования доз облучения персонала, участвующего в ликвидации последствий крупной радиационной аварии на ЧАЭС, показывает, что так же как и в случае аварийного облучения основную роль играет проникающее гамма-излучение. Однако, вклад бета-излучения и низкозергетического гамма-излучения, особенно в первые дни после аварии, может быть значительным.
и в последующий период, является существенным. Поступление радиоактивных продуктов за счет ингаляции было ниже ПДП, в чем не последнюю роль сыграли принятые меры защиты персонала и проведение широкомасштабных дезактивационных работ.

ЛИТЕРАТУРА


OPERACIONES DE RECUPERACION
EN EL EMLAZAMIENTO
TRAS UNA PERDIDA DE AGUA PESADA
DE UN CANAL DE REFRIGERACION
DE UN SISTEMA PRESURIZADO

O.E. AGATIELLO
Central Nuclear Atucha,
Lima, Provincia de Buenos Aires,
Argentina

Abstract–Resumen

ON-SITE RECOVERY OPERATIONS AFTER A HEavy WATER LEAK FROM A COOLANT CHANNEL IN A PRESSURIZED SYSTEM.

Often one of the main problems of a major nuclear power plant accident is that no detailed plans have been made for recovery operations that can be carried out under acceptable radiological conditions immediately following an accident, nor any arrangements made guaranteeing the availability and safety of affected systems and components in the medium term. Thus the situations encountered during on-site recovery make this stage of the emergency an unknown challenge for facility management, particularly with regard to radiation protection. The aim of this paper is to present the actions which were implemented to recover the reactor building of the Atucha Nuclear Power Plant following a breakdown in normal operation due to adverse radiological conditions caused by the loss of approximately 55 tonnes of reactor-grade heavy water.

OPERACIONES DE RECUPERACION EN EL EMLAZAMIENTO TRAS UNA PERDIDA DE AGUA PESADA DE UN CANAL DE REFRIGERACION DE UN SISTEMA PRESURIZADO.

Uno de los principales problemas de un accidente importante en una central nuclear de potencia es que normalmente no se encuentran planificadas en forma detallada las actividades de recuperación de condiciones aceptables desde el punto de vista radiológico, en primera instancia, ni tampoco las de la disponibilidad y la seguridad de los sistemas y componentes involucrados, a plazo medio. Las diferentes situaciones que se encuentran durante la recuperación en el emplazamiento hacen entonces que esta etapa particular de la emergencia signifique un desafio inédito para la organización de la instalación y, en particular, de la radioprotección. El objetivo de este trabajo es presentar las acciones llevadas a cabo para recuperar el edificio del reactor de la Central Nuclear Atucha, luego de que éste quedara inaccesible a la operación normal debido a las adversas condiciones radiológicas generadas como consecuencia de la perdida de aproximadamente cincuenta y cinco toneladas de agua pesada de calidad de reactor.
1. INTRODUCCION

La Central Nuclear Atucha I posee un reactor del tipo de recipiente de presión (PHWR), cuyo combustible es dióxido de uranio natural y que utiliza agua pesada como refrigerante y moderador, con una potencia bruta de 357 MW(e).

El núcleo del reactor está conformado por 253 elementos combustibles, cada uno contenido en un canal de refrigeración.

El recambio de los elementos combustibles se lleva a cabo en operación normal.

Tanto en operación normal como en estado de ‘‘parada caliente’’, el sistema primario (refrigerante/moderador) se encuentra a una presión de 115 kg/cm$^2$, con una temperatura promedio de 300°C y 250°C respectivamente, mientras que en estado de ‘‘parada fría con presión’’, dicho sistema está a 50°C y 35 kg/cm$^2$.

2. DESCRIPCION

2.1. Suceso iniciador [1]

El 21 de diciembre de 1987 se iniciaban las pruebas preoperacionales de puesta en marcha, luego de finalizar con las tareas de reparación correspondientes a la revisión programada de dicho año.

Luego de recepcionado el sistema primario en estado de ‘‘parada fría sin presión y medio caño’’ (~50°C; 0,9 ata; nivel de refrigerante/moderador cubriendo el plenum colector superior), el Sector de Operaciones inicia las tareas de evacuado, llenado y presurizado para llevar el sistema al estado de ‘‘parada fría con presión’’ (50°C; 35 kg/m$^2$), con el objeto de efectuar el arranque de prueba de las bombas de refrigeración principal.

A las 3.58 horas del 22/12/87, cuando la presión del sistema primario era de 20 km/cm$^2$, se produce la caída brusca de la misma, como asimismo del nivel del presurizador, debido a que el cuerpo de cierre de un canal de refrigeración, que se encontraba apretado pero no enclavado, es despedido permitiendo la apertura intempestiva del mencionado canal, produciéndose entonces una pérdida de agua pesada de calidad de reactor de aproximadamente 55 t.

Como consecuencia del accidente, dicha cantidad de agua pesada resultó degradada en los sumideros; el edificio del reactor (incluidos los recintos de servicio) (véanse las Figs. 1-4) quedó inaccesible para el personal; se emitió al medio ambiente un promedio de un 50% más de actividad de tritio que en operación normal; y seis agentes de la brigada de incendio, que circunstancialmente se encontraban en el edificio del reactor debido a la aparición de una alarma de incendio, recibieron una dosis, debida a incorporación de tritio, de 16,75 mSv, 10,93 mSv, 9,78 mSv, 2,33 mSv, 2,23 mSv y 1,78 mSv, respectivamente.
FIG. 1. Vista en planta (nivel +0,5 m) del edificio del reactor de la Central Nuclear Atucha.
FIG. 2. Disposición de ventiladores para aspiración y extracción de aire contaminado del recinto del reactor.
FIG. 3. Disposición de ventiladores para aspiración y extracción de aire contaminado del recinto del reactor. Aislación del recipiente de presión y su blindaje térmico.
FIG. 4. Puntos de acumulación de agua pesada derramada con difícil acceso para recolección.
2.2. Condiciones radiológicas en la instalación.
Análisis de la recuperación [1]

La concentración de actividad de tritio generada en el edificio del reactor fue mayor que $2.5 \times 10^{-2}$ Ci/m$^3$ ($9.25 \times 10^8$ Bq/m$^3$), permaneciendo accesibles el recinto anular comprendido entre la esfera de contención y el blindaje biológico de hormigón, así como los otros edificios de la zona radiológicamente controlada (véanse las Figs. 1, 2).

En un primer momento la liberación de actividad de tritio a la atmósfera ascendió a $42,222$ Ci ($1.6 \times 10^{12}$ Bq), totalizando al final del día una emisión de $164,711$ Ci ($6.1 \times 10^{12}$ Bq), lo cual representa un 19,9% de la emisión máxima permisible diaria y un 0,5% del límite máximo permisible anual fijado en la licencia de operación de la central.

La variación de la concentración de actividad de tritio con el tiempo, a partir del momento del accidente, se muestra en la Fig. 5.

Las acciones tomadas para recuperar la situación normal de la instalación fueron:

a) Inmediatas:
— se efectuaron trasvases desde el tanque de alivio del presurizador y de almacenamiento de agua pesada para recuperar nivel en el tanque de regulación de volumen;
— se pusieron a recircular manualmente los sistemas de ventilación de los recintos del reactor y generadores de vapor;
— se enfrió el presurizador a través del rociado auxiliar controlando las temperaturas de agua y vapor;
— se lavó y decontaminó rápidamente al personal de la brigada de incendio afectado y se efectuó muestreo de orina para acotar con mayor precisión la incorporación de tritio debido al accidente.

b) Corto plazo:
— trasvase de agua pesada derramada y contenida en el piso de la máquina de recambio de elementos combustibles;
— secado de la tapa del reactor.

c) Mediano plazo:
— recolección del agua pesada de los sumideros del reactor, generadores de vapor, recintos de servicio del reactor y del espacio entre el recipiente de presión y la aislación térmica;
— trasvase de las aguas mencionadas arriba a tanques de almacenamiento;
— alimentación y operación de las columnas de regradación de agua pesada;
— control de mantenimiento del estado de componentes afectados por el derrame;
— control operacional del correcto funcionamiento de sistemas y componentes afectados por el derrame.
FIG. 5. Evolución de la concentración de actividad de tritio postaccidente durante las tareas de recuperación.
d) Largo plazo:
— regradación a calidad de reactor de la totalidad del agua pesada degradada en el accidente.

2.3. Experiencias metodológicas y técnicas de descontaminación.
   Aspectos operacionales y radiológicos

   El sistema de ventilación de la C.N. Atucha permite ser operado en modo de recirculación y en modo de barrido.

   En operación normal, los recintos de servicio del reactor y anular están en modo de recirculación + barrido, mientras que los recintos de generadores de vapor y reactor están en modo de recirculación, manteniendo las diferencias de presiones entre recintos por medio del sistema de ventilación de mantenimiento de depresión.

   Así, el caudal de barrido de los recintos de servicio del reactor y anular es del orden de 15 000 m³/h, proveyendo una renovación del volumen de aire cada 3 a 4 horas, mientras que el caudal para mantener las depresiones es de 1000 m³/h, aspirando del orden de 100 a 200 m³/h de cada recinto.

   Cuando es necesario ingresar a los recintos del reactor y generadores de vapor, a los efectos de disminuir la concentración de actividad en aire, se los coloca en modo de barrido, siendo aspirado el aire fresco desde recintos de servicio y extraído hacia la chimenea.

   Aquí, las depresiones en estos recintos deben ser mantenidas por medio de la regulación de apertura de clapetas para controlar el caudal de aire extraído.

   El agua pesada derramada quedó acumulada fundamentalmente en la calota de la aislación térmica del blindaje biológico de hormigón del recipiente de presión y entre dicha aislación y el mencionado blindaje (véanse las Figs. 3, 4).

   De esta manera se llenaron con agua pesada los conductos de ventilación para refrigeración del blindaje biológico de hormigón y las cañerías de aspiración de las bombas de refrigeración de emergencia.

   También recibieron agua pesada los sumideros de los recintos de generadores de vapor y del sistema de transporte de elementos combustibles (Fig. 2).

   El agua derramada sobre la tapa del reactor y en el piso fue recolectada en bidones, los cuales fueron trasvasados a los tanques de almacenamiento de agua pesada.

   El agua derramada en los sumideros fue enviada a los tanques de recolección de agua pesada.

   El agua que quedó ocluida en los conductos de ventilación del blindaje biológico y en la calota de la aislación térmica del recipiente de presión fue enviada por medio de bombas sumergibles auxiliares hacia los sumideros, desde donde se la envió a los tanques de recolección de agua pesada.

   La metodología empleada para la recuperación de la instalación fue la de concentrar la recolección de agua pesada derramada en los sumideros del recinto de los
generadores de vapor y del reactor para lograr aislarlos de los recintos de servicio del reactor, y así poder descontaminar éstos, para tener más fácil acceso a donde se encontraba acumulada la mayor cantidad de agua.

Luego, progresivamente se recolectó toda el agua de los recintos de generadores de vapor, teniendo así acceso, aunque con protección de todo el cuerpo, a realizar trabajos de aislación de componentes y ventilación en el recinto del reactor.

Para descontaminar los recintos de servicio del reactor, se efectuaron inyecciones periódicas de vapor a través del sistema de ventilación, manteniendo éste en modo de recirculación a través de sus condensadores y conmutándolo alternativamente al modo de barrido en función de los límites máximos permisibles de emisión por chimenea.

Para descontaminar los recintos de generadores de vapor y reactor, se instalaron ventiladores adicionales con caloventores en el nivel inferior a los efectos de evaporar el agua ocluida en la aislación térmica del recipiente de presión.

Al mismo tiempo, se colocó una extracción forzada de los niveles superiores de los recintos de generadores de vapor hacia los conductos de salida a la chimenea. Asimismo, en esta etapa inicial, se colocó una aspiración y una extracción forzada de aire sobre la tapa del reactor al nivel del piso del recinto del reactor (Fig. 2). La aspiración tomaba aire desde los recintos de servicio del reactor ya descontaminados, y la extracción era forzada hacia los conductos de salida a la chimenea.

Simultáneamente, en detrimento de la concentración del agua pesada derramada, se lavaron en sucesivas oportunidades los sumideros de estos recintos con agua liviana (Figs. 3, 4).

Además, una vez que fue posible colocar los recintos de servicio del reactor en modo de barrido en forma continua sin producir el cierre de chimenea por alta actividad, al bajar significativamente la concentración de actividad de tritio en el aire, se abrió la esclusa principal comunicando el edificio de auxiliares con los recintos de servicio del edificio del reactor.

Aquí se tuvo especial cuidado de que, a causa de fluctuaciones espúreas en el mantenimiento de las depresiones, no cambiara el sentido de circulación del aire, que necesariamente debe ser desde el edificio de auxiliares hacia el edificio del reactor.

Asimismo, se instaló un monitoreo continuo de la concentración de actividad de tritio en la puerta de la esclusa y del total de aire del edificio de auxiliares, verificándose en todo momento que no hubiera pasaje del aire contaminado.

De esta manera se logró globalmente introducir aire limpio a los recintos de generadores de vapor y reactor a través de los recintos de servicio del reactor, a la vez que éstos se descontaminaban aún más. Simultáneamente, la zona de la calota del reactor, que era el punto de mayor concentración, recibía el aire tomado desde el recinto de generadores de vapor, siendo este barrido en el nivel superior hacia los conductos de salida a la chimenea.

Así, se logró recuperar el acceso a los recintos de servicio del reactor y generadores de vapor, sin protección respiratoria permanente, al cabo de 50 días después del accidente.
En esta etapa fue necesario aislar el recipiente de presión y su blindaje térmico y biológico mediante el tendido de un folio de polietileno sobre el piso del recinto del reactor que cubría totalmente a éste (Fig. 3).

Los ventiladores que aspiraban sobre la tapa del reactor fueron colocados debajo del folio de polietileno, lográndose extraer el aire más contaminado directamente hacia los conductos de salida a la chimenea.

Mediante la aislación y evacuación del aire contaminado, se logró comenzar a efectuar la revisión de los componentes y sistemas presuntamente afectados por el derrame de agua pesada, sin utilizar equipos de protección respiratoria o de todo el cuerpo que, de haber tenido que usarlos, al entorpecer los movimientos hubieran producido una tasa de dosis recibida por radiación gamma muy alta.

Así se inspeccionó y verificó el buen estado y/o funcionamiento de los siguientes sistemas y/o componentes:

- recinto del reactor en general y lugar de impacto del cuerpo de cierre expulsado;
- canal de refrigeración afectado y su manto de enclavamiento;
- barras de control, sus conectores, bobinas de medición, fines de carrera y cableado;
- sistema de inyección de ácido deuteróbórico de emergencia;
- soportes de cañerías de compensación de los sistemas refrigerante/presurizador;
- aislación térmica entre el recipiente de presión y el blindaje biológico de hormigón;
- resistencias calefactoras del presurizador;
- instrumentación;
- accionamiento eléctrico de la válvula multivía del sistema de detección/localización de elementos combustibles defectuosos;
- accionamientos eléctricos de la máquina de recambio de elementos combustibles;
- cajas de conexiones eléctricas (intermedias, accionamientos motorizados y mediciones);
- conductos del sistema de ventilación (por posible acumulación de agua).

Más adelante, y ya para darle al personal un medio ambiente de trabajo totalmente limpio, se aislaba a éste de la atmósfera del recinto del reactor mediante el tendido de un tubo de polietileno de 2 m de diámetro, parcialmente cerrado en un extremo y conectado a la entrada de aire de barrido en el otro extremo. Esto hace que el tubo de polietileno tome forma de conducto cilíndrico, permitiendo el desplazamiento del personal por su interior a los efectos de ejecutar tareas puntuales en el recinto (Figs. 1-3).
FIG. 6. Dosis ocupacional debida al accidente.
Esta técnica permitió reducir la dosis recibida, tanto la debida a la radiación gamma como a la incorporación de tritio, a un mínimo durante el período final de tareas de recuperación en el recinto del reactor.

2.4. Aspectos dosimétricos [2]

La dosis colectiva ocupacional total debida a las tareas de recuperación fue de 1,88 Sv·hombre, la cual representa casi el 40% de la dosis ocupacional promedio anual que normalmente recibe el personal y aproximadamente un 25% de la dosis colectiva ocupacional recibida en el año 1988 (Fig. 6).

En una primera etapa, al trabajar fundamentalmente con los sistemas de recolección de agua pesada y de ventilación, las dosis debidas tanto a la irradiación externa como a la incorporación de tritio se mantienen bajas, incluso la relación porcentual de la dosis debida al tritio respecto de la dosis total.

Pero, en una segunda etapa, al tener que acceder a recintos con alta tasa de exposición usando equipos de protección de todo el cuerpo, debido a la alta concentración de actividad de tritio, la mayor lentitud en la ejecución de las tareas y movimientos en general hace aumentar más la dosis recibida por irradiación externa en relación a la dosis recibida por incorporación de tritio.

En una tercera etapa, al tornarse los recintos de servicio del reactor y generadores de vapor accesibles sin trajes de protección pero con una cierta concentración de actividad de tritio por encima del valor normal, hace que, al ingresar un mayor número de personas, se mantenga alta la dosis colectiva total y aumente al mismo tiempo el porcentaje de dosis debida a incorporación de tritio.

2.5. Aspectos de la gestión de desechos. Efluentes [2]

Durante las operaciones de recuperación se generaron residuos sólidos y líquidos, aumentando considerablemente estos últimos y los efluentes gaseosos.

Los residuos sólidos generados fueron todos de baja actividad, no superando en volumen final prensado o compactado lo habitual en operación normal.

Los residuos líquidos fueron fundamentalmente generados por la operación de las columnas de regradación de agua pesada ya que, a los efectos de recuperar en el más breve plazo posible la reserva operativa de agua pesada de calidad de reactor, se operaron éstas aumentando la concentración en mol % de D₂O del producto de cabeza desde el valor normal de 0,2 a 0,4 mol % D₂O hasta 1,0 a 1,2 mol % D₂O.

De esta manera, las emisiones de actividad de tritio al río fueron del orden de 4 a 5 veces mayores que los valores normales (Fig. 7).

La emisión de actividad de tritio por chimenea estuvo dada fundamentalmente por el necesario barrido del aire de los recintos del reactor, generadores de vapor y servicios del reactor, produciéndose en esos meses liberaciones a la atmósfera del orden de 7 veces mayores que los valores normales (Fig. 7).
FIG. 7. Emisión mensual de actividad de rtío al medio ambiente postaccidente.
De todas maneras, en ningún momento fueron superados los límites máximos permisibles mensuales ni diarios de emisión de tritio al río o a la atmósfera.
Las emisiones para el año 1988 fueron finalmente, respecto de los límites máximos permisibles anuales, de un 53,4% al río y de un 73,3% a la atmósfera.
La reserva operativa de agua pesada de calidad de reactor fue recuperada al cabo de 90 días de procesamiento después del accidente, quedando por procesar, a la fecha de presentación del presente trabajo, 120 t con una concentración de 18 mol % de D₂O (21 t de D₂O de calidad de reactor).

3. CONCLUSIONES

La operación de recuperación de la instalación luego de una gran pérdida de agua pesada tritiada implica fundamentalmente trabajar modificando u optimizando los sistemas de recolección de agua y ventilación.
Las dosis ocupacionales pueden mantenerse dentro de los valores promedio que se tienen en una reparación programada de instalaciones de este tipo.
La población de las inmediaciones de la central no resultó afectada por el accidente ni por las tareas de recuperación de la instalación.

REFERENCIAS

OFF-SITE RECOVERY OPERATIONS
(NUCLEAR FACILITIES)

(Session III)

Chairman (Part 1)

P. HEDEMANN JENSEN
Denmark

Chairman (Part 2)

B.H. WEISS
International Atomic Energy Agency

Chairman (Part 3)

A.E.J. EGGLETON
United Kingdom

Chairman (Part 4)

E. KUNZ
Czechoslovakia
SOME ASPECTS OF POST-ACCIDENT WORK IN THE CONTROL ZONE OF THE CHERNOBYL NUCLEAR POWER PLANT.

Some results of post-accident activity in the control zone of the Chernobyl nuclear power plant are presented in the paper. It considers the formation of radioactive contamination, estimates the total quantity of nuclear fuel which erupted from the damaged reactor, and discusses the creation of a database of the radioactive contamination of the surrounding area. Some newly developed methods and equipment for monitoring and surveillance of contaminated areas are discussed. The dynamics of the contamination and the decontamination work are considered briefly.

НЕКОТОРЫЕ АСПЕКТЫ ПОСЛЕАВАРИЙНЫХ РАБОТ В КОНТРОЛИРУЕМОЙ ЗОНЕ ЧЕРНОБЫЛЬСКОЙ АЭС.

Рассмотрены некоторые аспекты послеаварийных работ в контролируемой зоне Чернобыльской АЭС: формирование радиоактивных загрязнений, оценка количества выброшенного из реактора топлива и создание банка оцененных данных по радиоактивному заражению территории. Указаны некоторые новые методы и средства для измерений и контроля за уровнем радиоактивного заражения территории. Кратко затронуты вопросы динамики поведения радиоактивных загрязнений и дезактивационных работ.

1. ВВЕДЕНИЕ

Для ликвидации последствий Чернобыльской аварии (рис. 1) на площадке АЭС и окружающей территории был выполнен огромный объем работ. Непосредственно на АЭС к ним относятся меры, принятые для прерывания активной стадии аварии, дезактивация территории станции,
РИС. 1. Разрушенный реактор IV-го энергоблока Чернобыльской АЭС.
создание "Укрытия" IV-го блока, возобновление эксплуатации первых трех блоков ЧАЭС, определение местонахождения и состояния основных масс ядерного топлива в разрушенном блоке и многие, многие другие.

Значительные по объему и сложности работы выполнены и продолжают выполняться на большой территории, примыкающей к ЧАЭС. На этой территории можно выделить ряд зон отличающихся уровнем первоначального заражения и, в частности, контролируемую зону, ограниченнную изодозовой линией с уровнем гамма-фона более 3 мР/час (на 10.05.1986 г.), а также зоны, ограниченные изодозовыми линиями с уровнями гамма-фона 5 и 20 мР/час, именуемые в дальнейшем, соответственно, зоной отселения и зоной отчуждения. В рамках данного доклада будет рассказано лишь о немногих примерах восстановительных и диагностических работ, в ходе которых проявлялась одна из наиболее важных особенностей аварии — загрязнение окружающей территории мелкодиспергированным ядерным топливом.

2. ФОРМИРОВАНИЕ РАДИОАКТИВНЫХ ЗАГРЯЗНЕНИЙ.
ОЦЕНКА КОЛИЧЕСТВА ВЫБРОШЕННОГО ИЗ РЕАКТОРА ТОПЛИВА. СОЗДАНИЕ БАНКА ОЦЕНЕННЫХ ДАННЫХ ПО РАДИОАКТИВНОМУ ЗАРАЖЕНИЮ ТЕРРИТОРИИ

К 26.04.86 г. IV-ый блок ЧАЭС проработал 735 эффективных суток (865 календарных суток). Большую часть (около 75%) из 192 т ядерного топлива, находившегося в реакторе, составляли кассеты первоначальной загрузки с выгоранием от 11 до 17 МВт-сут/кг(урана). С учетом "свежих" кассет среднее выгорание составляло 11 МВт-сут/кг(урана). Количество находившихся в активной зоне реактора биологически значимых долгоживущих радионуклидов указано в табл. I.

Выбросы чрезвычайно высокой активности (сотни тысяч кюри в час) происходили с переменной интенсивностью в течение всей активной стадии аварии, продолжавшейся приблизительно 10 суток с 26.04 по 06.05.86 г. При этом радиоактивная струя поднималась над развалом реактора более чем на 1200 м в первые два дня, затем высота ее подъема уменьшилась до 200-400 м. На рис.2 представлен график интенсивности выброса радиоактивных веществ из разрушенного реактора в атмосферу. Сразу же после взрыва возник вопрос о том, сколько радиоактивных веществ и в какой форме было выброшено из реактора.

Измерения происходили в необычных условиях, поскольку им препятствовали мощные радиационные поля. С большими трудностями удалось брать необходимые пробы и измерять параметры потока горячего воздуха, выносящего радионуклиды из разрушенного реактора в атмосферу. Работа по оценкам и расчетам выбросов проводилась непрерывно,
ТАБЛИЦА 1. СОДЕРЖАНИЕ ОСНОВНЫХ ДОЛГОЖИВУЩИХ БИОЛОГИЧЕСКИ ЗНАЧИМЫХ РАДИОНУКЛИДОВ В АКТИВНОЙ ЗОНЕ РЕАКТОРА IV-ГО БЛОКА ЧАЭС НА МОМЕНТ АВАРИИ

<table>
<thead>
<tr>
<th>Изотоп</th>
<th>Масса, кг</th>
<th>Активность, МКи</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-238</td>
<td>1,5 кг</td>
<td>-</td>
</tr>
<tr>
<td>Pu-239</td>
<td>420 кг</td>
<td>-</td>
</tr>
<tr>
<td>Pu-240</td>
<td>175 кг</td>
<td>-</td>
</tr>
<tr>
<td>Pu-241</td>
<td>50 кг</td>
<td>-</td>
</tr>
<tr>
<td>Pu-242</td>
<td>14 кг</td>
<td>-</td>
</tr>
<tr>
<td>Cs-137</td>
<td>81 кг</td>
<td>(7,2 МКи)</td>
</tr>
<tr>
<td>Cs-134</td>
<td>3 кг</td>
<td>(4,0 МКи)</td>
</tr>
<tr>
<td>Sr-90</td>
<td>4,3 кг</td>
<td>(6,0 МКи)</td>
</tr>
</tbody>
</table>

РИС.2. Интенсивность радиоактивного выброса в течение активной стадии аварии.
Однако модели были весьма приблизительными из-за сложного и недостаточно изученного характера процессов, формирующих выброс, постоянно меняющейся погоды и влияния активных воздействий на реактор, проведенных в начальный период аварии. Все это привело к большим ошибкам при определении мощности выброса (рис. 2).

В результате аварии образовались три основных "языка", по которым течение воздушных масс распространило в период с 26 апреля по 6 мая радиоактивные вещества: северный, западный и южный. Они изобра-
жены на рис. 3. Цифрами 1–8 указано число суток между моментом ава-рии и временем прибытия загрязненных воздушных масс в данную точку. Следует отметить, что "северный язык" сформировался главным образом во время первого радиоактивного выброса.

Перейдем теперь к оценкам количества и состава выброшенных радионуклидов.

Задача обнаружения и количественного определения радионуклидов, выброшенных на территорию площадью порядка тысяч квадратных километров с сотнями населенных пунктов, чрезвычайно трудоемка, и в общем случае трудно представить путь ее решения в короткие, порядка дней и месяцев сроки. Рассмотрим, в частности, изотопы плутония и стронция, которые вместе с Cs–137 определяют по существу заражение местности по прошествии нескольких лет после аварии. Pu–238, Pu–239, Pu–240 — практически чистые α–излучатели, Sr–90 — жесткий β–излучатель и их количественное определение связано с взятием почвенных проб и проведением сложных и длительных радиохимических анализов. Достаточно сказать, что такого рода полных анализов за первые полтора года удалось провести всего несколько сотен. Эти данные никак не могли решить вопроса о создании карт загрязнения местности плутонием и стронцием. На помощь пришла специфика Чернобыльской аварии. На всех стадиях аварии выброс радионуклидов, за исключением инертных газов и ряда легколетучих типа иода, цезия, теллура, происходил в составе частиц мелкодисперсированного топлива. Это показали уже первые результаты анализов воздушных фильтров, взятых над развалом реактора.

ТАБЛИЦА II. РЕЗУЛЬТАТЫ СРАВНЕНИЯ АНАЛИЗА ФИЛЬТРА (27.04.86 г.) С РАСЧЕТНЫМ СОДЕРЖАНИЕМ РАДИОНУКЛИДОВ

<table>
<thead>
<tr>
<th>Радионуклид</th>
<th>Отношение экспериментально измеренной активности фильтра и расчетной активности топлива, (отн. ед.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nb–95</td>
<td>1,06</td>
</tr>
<tr>
<td>Zr–95</td>
<td>1,07</td>
</tr>
<tr>
<td>Ru–103</td>
<td>0,92</td>
</tr>
<tr>
<td>Ru–106</td>
<td>1,20</td>
</tr>
<tr>
<td>Cs–134</td>
<td>3,11</td>
</tr>
<tr>
<td>Cs–137</td>
<td>3,20</td>
</tr>
<tr>
<td>Ce–141</td>
<td>0,85</td>
</tr>
<tr>
<td>Ce–144</td>
<td>1,05</td>
</tr>
</tbody>
</table>
В табл. II. показаны результаты сравнения одного из таких анализов (фильтр взят 27 апреля, приблизительно через сутки после начала аварии) с расчетным содержанием соответствующих радионуклидов в ядерном топливе IV-го блока. Данные приведены для основных гамма-излучающих радионуклидов с периодом полураспада более 1 года.

Из таблицы хорошо виден "топливный" характер выброса для слабо-летучих радионуклидов и значительное обогащение изотопами цезия. Устойчивость соотношений между количеством различных радионуклидов в выбросах и их совпадение со средним соотношением радионуклидов в топливе IV-го блока подсказали простое решение вопроса о количественном определении заражения местности различными биологически значимыми радионуклидами и общим количеством топлива. Кропотливые радиохимические анализы могли быть успешно заменены существенно более простыми и оперативными измерениями гамма-активности, с последующим определением например активности плутония по соотношению:

\[ A(\Sigma Pu) = K_k \cdot A_{\gamma_i}, \]

где \( A(\Sigma Pu) \) — суммарная альфа-активность Pu-238, Pu-239, Pu-240; \( A_{\gamma_i} \) —
активность какого-либо гамма-активного радионуклида; $K_k$ — коэффициент корреляции. Гамма-активный "опорный" радионуклид было целесообразно выбирать долгоживущим и достаточно полно связанным с топливной матрицей из двуокиси урана. Этим требованиям удовлетворял $\text{Ce}-144$, с периодом полураспада равным (284 дня). Величина коэффициента корреляции для активности суммы изотопов плутония и активностью $\text{Ce}-144$ составляла на 26.04.86 г. около $9 \times 10^{-4}$ (расчетное значение).

Поскольку $K_k$ относительно слабо зависел от величины выгорания топлива, а выгорание большинства ТВС мало отличалось от среднего, коэффициент корреляции для всего топлива реактора в первом приближении можно было считать постоянным (рис. 4).

К середине июля 1986 г. институтами Министерства среднего машиностроения, Госкомгидромета, Министерства обороны были независимо выполнены измерения и расчеты, показавшие, что за пределы IV-го блока выброшено от 2 до 6% первоначальной загрузки, что составляет от 4 до 12 т собственно топлива.

К этому времени уже сложилась трехзвенная схема определения топливных загрязнений территорий:

— во первых, измерение мощности дозы гамма-полей над территориями с помощью аэрогаммаразведки (первое "грубое" приближение, использующее соотношение "мощность дозы—количество топлива");
— во-вторых, оперативное исследование почвенных проб на полупроводниковых гамма-спектрометрах (уточняющие измерения с использованием коэффициента корреляции по цернию-144);
— и, наконец, медленные и тщательные радиохимические анализы (проверка $K_k$ для данной местности).

Информация о величине выброса радиоактивных веществ из реактора IV-го блока была доведена до сведения мировой общественности в докладе, подготовленном ГКАЭ СССР на совещании экспертов МАГАТЭ в Вене в августе 1986 года [1].

Основные выводы доклада:

— радиоактивные инертные газы практически полностью выброшены из реактора;
— выброшено значительное количество иода;
— выброшено (13 ± 7%) цезия;
— выброшено 3% топлива (с ошибкой ± 1,5%), содержащего продукты деления и трансурановые элементы. Именно эта часть выброса определяет долговременное радиоактивное заражение и дозовые нагрузки в зоне отчуждения.
КОЛИЧЕСТВО ВЫБРОШЕННОГО ТОПЛИВА (% ОТ ПОЛНОЙ ЗАГРУЗКИ)

ТЕРРИТОРИЯ ВЫБРОСА:
— Промплощадка ЧАЭС ............ ≤ 0.3
— Прилегающая 80-км зона ...... ≤ 1,5
— Остальная территория СССР ≤ 1,5
— За пределами СССР............... ≤ 1
— Здание IV-го блока и “крыши” ≥ 96

РИС. 5. Распределение выпавшего из выбросов плутония.

По предварительным оценкам ИАЭ им. И.В.Курчатова выброшенное топливо распределилось по территориям так, как показано на рис. 5.

В течение августа-сентября 1986 г. была проведена программа тепловых измерений на поверхности раз渲а и на периферии разрушенного реактора. Сравнивая выделяющееся тепло с расчетами, удалось установить, что внутри “Укрытия” находится не менее 87% топлива (оценку удалось сделать только снизу).
Многие различные организации занимались сбором данных о характере и уровнях загрязнения почв, воды и воздуха в контролируемой зоне, особенно в первые после аварии месяцы. Были собраны тысячи проб, однако первоначальная несогласованность подходов и методик отбора, обработки и описания проб, отсутствие специальных информационных систем для хранения и, главное, оценки данных не позволили эффективно использовать всю информацию и правильно оценивать уровень ее достоверности. Именно в это время (май-июнь 1986 г.) в ИАЭ им. И.В. Курчатова была заложена основа Автоматизированной информационной системы (АИС) "ПРОБА" (аббревиатура полного названия "Показатели радиационной обстановки после аварии") для хранения и оценки информации о поверхностном радиоактивном загрязнении почв топливыми выпадениями после аварии. Как уже говорилось, уже к лету 1986 г. были сделаны выводы о пятнистом характере выпадений отдельных радионуклидов — обнаружены, например, так называемые "цезиевые пятна". К осени того же года АИС давала первые оценки масштабов и границ загрязненных площадей и первые оцененные карты поверхностных загрязнений изотопами плутония, по которым были сделаны достаточно представительные оценки топливного выброса. В аппарате АИС был наложен сбор разнообразной информации:
- данных спектро- и дозиметрических измерений;
- результатов радиохимических анализов проб;
- данных о результатах анализов проб ядерно-физическими методами.

Проводилась постоянная экспертиза представительности серий или отдельных проб по критериям:
- точности применяемой аппаратуры;
- соотношениям между активностями исследуемых α-, β-, γ-излучателей;
- статистической достоверности;
- совместимости (временной, географической) с другими аналогичными измерениями.

Чтобы не возвращаться более к вопросу о выбросе топлива, скажем еще, что прошедшие годы оценка его количества продолжала уточняться. Так в 1988 году на основе данных АИС "ПРОБА", содержащей полные сведения о десятках тысяч проб почвы, взятых на территории СССР, и данных зарубежных исследований был проведен расчет топлива, выброшенного при аварии. Полученное значение составляет 3,5% с ошибкой ± 0,5%.
ТАБЛИЦА III. НОВЫЕ И МОДИФИЦИРОВАННЫЕ МЕТОДЫ ИЗУЧЕНИЯ РАДИОАКТИВНЫХ ВЫПАДЕНИЙ ВЫБРОСОВ ПОСЛЕ АВАРИИ НА IV БЛОКЕ ЧАЭС

<table>
<thead>
<tr>
<th>Название метода и назначение</th>
<th>Зоны применимости метода</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td><strong>Укрытие</strong></td>
</tr>
<tr>
<td>&quot;УФ&quot;</td>
<td>x x x x x</td>
</tr>
<tr>
<td>Обнаружение и локализация сильных источников по свечению воздуха в ультрафиолетовой области</td>
<td>x x x</td>
</tr>
<tr>
<td>&quot;Коллимированный γ-дозиметр.&quot;</td>
<td>x x x x x</td>
</tr>
<tr>
<td>Обнаружение и локализация источников</td>
<td>x x x x x</td>
</tr>
<tr>
<td>&quot;Полевая полупроводниковая спектрометрия.&quot; Детальное исследование радионуклидного состава почвенных загрязнений</td>
<td>x x x x x</td>
</tr>
<tr>
<td>&quot;Быстрая радиохимия и сцинтилляционный анализ.&quot; Определение в пробых плутония и стронция</td>
<td>x x x x x</td>
</tr>
<tr>
<td>Методы изучения и выделения &quot;горячих&quot; частиц</td>
<td>x x x x x</td>
</tr>
</tbody>
</table>

Условное обозначение: x x x — зона применимости метода.

3. МЕТОДЫ И СРЕДСТВА, ПРИМЕНЯВШИЕСЯ ДЛЯ ИЗМЕРЕНИЯ И КОНТРОЛЯ ЗА СТЕПЕНЬЮ РАДИОАКТИВНОЙ ЗАРАЖЕННОСТИ МЕСТНОСТИ

В ходе работ по ликвидации последствий аварии были использованы не только хорошо известные методы и средства контроля и диагностики, но и целый ряд усовершенствованных или совсем новых методов. Примеры таких методов приведены в табл. III.
Как уже указывалось, прочная связь плутония с топливной матрицей, образующей радиоактивные частицы выброса, позволила определять его содержание в пробах, используя коэффициенты корреляции с Се-144.

Но шло время и от этого простого метода пришлось бы отказаться, как из-за распада Се-144 (T_{0,5} = 284 дня), так и из-за постепенного разрушения частиц и нарушения корреляции в радиоизотопном составе. Необходимо было найти достаточно быстрый метод прямого измерения плутония, но здесь преимущество его нахождения в составе топливных частиц перешло в недостаток. Для получения статистически достоверного результата требовались пробы, содержащие достаточно большое количество частиц, т. е. достаточно большое количество почвы. В этом случае непосредственные α-спектрометрические измерения не давали нужных результатов, т. к. достигнуть необходимой степени гомогенизации вещества пробы в весьма малых объемах, необходимых для изготовления мишений, оказалось невозможным. Решение было найдено на пути объединения двух методов — быстрой радиохимии и сцинтилляционной техники.

| Взятие образцов для анализа (проба почвы, воздушный фильтр, и т. д.) |
| Озоление (сжигание) |
| Гомогенизация и изготовление представительной навески (около 20 г.) |
| Извлечение плутония из навески раствором KBr в HNO₃ |
| Экстрагирование органическим растворителем |
| Введение в жидкий сцинтиллятор |
| Регистрация α-частиц на установке ПСД (с разделением импульсов по форме) |
| Расчет количества Pu в исследуемом образце |

РИС.6. Структурная схема одного из методов анализа на плутоний.
Упрощенная схема радиохимических операций показана на рис. 6. Если традиционная радиохимия требовала для проведения анализов с полным вскрытием пробы, извлечением плутония и изготовлением мишени нескольких дней квалифицированной работы, то описываемый метод занимает около 8 часов. Средняя величина выщелачивания плутония (95–5%), его переход в экстрагент близок к 100%. Таким образом практически весь плутоний переходил в раствор экстрагента. Затем этот раствор вливался в кювету с жидким сцинтиллятором. Регистрация велась на аппаратуру, позволяющей выделять импульсы характерной для $\alpha$-частиц формы.

Выбранный метод обладает значительными преимуществами:
— извлечение плутония из большого объема вещества;
— большой эффективностью (объемная регистрация в жидкой среде);
— малым фоном (за счет селективности регистрации импульсов), что особенно важно при работе в условиях повышенного радиационного фона в Чернобыле.

Сцинтилляционная установка ПСД, используемая в этих измерениях, показана на рис. 7.
4. ДИНАМИКА ПОВЕДЕНИЯ РАДИОАКТИВНЫХ ЗАГРЯЗНЕНИЙ

В течение прошедших лет (1986–1988 гг.) физико-химические свойства топливных частиц определяли динамику изменений радиоактивных загрязнений в ближайшей зоне. Эта динамика определялась радиоактивным распадом, гидро- и аэропереносом, диффузией в почву.

Уже первые исследования, проведенные в 1986 г., указывали на весьма малую вымываемость радиоактивности из почвенных проб. В опытах имитировались атмосферные осадки или паводковые воды. В качестве образцов чаще всего использовались песчаные почвы, поскольку эксперименты установили, что торф, чернозем, суглинки и глины удерживают радиоактивные загрязнения значительно лучше. На рис. 8 приведены результаты одного из экспериментов (1988 г.), имитирующих атмосферные осадки. В данном опыте количество воды составляло около 2,5 л. Из графика видно, что доля вымываемой при этом радиоактивности не превышает 0,5% от полной. В табл. IV приведены значения относительной активности различных фракций в одной из почвенных проб, залитой водой и пробывшей в таком состоянии 10 дней (имитация паводка). Из пробы были извлечены плавающие частицы, растительность, затем профильтрован слой воды над почвой.
Устойчивость топливных частиц к вымыванию радионуклидов привела к слабому гидропереносу радиоактивности. В этом смысле показатель паводок 1987 г., который ожидался в контролируемой зоне большими опасениями. Однако, (в полном согласии с результатами лабораторных опытов) увеличения концентрации радионуклидов в реках и водоемах не было. Во время весенних паводков 1988 и 1989 гг. концентрация радионуклидов в воде была еще ниже. Например, содержание цезия в Киевском водохранилище, р. Припять, р. Днепр не превышало сотых долей от допустимой концентрации. Слабая вымываемость радионуклидов из почвы сохранилась до настоящего времени. Содержание радионуклидов в воздухе в целом было незначительно, например, концентрация плутония в зоне отчуждения (см. Введение), даже при скорости ветра 10 м/сек была ниже предельно допустимой регламентированной концентрации.

В сотнях экспериментов изучалось глубинное распределение продуктов деления и трансурановых элементов в ближней зоне. Практически для всех исследованных районов и типов почв верхний слой в 3–5 см удерживает более 90% активности радионуклидов.

Так в опытах 1988 г. были получены распределения активности плутония и церия-144 в зависимости от глубины залегания в почве (плутоний определялся по экспресс-методике, описанной в п. 2). В обоих случаях зависимости аппроксимируются экспоненциальной функцией:

\[ A_{Pu}(x) = \exp(-0,88x) , \]

где \( A_{Pu}(x) \) — суммарная активность по плутонию, соответствующая зависимость для церия-144:

\[ A_{Ce}(x) = \exp(-0,87x) , \]

где \( x \) — глубина залегания в см.
Близкие данные получены в аналогичных экспериментах, проведенных весной и летом 1989 г. Такой результат подтверждает вывод о высокой устойчивости топливных частиц.

5. ДЕЗАКТИВАЦИОННЫЕ РАБОТЫ

В предыдущих разделах рассматривались диагностические аспекты послеаварийных работ в контролируемой зоне, далее будут рассмотрены некоторые вопросы, связанные с проведением дезактивационных работ. Специфика радиоактивных загрязнений и многообразие объектов, подлежащих дезактивации, обусловили использование большого количества различных технологий, а работы по дезактивации оказались червячайно трудоемкими и длительными. В ряде случаев, например при дезактивации объектов жилой застройки и дорог, использованные методы были, в целом, недостаточно эффективны. Дезактивационные работы начались с 27 апреля 1986 года (на следующий день после начала аварии) и проводились методом смысла водой радиоактивных загрязнений с покрытия улиц, площадей и скверов в городах, дворах и строениях в населенных пунктах, с транспортных магистралей в районах, непосредственно прилегающих к АЭС и образовавших в дальнейшем зону отчуждения. В этой зоне из-за высоких уровней радиации и малой эффективности дезактивация была прекращена (за исключением г. Припять). Население из зоны было полностью эвакуировано, доступ в нее строго регламентирован. В зоне временного отселения и в контролируемой зоне работы проводились в широких масштабах. Основными объектами были местность и лесные массивы, дороги, населенные пункты, технические и транспортные средства.

Для дезактивации местности, в зависимости от ландшафта и конкретных особенностей радиоактивных выпадений, использовались:

— срезание и перекопка грунта;
— грейдирование;
— отсыпка чистым грунтом;
— обваловка и засыпка чистым грунтом;
— пленочное полимерное покрытие.

Загрязненные лесные массивы играли особую роль, поскольку леса являлись наиболее мощным источником вторичного радиоактивного загрязнения при ветровом переносе. Общая площадь зараженных лесов в УССР более $700 \times 10^3$ га, в 30-ти км зоне 106,1 $\times 10^3$ га, из них погибло 450 га. Наибольшие уровни загрязнений в лесах образовывались на опушках со стороны движения радиоактивного выброса в атмосфере, что при водило к ленточному поражению или гибели деревьев на границах лесных.
РИС. 9. Лесной массив, получивший значительное радиоактивное загрязнение (так называемый "рыжий лес").
массивов. Из всех компонентов лесного ценоза наиболее загрязнялась лесная подстилка, в ней сосредотачивалось от 30% до 90% радионуклидов, выпавших на лесные массивы. Как и в большинстве других случаев, радионуклиды лесной подстилки и верхних слоев почвы были устойчивы к вымыванию. Дезактивация лесов проводилась путем закрепления (фиксацией) радионуклидов в лесных массивах и захоронению наиболее загрязненных участков. Для примера приведем работы по захоронению лесного массива площадью около 400 га, примыкающего к ЧАЭС с запада. Этот массив получил значительное радиоактивное загрязнение и из-за большого числа погибших деревьев приобрел характерную рыжую окраску. Так называемый "рыжий лес" (рис. 9) в 1987 г. был окружен насыпным валом высотой 2,0-2,5 м, деревья и мелколесье были повалены и засыпаны слоем земли до 1 м. Сверху был насыпан слой гумуса толщиной до 20 см и посеяны многолетние травы. Объем захороненного леса составил около 4000 м³, средний уровень гамма-излучения уменьшился в 35-40 раз, а в наиболее зараженных местах на три порядка. По окончании этих работ (вторая половина 1987 г.) наибольший уровень радиации на поверхности составил 180 мР/ч.

ЛИТЕРАТУРА

USE OF DOSE ASSESSMENT MODELS TO FACILITATE OFF-SITE RECOVERY OPERATIONS FOR ACCIDENTS AT NUCLEAR FACILITIES

M.H. DICKERSON, K.T. FOSTER
Lawrence Livermore National Laboratory,
Livermore, California,
United States of America

Abstract

USE OF DOSE ASSESSMENT MODELS TO FACILITATE OFF-SITE RECOVERY OPERATIONS FOR ACCIDENTS AT NUCLEAR FACILITIES.

One of the most important uses of dose assessment models in response to accidents at nuclear facilities is to help provide guidance to emergency response managers for identifying and mitigating the consequences of an accident once the accident has been terminated. By combining results from assessment models with radiological measurements, a qualitative methodology can be developed to aid emergency response managers in determining the total dose received by the population and in minimizing future doses through the use of mitigation procedures. To illustrate the methodology, this paper focuses on the use of models to estimate the dose delivered to the public both during and after a nuclear accident. A three-dimensional, mass conservative, diagnostic wind field model coupled to a particle-in-cell diffusion code is used to simulate the radiological effects from a hypothetical nuclear power plant accident. Although the power plant used in this simulation is hypothetical, the meteorological and topographical input to the models was taken from real-world measurements for a particular location in the northwestern USA. Because these models are three dimensional, however, they can be generalized for application to different sites without tailoring for each unique location. Although these models can be used for operational real time calculations, this discussion will emphasize their use for post-accident assessment. Results of the calculations for this paper are displayed in graphical form to show the area affected by the accident as it pertains to various doses and potential doses received by the public within about 100 kilometres of the plant site. Radiological effects from three selected nuclides (\textsuperscript{137}I, \textsuperscript{137}Cs and \textsuperscript{133}Xe) are modelled for demonstration purposes. The information depicted in these calculations can be used to help determine any additional mitigation measures that need to be implemented. Examples of the use of these calculations by the emergency response manager to better manage available resources are also described in the paper.
1. INTRODUCTION

One of the most important questions asked after a nuclear accident is, 'what dose has been delivered to the public?'. In most cases it is important to identify dose by the different pathways and also by total dose received during the accident. In addition, officials want to know the dose delivered to the public living in a contaminated area so that decisions can be made regarding relocation and/or cleanup of the contamination.

To estimate the doses received by the public during an accident and the doses they will receive living in a contaminated area after the accident, both model calculations and environmental measurements are available to the emergency response manager. Ideally, these two resources should be combined quantitatively to provide these estimates; however, the technology to accomplish this task is in its infancy and is not yet available in an operational setting. Therefore, for the purposes of this paper, the discussion will be focused on the use of models to estimate the dose delivered to the public both during and after a nuclear accident. Results from these calculations can be a valuable aid in the recovery operations. In order to keep the length of material in this paper reasonable, only three radionuclides (\textsuperscript{131}I, \textsuperscript{133}Xs and \textsuperscript{137}Cs) will be used in the dose calculations. Obviously, for an accident, all radionuclides that contribute to the dose should be included.

For the dose calculations shown in this paper the MATHEW/ADPIC models [1, 2], that are part of the US Department of Energy's Atmospheric Release Advisory Capability (ARAC) [3], have been used. These models are three dimensional and include terrain explicitly. They are diagnostic, and therefore use measured wind speeds and directions, interpolated to a grid volume and adjusted to satisfy the mass continuity equation. Although the location and meteorological and topographical data are real, the reactor used in this study is not. In the last section we show a final estimate of the dose due to the \textsuperscript{133}Xe that was released during the Three Mile Island (TMI-2) accident in 1979. Calculations shown in this example were used by the US President's Commission to help estimate the final dose to the public living around the reactor site. Finally, the intent of this paper is to show a methodology and not explicitly to describe effects or suggest protective actions.

2. THE PROBLEM

2.1. Description of the terrain

Figure 1 shows a 200 \times 200 km computer generated view of the terrain features around the hypothetical nuclear power plant site. The facility is assumed to be located in the centre of the figure in an area that is relatively flat. To the southeast and the southwest of the facility there are valleys and ridges with terrain variations
up to 1500 metres. Towards the northwest the terrain is relatively smooth with a valley oriented east-west along the southern edge of the quadrant. The northeast quarter varies from relatively smooth close to the facility to ridges and valleys away from the facility. In addition to influencing the wind flow patterns, these terrain features can and do influence the deposition patterns due to partial impacting of the airborne material as it moves over valleys and approaches ridges on the downwind side. This terrain database was generated from a real database and represents an area in the northwestern part of the USA.
2.2. Description of the geography

Figure 2 depicts the geographical features over an area 160 × 160 km around the hypothetical nuclear power plant. This base map will be used for overlaying the contour patterns. Towns A and B represent the most populated areas of the region with populations of about 20 000 each. Much of the region is forested, with most of the agricultural land lying to the northwest of the plant. The major national roads in the area are I-12 and I-7, with other roads such as S-222, S-216 and S-219 representing local road systems. Owing to the rugged terrain, areas in the southeast, southwest and northeast do not have national or local roads; however, these areas would likely have farm roads and trails. The national roads are heavily travelled since they carry much of the truck and car traffic moving between states.

2.3. Accident sequence

The methodology may best be illustrated by application to a simulated accident at a hypothetical nuclear power plant. The plant is a 1000 MW(e) pressurized water
TABLE I. SOURCE TERMS

<table>
<thead>
<tr>
<th></th>
<th>Iodine-131</th>
<th>Caesium-137</th>
<th>Xenon-133</th>
</tr>
</thead>
<tbody>
<tr>
<td>Amount released in 6 h (Bq)</td>
<td>$1.8 \times 10^{17}$</td>
<td>$3.7 \times 10^{14}$</td>
<td>$6.1 \times 10^{18}$</td>
</tr>
<tr>
<td>Source rate (Bq/s)</td>
<td>$8.3 \times 10^{12}$</td>
<td>$1.7 \times 10^{10}$</td>
<td>$2.8 \times 10^{14}$</td>
</tr>
</tbody>
</table>

reactor that has been in operation for several years. The postulated accident under consideration is a small loss of coolant accident that leads to a failure of the emergency cooling injection system. This loss of coolant causes the core to be partly uncovered; a condition that subsequently leads to partial fuel melting. Radioactivity escapes from the core into the containment atmosphere and is partially removed by various mitigation systems, such as water sprays and filters, prior to release into the environment. On the basis of the inventory of radionuclides in the core, the expected fractional releases of various radionuclide transport groups for this particular accident sequence, the reduction due to the mitigation systems, combined with measurements (hypothetical) monitored by an onsite computer, derived source term estimates for this example are given in Table I.

2.4. Meteorological conditions

Real meteorological conditions were used for these calculations. The data started at 1700Z, 22 February 1989 and ended 32 hours later when the final calculations were complete. At 1700Z, when the release starts, synoptic conditions show the accident area approximately equidistant from a surface high to the southeast and a surface low to the northwest. The cold front associated with the low is about 700 kilometres to the west. North-northeasterly flow is dominant in the release point area with a speed of 3 m/s at 020°. During the time period 2000 – 2200Z, surface stations near the accident site report a shift from the northeasterly winds to more southeasterly (the surface wind field thus has a more northwesterly component). The release ends at 2300Z. Reports available from the site at that time show light (2 m/s) southerly winds. Surface winds remain easterly to the north of the release area. The station 120 kilometres to the northwest of the release area reports, at 0000Z, 23 February 1989, a sudden wind shift to west-southwesterly at 5 m/s. The station 80 kilometres to the northwest of the release area reports a similar wind shift, giving uniform westerly flow to the north of the accident point. Surface winds at the accident point shift to become southwesterly (240 at 5 m/s) as the surface trough passes
over the area with the weakening cold front approximately 150 kilometres to the west. Conditions remain fairly constant through the night (0200Z - 1200Z). During the period 1200Z-0000Z the surface high drifts slowly east-southeast, setting up a fairly uniform easterly surface flow over the entire area.

3. RESULTS

Figure 3 shows the $^{131}$I thyroid dose committed for 50 years via the inhalation pathway. Within the largest contour value, which includes about 98 km$^2$, the thyroid dose to an unsheltered adult is greater than 1 Sv. In an area of about 672 km$^2$ adults received a thyroid dose of between 0.1 and 1 Sv under the same conditions of no credit taken for shelter or evacuation.

Total $^{131}$I deposition is shown in Fig. 4. The potential dose to an adult drinking milk from cows grazing within the $1 \times 10^7$ Bq/m$^2$ contour area (about 207 km$^2$) is approximately 15 Sv. Again, this dose estimate assumes no mitigation and is based on available $^{131}$I on the ground immediately after the accident. With an eight day half-life this estimate will be reduced within a relatively short period of time.

Figures 5 and 6 show the inhalation pathway effective whole body dose and deposition for $^{137}$Cs. The inhalation dose is about two orders of magnitude less than it is for $^{131}$I (0.03 $\times$ $^{131}$I adult thyroid dose equals the inhalation effective whole body dose). The dose delivered through the cow–milk pathway due to deposition of $^{137}$Cs is approximately 15 mSv, about three orders of magnitude less than it is for $^{131}$I. However, since the half-life of $^{137}$Cs is over 30 years, its deposition near the reactor site could pose a problem regarding land use. Figure 7 shows the $^{133}$Xe effective whole body dose for the air immersion pathway. This dose is an order of magnitude less than the effective whole body dose received through $^{131}$I.

Figures 8 and 9 show combined effective whole body dose and dose rate for $^{131}$I, $^{133}$Xe and $^{137}$Cs inhalation and $^{131}$I and $^{137}$Cs deposition. The effective whole body dose (Fig. 8) is largely given by $^{131}$I, with minor contributions from $^{133}$Xe and $^{137}$Cs. Within the 10 mSv contour, the public over an area of about 300 km$^2$ received doses greater than 10 mSv, assuming no protective actions were taken. For dose from deposition (groundshine), Fig. 9 shows a dose rate over $10^{-5}$ mSv/s from $^{131}$I and $^{137}$Cs for an area of approximately 118 km$^2$. This dose rate translates to approximately 1 mSv/d immediately after the accident and then decays due to the eight day half-life of $^{131}$I.

4. TMI-2 ACCIDENT ASSESSMENT

Although the TMI-2 accident in 1979 released significant amounts only of $^{133}$Xe and therefore recovery operations for the environment around the reactor site
Remarks:
Inhalation pathway
50 year committed adult thyroid dose

Integrated:
Feb. 22, 89 1700 Z to Feb. 24, 89 0100 Z

Material:
I-131 at 1.5 m

Contour values (in mSv)

\( > 10^3 \)
Area covers 98 sq. km.

\( > 10^2 \)
Area covers 672 sq. km.

\( > 10^1 \)
Area covers 4927 sq. km.

FIG. 3. Fifty year committed adult thyroid dose from the \(^{131}I\) inhalation pathway (mSv).

Remarks:
Cumulative deposition

Integrated:
Feb. 22, 89 1700 Z to Feb. 24, 89 0100 Z

Material:
I-131 at 0.0 m

Contour values (in Bq/m\(^2\))

\( > 10^7 \)
Area covers 207 sq. km.

\( > 10^5 \)
Area covers 3677 sq. km.

\( > 10^3 \)
Area covers 4079 sq. km.

FIG. 4. Iodine-131 deposition pattern (Bq/m\(^2\)).
Remarks:
Inhalation pathway
50 year committed
effective whole body dose

Integrated:
Feb. 22, 89 1700 Z
to Feb. 24, 89 0100 Z

Material:
CS-137 at 1.5 m

Contour values
(in mSv)

\[ \begin{align*}
1 & > 10^1 \\
1 & > 10^3 \\
1 & > 10^5
\end{align*} \]

Area covers 38 sq. km.
Area covers 4037 sq. km.
Area covers 7743 sq. km.

FIG. 5. Effective whole body dose from the \(^{137}\)Cs inhalation pathway (mSv).

Remarks:
Cumulative deposition

Integrated:
Feb. 22, 89 1700 Z
to Feb. 24, 89 0100 Z

Material:
CS-137 at 0.0 m

Contour values
(in Bq/m\(^2\))

\[ \begin{align*}
1 & > 10^5 \\
1 & > 10^4 \\
1 & > 10^3
\end{align*} \]

Area covers 60 sq. km.
Area covers 361 sq. km.
Area covers 2729 sq. km.

FIG. 6. Caesium-137 deposition pattern (Bq/m\(^2\)).
FIG. 7. Effective whole body dose from the $^{133}$Xe air immersion pathway (mSv).

FIG. 8. Effective whole body dose: total dose from $^{131}$I, $^{133}$Xe and $^{137}$Cs (mSv).
Remarks:
Combined I-131 & CS-137
effective whole body
ground exposure dose rate

Integrated:
Feb. 22, 89 1700 Z
to Feb. 24, 89 0100 Z

Material:
I-131, and CS-137
at 0.0 m

Contour values
(in mSv/s)
> 10^-5
Area covers 118 sq. km.
> 10^-6
Area covers 803 sq. km.
> 10^-7
Area covers 2952 sq. km.

FIG. 9. Effective whole body ground exposure dose rate due to deposition of 131I and 137Cs (mSv).

were not necessary, model calculations were used to help estimate individual doses to the public and population doses to those persons living within about a 50 km radius of the reactor site. Figure 10 shows the integrated individual inhalation — immersion dose for a ten day integration period delivered by the 133Xe escaping from the TMI-2 reactor buildings. This dose estimate was used by the US President’s Commission on Three Mile Island to estimate the dose received by the population. This ten day integration used hourly measured windspeeds and directions from five locations. The maximum individual dose received from this calculation was 0.12 mSv. From this calculation, and including the population distribution around the reactor site, approximately 35 man-Sv were delivered to the population living in the calculational area during the accident. Since only 133Xe was released, the total dose calculation was considerably easier to estimate than would have been the case if a suite of fission products had escaped from the reactor.
5. INTERPRETATION AND USE

As the model calculations become refined in the post-accident environment, whether from the incorporation of radiological measurements or from fine tuning of model inputs based on the increased understanding of the accident scenario, a clear and concise conceptualization of the radiological effects should emerge. It is the presentation of this current level of understanding in a simple and easily interpreted manner which is of prime importance to the emergency response manager. Once confidence in the model simulation of a given event is established, detailed information may be extracted from graphical presentations of assessment calculations such as those shown above.

Typical uses of model simulations may be illustrated by referring to Figs 3 and 4, as an example. For instance, once model projections of $^{131}$I deposition in selected areas are confirmed by measurement data, available resources for $^{131}$I dose
mitigation may be concentrated, not just in the area immediately adjacent to the accident source location, but also in the isolated areas indicated as potential hot spots by the model dose estimates (such as those indicated by the 100 mSv contours).

Although calculations may be generated for shorter integration periods, or even instantaneous points in time, the spread of the radiological effects as a function of time may be inferred from the dose and deposition patterns integrated for the entire accident period, as shown here. Knowledge of the surface wind history during the 32 hour integration period explains the tridirectional innermost contours of the dose calculations. These, of course, reflect the three dominant wind directions (to the southwest, then northwest, finally north) during the 32 hour period. However, careful interpretation of this would conclude that all areas in a clockwise arc from southwest to northeast of the accident site would have been exposed to the full concentration of the plume for shorter periods of time.

It would, therefore, be from the utilization of such dose assessment products, along with all other information available to the emergency response manager, that appropriate decisions could be made for allocation of proper resources and institution of mitigation efforts.

6. SUMMARY

Discussions in this paper were designed to illustrate a methodology, using dose assessment models, that can be used to help authorities assess the effort of a nuclear accident on the public health and environment and provide them information that can be used during recovery operations. The example accident and three radionuclides used in this study were chosen to illustrate a general methodology.

The integration of model output and field measurements of airborne radioactivity, we believe, has an even greater potential of being a powerful tool for the authorities to use during recovery operations; however, as we discussed earlier, this work is in its infancy and has yet to be formulated for use in an operational environment. Recently (1989) considerable attention has been focused on developing and testing these tools (see, for example, Ref. [4]) and we believe that in the near future operational tools will begin to emerge that will allow the combining of both the modelling and measurements to produce a more complete and accurate dose assessment.

ACKNOWLEDGEMENT

This work was performed under Contract W-7405-Eng-48 from the US Department of Energy to the Lawrence Livermore National Laboratory.
REFERENCES


TACTUS: A CODE FOR SIMULATION OF THE FLOW OF CAESIUM-137 IN URBAN SURROUNDINGS

K.G. ANDERSSON
Rise National Laboratory,
Roskilde,
Denmark

Abstract

TACTUS: A CODE FOR SIMULATION OF THE FLOW OF CAESIUM-137 IN URBAN SURROUNDINGS.

When discussing the impact of a radioactive pollution event on larger populated areas, and subsequent countermeasures to be taken, it is essential to evaluate the consequences with respect to time. In an accident phase it is of greatest importance to gain sufficiently detailed knowledge on how the deposited activity will migrate on various types of surfaces so that predictions can be made as to which reclamation techniques would be the most effective at a given time, and whether or not evacuation programmes should be carried out. The need to model the behaviour of radionuclides deposited in an urban environment was emphasized during the Chernobyl cleanup phase. While most previous computer models, like almost all the literature on the subject written before 1986, dealt with activity flow in rural surroundings, the aim of this model, TACTUS (transfer of activity in urban surroundings), is to use experimental data to reflect the system of retention and migration processes which might typically be found in an urban environment. In its present form, the model is restricted to cover the migration of a single radionuclide — $^{137}$Cs. The urban area under examination can be either user defined or defined by one of four standard environments.

1. INTRODUCTION

The code TACTUS is based on the TAMDYN code (in Pascal) for radioecological modelling [1]. It is based on the linear compartmental model theory; thus the transfer rate for a given surface, $m$, can be written as:

$$\frac{dX_m}{dt} = \sum_{n=1}^{p} S_{nm}X_n - \left( \sum_{n=1}^{p} S_{mn} \right)X_m - L_mX_m + F_m$$

in which $X_m$ and $X_n$ represent the activity in compartments $m$ and $n$, respectively, at a time $t$. $S_{nm}$ is the transfer coefficient from compartment $n$ to $m$. $L_m$ is the transfer coefficient for flow out of the system, etc. (for instance, loss by radioactive decay), while $F_m$ is the initial input to compartment $n$. 

217
This system is time integrated, employing the 4th (5th) order Runge–Kutta method.

In principle, an accurate model would take into consideration the conditions for migration of all radionuclides that might plausibly arise from any accident fallout and on any possible urban surface. To do so, however, one would need an immense data capacity, and since the running time for the present code structure is proportional to the square of the number of compartments, it is recommended that certain generalizations be made. At any rate, the field measurements are so site specific, that no further accuracy would be gained by going into detail.

2. GENERAL MODEL DESCRIPTION

2.1. Programme structure

In its present version, the model comprises 16 compartments, each representing a state in which $^{137}$Cs may be found on an urban surface.

The flow diagram (Fig. 1) shows the principle of the migration model and its assumptions. The dotted lines indicate that the processes taking place are discrete events. The term ‘impermeable surfaces’ covers a mean of all surfaces that are not easily penetrated by water, such as asphalt, concrete, flagstones and cobble stones.

The ‘internal surfaces’ compartment contains the amount of activity deposited on inside walls, furniture, etc.

For those ‘hard’ surfaces on which weathering processes are likely to cause a migration of activity from one type of surface to another (impermeable surfaces, walls and roofs), the migration/retention is accommodated by splitting the activity into three ‘pools’. These represent three different states in which radioactive matter may be found on the particular surface. The first is the mobile phase, representing part of the initially deposited material. The second state is the more strongly bound. However, weathering processes will slowly mobilize the material in this state, and it is here that the third state arises, representing the activity remobilized. It is suggested [2] that the mobile fraction is so loosely bound to the surface that a heavy rain shower giving at least 3 mm of rain within a few hours (depending on the time of the year, porosity, etc.) can wash out the entire content of this fraction. This, though, is likely to be important only in case heavy rain occurs shortly after the contamination. Naturally, this sort of sectioning is rough, since it is made without regard to the specific processes causing the fixation and thus yields no information on the degree of the fixation.

In the model the term ‘fixed’ is supposed to be read as ‘not soluble’. Here it is important to bear in mind that the model parameters are obtained by field measurements which have at least not yet been carried out to such an extent as to give the
FIG. 1. Urban contamination flow chart.
needed information of the strength of fixation. Also, the model structure itself limits the information to be used.

Since the task was to make evaluations for a specific ‘typical European (sub)urban area’ and the fixation processes are highly material dependent, the choice of distinct materials would make the model less generally applicable.

For the activity deposited on trees the model structure is different. On the basis of the available literature it is hard to evaluate the effect of heavy rain or other weathering processes on the activity deposited on the trees. However, a slow transfer of activity is modelled from the trees to the grass, due to rain and wind. The activity fixed on the leaves is removed by leaf fall in the autumn. It is assumed that the leaves all fall on a grassed area during the month of October and that an adjustable fraction is left there to moulder.

2.2. Special features

On comparing TACTUS with other models such as the UK NRPB model EXPURT [3] which has been developed for the same purpose (to provide guidelines for cleanup of radionuclides in the urban environment), one of the evident improvements in TACTUS is the addition of a ‘trees’ compartment. The activity on trees may play a very important role, considering the average doses, as the dry deposition velocity on trees is high because of the size and roughness of the surface area. Also, the gamma ray transport from trees into houses often will pass through windows rather than walls, thereby obtaining a smaller shielding factor than, for instance, that for activity deposited on a soil surface.

The splitting of the soil into different depth compartments allows a simulation of removal of topsoil layers or surface or deep ploughing. It also allows shielding calculations to be made. In principle, it is possible to simulate any decontamination procedure at any time following deposition in this model by removing a part of the activity in a number of compartments.

3. DEGREE OF PROBABILITY IN THE MODEL

One of the things one must make absolutely clear before modelling a process is whether one wishes to draw up guidelines in relation to a specific event using local field measurements for some of the parameters or alternatively to set up a probabilistic model which, given a hypothetical input, can give predictions on the impact which a possible series of accidents might typically have on what are considered to be typical surroundings.

As an example, consider the modelling of the time of the first heavy rain following contamination. As mentioned above, in the model this is taken into account by removing the as yet unfixed part of the radioactivity on the surface, as a discrete
event. The time of this event can, in a probabilistic model, be simulated by a randomizing procedure, but by doing so one would have to accept that the relation to a concrete event is lost. The sensitivity of the model output towards an early heavy rainshower is extreme, since the activity could be distributed differently at an early stage.

In the model TACTUS it is possible to ‘randomize’ this parameter (in practice by associating the event of heavy rain within a time interval with a typical probability). It is, however, suggested that meteorological data and short term predictions, if accessible, be used for guiding countermeasures in relation to a specific event, which is the main task of this code.

From the problem of degree of probability another question arises, concerning the deposition of airborne pollutants on various types of surface.

It seems that once again there are two possibilities: one is to assume that measurements of total depositions on various surfaces are made following the departure of the polluting cloud. As the purpose is to obtain information related to a specified environment, this solution is ideal. But if the model is meant to be entirely probabilistic (which is not the case) a simplified deposition model would be required.

It would be a difficult task to work with the latter, as the standard deviations of the dry deposition velocities are large. Dry deposition, especially in urban areas, is indeed a complicated subject, as it is influenced by a multitude of different phenomena. For instance, the deposition processes are conditioned by the size of the carrier aerosol. Factors such as surface roughness, turbulence, and heat transfer between the city and the canopy make it difficult to determine a reliable dry deposition velocity mean value.

TACTUS contains no air compartment; however, one that takes care of the exchange between the city and the air above could be implemented.

4. INPUT PARAMETERS

4.1. Input structure

The input parameters in TACTUS are estimated values which have been derived from various experiments. These parameters are clearly connected with considerable deviations, as they represent a mean value of a number of results of independent measurements obtained under different conditions and on different materials, all of which fall within the same model category.

The transfer coefficients used in the present version of the model are listed below in the manner in which they are read into the main program.

The first column in Table I indicates that the parameter deals with the transfer of activity from one compartment to another. The next is a single letter indicating the statistical function which the parameter variations follow. This variation by dis-
<table>
<thead>
<tr>
<th>TABLE I. INPUT FILE</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>roof</th>
<th>wall</th>
<th>imps</th>
<th>tree</th>
<th>gras</th>
<th>ints</th>
<th>wfix</th>
<th>wrun</th>
<th>ifix</th>
<th>irun</th>
<th>rfix</th>
<th>sewr</th>
<th>tfix</th>
<th>soil1</th>
<th>soil2</th>
<th>soil3</th>
</tr>
</thead>
<tbody>
<tr>
<td>loss G</td>
<td>1.1500E-01</td>
<td>1.1500E-02</td>
<td>1.0000E-02</td>
<td>4.0000E-01</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>wall</td>
<td>2.3000E-01</td>
<td>2.3000E-02</td>
<td>1.0000E-01</td>
<td>4.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>imps</td>
<td>2.3000E-01</td>
<td>2.3000E-02</td>
<td>1.0000E-01</td>
<td>4.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>roof</td>
<td>2.3000E-01</td>
<td>2.3000E-02</td>
<td>1.0000E-01</td>
<td>4.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>sewr</td>
<td>2.3000E-01</td>
<td>2.3000E-02</td>
<td>1.0000E-01</td>
<td>4.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>tree</td>
<td>2.3000E-01</td>
<td>2.3000E-02</td>
<td>1.0000E-01</td>
<td>4.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>wfix</td>
<td>1.0000E-04</td>
<td>1.0000E-05</td>
<td>1.0000E-05</td>
<td>5.0000E-04</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>wfix</td>
<td>1.0000E-02</td>
<td>1.0000E-03</td>
<td>1.0000E-03</td>
<td>5.0000E-02</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ifix</td>
<td>1.0000E-02</td>
<td>1.0000E-03</td>
<td>1.0000E-03</td>
<td>5.0000E-02</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ifix</td>
<td>4.0000E-04</td>
<td>4.0000E-05</td>
<td>4.0000E-05</td>
<td>9.0000E-04</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>rfix</td>
<td>2.0000E-03</td>
<td>2.0000E-04</td>
<td>1.0000E-03</td>
<td>4.0000E-03</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>rfix</td>
<td>4.6000E-04</td>
<td>4.6000E-05</td>
<td>4.0000E-04</td>
<td>5.0000E-04</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>wrun</td>
<td>4.0000E+00</td>
<td>4.0000E-01</td>
<td>8.0000E-01</td>
<td>6.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>irun</td>
<td>4.0000E+00</td>
<td>4.0000E-01</td>
<td>8.0000E-01</td>
<td>6.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>wrun</td>
<td>4.0000E+00</td>
<td>4.0000E-01</td>
<td>8.0000E-01</td>
<td>6.0000E+00</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>gras</td>
<td>4.6000E-02</td>
<td>4.6000E-03</td>
<td>2.0000E-02</td>
<td>7.0000E-02</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>soil1</td>
<td>4.0000E-04</td>
<td>4.0000E-05</td>
<td>2.0000E-04</td>
<td>6.0000E-04</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>soil2</td>
<td>3.0000E-04</td>
<td>3.0000E-05</td>
<td>2.0000E-04</td>
<td>4.0000E-04</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>soil3</td>
<td>6.0000E-05</td>
<td>6.0000E-06</td>
<td>4.0000E-05</td>
<td>8.0000E-05</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>tfix</td>
<td>1.0000E-03</td>
<td>1.0000E-02</td>
<td>1.0000E-04</td>
<td>5.0000E-03</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

grassconc := gras.
soilconc := soil1.
soil2conc := soil2.
soil3conc := soil3.
sewerconc := sewr.
wallconc := wall + wfix + wrun.
tribution is used only in connection with the Monte Carlo uncertainty analysis (see next paragraph). The letter 'N' means that the distribution is normal. 'U' means it is uniformly distributed over the interval, while 'G' indicates a lognormal distribution function. The third column gives the mean value which is used by itself in deterministic simulations. The fourth column shows the standard deviation for the distribution. The last two columns represent the maximum and minimum values of the parameter produced for the Monte Carlo analysis.

The parameters are written in the dimension of days\(^{-1}\), as the model time step has been chosen to run in units of days.

4.2. Input values

The factor ints > loss has been modelled from the assumption that about 90% of the deposited activity on internal surfaces would have been removed after a 20 day period due to general house cleaning. wall > wfix, the transfer coefficient for fixation of activity deposited on walls, like that for impermeable surfaces (imps), roof material and trees has been set to 0.23 days\(^{-1}\), corresponding to the assumption that 90% of the material would be fixed within 10 days if no activity were removed by weathering processes.

Alas, too little experimental information is available on the subject and the parametrization in TACTUS has been based on very coarse experiments by F.J. Sandalls [2] and L. Warming [4]. To make proper approximations, further investigations in this field will be needed.

It is suggested that the residence period of activity in sewage systems is also about 10 days; hence sewr > loss has been given the value 0.23 days\(^{-1}\).

The velocity with which activity is typically removed by weathering processes can vary widely, depending on the surface type and deposition and weathering conditions. On the basis of available literature [5–7], the weathering on ‘hard’ surfaces was modelled as:

<table>
<thead>
<tr>
<th>Fraction</th>
<th>Half-life</th>
<th>Fraction</th>
<th>Half-life</th>
</tr>
</thead>
<tbody>
<tr>
<td>Roofs:</td>
<td>0.5</td>
<td>1.5 (\times) 10(^3) d</td>
<td>0.5</td>
</tr>
<tr>
<td>Imperm.:</td>
<td>0.5</td>
<td>1.7 (\times) 10(^3) d</td>
<td>0.5</td>
</tr>
<tr>
<td>Walls:</td>
<td>0.8</td>
<td>7.0 (\times) 10(^3) d</td>
<td>0.2</td>
</tr>
</tbody>
</table>

It is assumed that all material weathered off roofs will eventually be lodged in the sewage systems. The roof-drain system is considered to be a part of the roof itself in the model. The flow velocities of washed-off \(^{137}\)Cs from impermeable surfaces and walls to grassed areas are considered to be equal in magnitude. The time constant for the flow of loosened activity is generally set to 4 days\(^{-1}\). Exact knowledge is unobtainable from the available literature.
The behaviour of $^{137}$Cs deposited on grassed areas has been modelled on the grounds of recommendations from Krieger and Burmann [8], who experimentally recorded a half-life of 15 days for the weathering of grass-deposited activity on the topsoil. It is recommended to adjust the assumed wet/dry deposition relationship in the light of the relationship between the fraction initially deposited on the ground and that deposited on the rougher grassed surface.

The migration of $^{137}$Cs in soil has been modelled from various experiments [9, 10]. Many soil profiles have been examined through the years for a number of different soil textures and different time steps from deposition. The migration rates in this code have partly been based on calculations on which activity at a particular soil depth would correspond to the dose rates at 1 m above the ground recorded by Gale et al. [9].

Of course, the parameters thereby obtained may be applicable only within a limited period (a few years), but discrete changes of all parameters in TACTUS can be realized, if necessary.

If a probabilistic solution is desired, a number of other inputs regarding, for instance, the deposition can be specified from the input file. These are, for example, parameters such as the angle of rainfall, dry deposition velocities on various surfaces and interception factors for wet deposition.

### 4.3. Input restrictions

To run the code on a personal computer the total number of parameters in the input file (transfer coefficients and secondary parameters) must be kept to a maximum of 38, of which no more than 22 may be transfer coefficients. A maximum of 10 outputs can be taken at a time, the total number of parameters and outputs being no larger than 42, though. The maximum number of events controlled by the input file is 50. Table I shows an input file example in which the surface concentrations for six different surfaces are taken as output and can be displayed on the screen, printed out, or saved as a disk file for further operations.

### 5. RELIABILITY OF THE MODEL OUTPUTS

It was reported by the UNSCEAR committee in their 1982 report on ionizing radiation [11] that a regrettable tendency, at least at that time, had been noticed in relation to modelling techniques, namely, that the results of more or less complicated predictive models were evaluated too uncritically. The committee made a proposal one can regard as common sense: before the results of a model are used for recommendatory purposes and relied upon, the model should be validated.
TABLE II. ARITHMETIC MEANS AND STANDARD DEVIATIONS
(Sample size = 1000)

<table>
<thead>
<tr>
<th>Variable</th>
<th>Mean</th>
<th>Std Dev</th>
<th>Minimum</th>
<th>Maximum</th>
<th>C. V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>ints&gt;loss</td>
<td>0.08E-01</td>
<td>0.05E-01</td>
<td>1.00E-02</td>
<td>4.00E-01</td>
<td>97.42</td>
</tr>
<tr>
<td>wall&gt;wfix</td>
<td>1.98E+00</td>
<td>1.10E+00</td>
<td>1.04E-01</td>
<td>3.99E+00</td>
<td>55.65</td>
</tr>
<tr>
<td>imps&gt;ifix</td>
<td>2.06E-00</td>
<td>1.10E+00</td>
<td>1.05E-01</td>
<td>3.99E+00</td>
<td>53.61</td>
</tr>
<tr>
<td>roof&gt;rfix</td>
<td>2.00E-00</td>
<td>1.12E+00</td>
<td>1.07E-01</td>
<td>4.00E+00</td>
<td>56.27</td>
</tr>
<tr>
<td>sewr&gt;loss</td>
<td>2.29E-01</td>
<td>2.27E-02</td>
<td>1.63E-01</td>
<td>2.93E-01</td>
<td>9.91</td>
</tr>
<tr>
<td>tree&gt;tfix</td>
<td>2.07E-00</td>
<td>1.13E+00</td>
<td>1.02E-01</td>
<td>4.00E+00</td>
<td>54.55</td>
</tr>
<tr>
<td>wfix&gt;wrun</td>
<td>9.36E-05</td>
<td>9.89E-06</td>
<td>5.92E-05</td>
<td>1.34E-04</td>
<td>9.93</td>
</tr>
<tr>
<td>ifix&gt;irun</td>
<td>3.97E-04</td>
<td>3.88E-05</td>
<td>2.64E-04</td>
<td>5.22E-04</td>
<td>9.76</td>
</tr>
<tr>
<td>rfix&gt;sewr</td>
<td>4.54E-04</td>
<td>2.70E-05</td>
<td>4.00E-04</td>
<td>5.00E-04</td>
<td>5.96</td>
</tr>
<tr>
<td>wrun&gt;gras</td>
<td>4.00E+00</td>
<td>3.97E-01</td>
<td>2.79E+00</td>
<td>5.44E+00</td>
<td>9.90</td>
</tr>
<tr>
<td>irun&gt;gras</td>
<td>3.98E+00</td>
<td>3.98E-01</td>
<td>2.87E+00</td>
<td>5.28E+00</td>
<td>9.99</td>
</tr>
<tr>
<td>irun&gt;sewr</td>
<td>4.01E+00</td>
<td>4.00E-01</td>
<td>2.60E+00</td>
<td>5.07E+00</td>
<td>9.97</td>
</tr>
<tr>
<td>wrun&gt;imps</td>
<td>4.03E+00</td>
<td>3.83E-01</td>
<td>2.95E+00</td>
<td>5.37E+00</td>
<td>9.51</td>
</tr>
<tr>
<td>gras&gt;soil</td>
<td>4.61E-02</td>
<td>4.56E-03</td>
<td>3.23E-02</td>
<td>5.92E-02</td>
<td>9.89</td>
</tr>
<tr>
<td>soil2&gt;soil2</td>
<td>4.01E-04</td>
<td>3.91E-05</td>
<td>2.87E-04</td>
<td>5.10E-04</td>
<td>9.76</td>
</tr>
<tr>
<td>soil2&gt;soil3</td>
<td>3.01E-04</td>
<td>3.03E-05</td>
<td>2.00E-04</td>
<td>4.00E-04</td>
<td>10.05</td>
</tr>
<tr>
<td>soil3&gt;loss</td>
<td>6.00E-05</td>
<td>5.84E-06</td>
<td>3.79E-05</td>
<td>7.89E-05</td>
<td>9.74</td>
</tr>
<tr>
<td>tfix&gt;loss</td>
<td>1.00E-03</td>
<td>9.90E-05</td>
<td>6.42E-04</td>
<td>1.37E-03</td>
<td>9.85</td>
</tr>
<tr>
<td>grassconc</td>
<td>1.06E+03</td>
<td>5.75E+01</td>
<td>9.05E+02</td>
<td>1.25E+03</td>
<td>5.42</td>
</tr>
<tr>
<td>soil1conc</td>
<td>5.00E+03</td>
<td>5.73E+01</td>
<td>4.81E+03</td>
<td>5.15E+03</td>
<td>1.15</td>
</tr>
<tr>
<td>soil2conc</td>
<td>2.24E+01</td>
<td>2.19E+00</td>
<td>1.59E+01</td>
<td>2.86E+01</td>
<td>9.79</td>
</tr>
<tr>
<td>soil3conc</td>
<td>3.94E-02</td>
<td>5.56E-03</td>
<td>2.57E-02</td>
<td>5.92E-02</td>
<td>14.12</td>
</tr>
<tr>
<td>sewerconc</td>
<td>1.47E+01</td>
<td>1.53E+00</td>
<td>9.70E+00</td>
<td>1.93E+01</td>
<td>10.40</td>
</tr>
</tbody>
</table>

The code presented offers several options for evaluating the inherent uncertainties. As mentioned in Section 4.1., it is possible to vary the input parameters in order to make a large number of simulations and thereby observe the deviations.

The option 'Monte Carlo simulation' is based on a variation of the parameters within certain sections of distributions characteristic of the variation of the specific parameter. The verification of the code itself can be made by examining the squared correlation coefficients (as percentages) between input parameters and the content at a time in the compartments (Monte Carlo uncertainty analysis). Hereby it is possible to evaluate the output dependency on inputs and thus suppress those model pathways which in all simulation runs at all times on different deposition inputs are revealed to have no or very little effect on the model predictions. It is against this background that the model has reached its present shape.

It is, however, important to bear in mind that the effect of a pathway greatly depends on the possible discrete events (e.g. rain showers) which may cause a completely different situation, so the uncertainty analyses should be examined thoroughly.
Table II shows the result of 1000 Monte Carlo simulations 12 days after deposition in an environment of semi-detached houses. It is stipulated that there had been no heavy rain.

If correlations are known from previous experiences, it is possible to introduce these in the model. Any two input parameters may be correlated before running the program.

The option sensitivity analysis is performed in a manner similar to the uncertainty analysis. Still, if the sensitivity analysis is chosen, the input is varied randomly within 1% of the input-defined mean value of an interval centred at the mean value.

6. SIMULATION RUNS

In order to survey the consequences of depositions on different urban surfaces, the relations between kerma rate and Bq/m² in four different urban surroundings (as published by R. Meckbach et al. [12]) have been implemented. Hereby, it is possible
to observe the development of the kerma rate with time at a given location in or outside a building. Figure 2 shows the dynamic progress of the kerma rate to the attic of a house of prefabricated parts [12], due to a combined wet/dry deposition on various surface-types on June 1st. A heavy rainshower occurred after 7 days. The simulations were based on experimentally found relative source strengths for Chernobyl wet and dry deposited material [13].

7. CONCLUSION

The computer program TACTUS has been developed in order to estimate the transfer of $^{137}\text{Cs}$ within an urban environment. It is intended to contribute to the assessment of consequences to humans.

The code uses experimental data. Its purpose is to give as realistic predictions as possible of the activity level in a specific environment at any time following deposition.

The present version of the code enables the operator to obtain dynamic solutions of the compartment system reflecting $^{137}\text{Cs}$ transfer and to evaluate the parameter sensitivities by 'controlled randomizing'. As an additional option, kerma rates for residences in four different (sub)urban environments can be calculated.

The compartment structure permits a simulation of a decontamination.

The code allows for evaluating parameter importance by input–output correlations.

To give proper recommendations, further studies on some model parameters will be required.

REFERENCES


PECULIARITIES OF NUCLEAR FUEL INSIDE THE CONFINEMENT SYSTEM OF THE DAMAGED REACTOR AT CHERNOBYL

S.T. BELYAEV, A.A. BOROJOY, A.Yu. GAGARINSKIJ,
I.V. Kurchatov Institute of Atomic Energy,
Moscow,
Union of Soviet Socialist Republics

Abstract

PECULIARITIES OF NUCLEAR FUEL INSIDE THE CONFINEMENT SYSTEM OF THE DAMAGED REACTOR AT CHERNOBYL.

The paper describes the contents of the encapsulating envelope surrounding the damaged Chernobyl reactor. These include metal deformed by the explosion, remnants of the core and fuel in various forms, including a solidified lava-like block. Information is given on the fuel mass distribution within the confined area. It is concluded that there is no self-sustaining nuclear chain reaction currently within the area. The maximum future potential hazard is probably the reactor core parts which remained intact. Over forty holes have been bored to date to enable the examination of core samples. Various methods have been applied to determine the radiation hazard and systematic core sampling has been performed. It is concluded that all variants of the fuel within the confined area are well below criticality. Now that access to the main areas has been achieved where fuel has accumulated, neutron absorbers are being introduced to exclude the occurrence of a chain reaction in the future; measures are being taken to prevent the discharge of radioactive dust and to maintain the reliability of the structural elements.

The construction of a confinement system (or an encapsulating envelope) for the damaged Chernobyl reactor was finished in November 1986 (see Fig. 1 and the cover photograph of this book). The confinement system eliminates the danger of radioactivity spreading from the reactor debris, and its thick side walls reliably absorb penetrating radiation. This marks the end of one of the most important stages of the remedial work following the Chernobyl breakdown.

What does this construction towering 60 m over the ground contain?

It contains metal deformed by the explosion, remnants of the reactor core and several hundreds of completely destroyed, partially collapsed or practically intact rooms of the NPP’s damaged fourth block, which contain more than 95% (i.e. more than 180 t) of the nuclear fuel. By the autumn of 1986 each gram of this fuel had an activity of ≈ 1 Ci.

At the time the confinement system construction was completed it was known that the fuel inside it was contained at least in three different variants.
FIG. 1. A photo of the confinement system, taken from a helicopter, as viewed from the northwest.

The first consists of fragments of the destroyed reactor core: fuel assemblies, fuel elements, etc.

The second comprises dispersed fuel. During the breakdown it was thrown in greater or smaller amounts into all the rooms of the system; it lay on surfaces; penetrated into concrete; was dispersed in the air in the form of aerosols. These aerosol particles, measuring from fractions of a \( \mu m \) to hundreds of \( \mu m \) and having all the characteristic features of the fourth block fuel, are called 'hot fuel particles'.

The third variant was found in one of the column reactor rooms. In this room there is a solidified 'drop' with a volume of a few cubic metres, which consists of a vitreous black mass. The radiation, 8000 R/h, did not allow one to approach this drop (see Fig. 2). From studying the drop material, it was found that it is comprised of silicon dioxide (95-96%), fuel (2-3%), graphite, iron, etc. Evidently, this substance flowed in a molten state in the form of a peculiar lava from the storeys above and solidified after travelling some tens of metres from the reactor core.

This variant of the fuel mass was also found on the 1st and 2nd floors of the suppression pool (levels 0 and 3 m). Evidently, here the lava got into the rooms filled with water (the water remained there till the end of the active stage of the accident).
FIG. 2. A photo of solidified melt of the fuel-containing 'lava' found in one of the column reactor rooms at a distance of a few tens of metres from the reactor core.

FIG. 3. Density distribution of thermal fluxes above the damaged reactor surface in October 1986: 1 — contour of reactor pit; 2 — contour of one of the decay tanks for exhausted fuel.
via steam discharge tubes and the vitreous mass resembled pumice. Finally, in 1989 an analogous 'lava' was found in 'steam-stream' rooms through which steam would flow in the event of an accident.

What is known about the fuel mass distribution inside the confinement system itself?

The data obtained by 1987 were fairly general. Thus, the thermal measurements performed according to the 'Buy' program showed that in the initial period after the accident, heat removal from the sources contained within the confinement system was realized mainly by natural convection. This made it possible to estimate the lower limit of the amount of fuel within the confinement system by using the measured distribution of convective thermal fluxes above the reactor debris (Fig. 3). With a confidence level of 0.95 this amount exceeds 87\% of the fuel loading of reactor No. 4 at the moment of breakdown.

It was known that the main masses of fuel were situated in the following rooms:

— in the reactor central hall and under the cascade wall of the confinement system (thrown out during the explosion);
— in the decay tank (contained there before the accident);
— in the reactor pit (remnants of the reactor core);
— in the ‘column reactor’ rooms using the space between the reactor supporting columns, including the load bearing, cross-shaped room (room no. 305/2), in the adjacent corridors and rooms, in the steam-stream rooms, on the two floors of the suppression pool (thrown out during the accident and flowing in the form of lava).

Is the fuel which remains in the confinement system dangerous?

This question refers primarily to the radiation hazard. Immediately after the breakdown, the question was posed whether a nuclear chain reaction could occur in the damaged reactor. Unambiguous evidence was obtained that no observable self-sustaining chain reaction was taking place. The experience obtained during erection of the confinement system and at the beginning of its operation (1986–1987) also supports the absence of any self-sustaining chain reaction. Figure 4 presents the values for the thermal flux density, the residual heat release power and that of the gamma radiation dose in the rooms as of 1 January 1987 to 20 April 1987. One can see that the decrease of thermal and radiation fields with time corresponds to the calculations.

In spite of the fact that the reactor rooms and, consequently, the fuel situated in them were subjected to active external effects after the breakdown (vibrations in filling up the reactor debris, hitting of concrete in constructing the confinement system, etc.), no dangerous tendencies were revealed in the behaviour of this fuel. All the above-mentioned factors allow one to conclude that the fuel accumulations which formed as a result of the breakdown are subcritical by a large margin.
Nevertheless, there was no guarantee that in the course of subsequent years a 'critical assembly' would not be formed as a result of unpredictable destruction of the damaged structural elements inside the confinement system and displacement of the fuel masses.

The reactor core parts which remained intact in the reactor plenum could represent the maximum potential hazard.

Up to 1988 the measurements were taken mainly at the periphery of the reactor. It was imperative to find methods and means of approaching the places where fuel had accumulated, so as to be able to examine potentially dangerous areas. Therefore the work scheme for 1988 envisaged getting into the reactor plenum, the column reactor rooms and the steam-stream rooms by boring holes from the accessible and decontaminated rooms. Via these holes, visual, photographic and teleobservations were to be performed, radiation and thermal measurements were to be taken to detect the position of the fuel and to obtain more accurate information on the condition of the engineering constructions.

To date more than 40 holes have been bored. Almost all of them required the development of a unique procedure for extraction of highly active core samples.
It has been established via these holes that not only the 2000 t upper 'lid', i.e. the ‘E’ system, was blown away by the explosion but also the foundation plate of the reactor, i.e. the ‘OR’ system was pressed in by approximately 4 m having crushed the metalwork in room 305/2.

There are no significant volumes of regular graphite moderator blocks in the plenum of the apparatus. The debris, consisting of graphite blocks, structural elements of the reactor and concrete spilled in constructing the confinement system, was found in the ‘OR’ system. This concrete also filled up room 305/2, into which a considerable part of the fuel had been discharged.

Having melted the sand and other materials as a result of the fire and internal heat release, some of the fuel formed flows of lava in the corridors and rooms of the lower part of the reactor as mentioned above. Since September 1988, the investigators have several times penetrated the northern part of the column reactor room, the northern part of the steam-stream rooms and elsewhere.

To monitor the object, the holes are equipped with neutron and gamma chambers, thermal control devices, of which there are more than 70, and which form a surveillance monitoring system. All the detectors are connected to the ‘Finish’ surveillance data acquisition system, which performs detection, storage, and representation of information on the spot.

The radiation hazard from the fuel was determined using different methods, including the direct measurement of subcriticality. In this case, neutrons were injected into the fuel masses by a pulsed source and the breeding properties of the medium were determined by the decay time of the neutron flux after injection.

A systematic sampling of core samples was performed in the course of boring the holes. Gamma spectrometry and radiochemical methods were used to study the characteristics of the fuel contained in the samples: the degree of fuel burnup, the nuclide content, etc.

Taken as a whole, the information available on the fuel inside the confinement system indicates that it is subcritical by a large margin. This applies to all the fuel modifications observed.

Access to the main areas of the fuel accumulation allowed us to begin dealing with it, e.g. by implantation of neutron absorbers (in order to exclude the occurrence of a self-sustaining chain reaction with any imaginable shifts of the fuel), spraying of special aerosol-depressing compounds (in order to exclude the discharge of radioactive dust) and active measures for maintaining the reliability of the structural elements.
EXPERIENCE OF THE GOVERNMENT OF THE UKRAINIAN SSR IN CLEANING UP THE ACCIDENT AT THE CHERNOBYL NUCLEAR POWER PLANT.

The paper examines problems arising in connection with the cleanup activities implemented by the Government of the Ukrainian SSR in response to the Chernobyl nuclear power plant accident.

ИЗ ОПЫТА РАБОТЫ ПРАВИТЕЛЬСТВА УССР ПО ЛИКВИДАЦИИ ПОСЛЕДСТВИЙ АВАРИИ НА ЧЕРНОБЫЛЬСКОЙ АЭС

Рассмотрены вопросы, связанные с практической деятельностью Правительства Украинской ССР по ликвидации последствий на Чернобыльской АЭС.

Три с половиной года прошло с момента аварии на Чернобыльской АЭС, но и сегодня вопросы, связанные с ликвидацией ее последствий, волнуют не только нас, но и многих людей мира. Не случайно в мае прошлого года в городе Киеве состоялась научная конференция по медицинским аспектам аварии на Чернобыльской АЭС, в которой приняли участие ученые из зарубежных стран — руководство МАГАТЭ. Эта авария явилась беспрецедентной по масштабам причиненного ущерба, объему работ по ликвидации ее последствий, а также по психологическому воздействию на общественное мнение.

Как известно, авария произошла на Чернобыльской АЭС ночью 26 апреля в 1 час 23 минуты.

Непосредственно после возникновения аварии подразделения пожарной охраны АЭС городов Припять и Чернобыль приступили к ликвидации очагов пожара. К 5 часам утра пожар на IV блоке был ликвидирован. Здесь уместно напомнить о подвиге, совершенном в первые минуты после аварии нашими пожарными. Они ценой своих жизней предотвратили катастрофу, последствия которой даже трудно себе представить. Им за это должен быть благодарен не только наш народ, но и все человечество.
РИС. 1. Зоны радиоактивного загрязнения (по состоянию на 30.06.86 г.)
На недавней сессии Верховного Совета УССР поддержано предложение правительства республики о сооружении в Киеве памятника-мемориала героям-пожарным, чтобы не только мы, но и последующие поколения знали и чтили имена тех, кому они обязаны своей жизнью.

В 3 часа 30 минут о случившемся стало известно руководству республики, а в 6 часов утра в район бедствия прибыли представители Совета Министров УССР, руководители Киевского облисполкома, министерств и ведомств республики.

Отмечу, что никакой растерянности не было. Все действовали четко и оперативно.

В первые часы, одновременно с мерами по тушению пожара, проводилась оценка радиационной обстановки. Силами гражданской обороны, Укргидромета, службы дозиметрического контроля станции была проведена воздушная и наземная радиационная разведка в районе аварии и прилегающей территории. Первая информация о радиационной обстановке была предоставлена правительству в 10 часов утра (рис. 1).

Анализ обстановки показал, что радиоактивное загрязнение местности с повышенной дозой излучения охватило в республике территорию с населением свыше 140 тыс. человек. В зоне загрязнения оказались 169 населенных пунктов, в том числе города Припять и Чернобыль. В этих условиях от правительства Украины потребовалось экстренное осуществление целого ряда организационных мер.

Поэтому немедленно в министерствах, ведомствах и органах Советской власти были созданы штабы и оперативные группы, укомплектованные квалифицированными специалистами. Размещались они в непосредственной близости к зоне работ — в г. Чернобыле. Все органы управления, принимавшие участие в ликвидации последствий аварии, перешли на круглосуточную работу. Их деятельность на месте координировал один из заместителей Председателя Совета Министров республики.

Такая организация работы правительства Украины позволила в кратчайшие сроки сосредоточить производственный и научный потенциал, необходимые людские и материальные ресурсы, на решение неотложных задач:

— обеспечение безопасности населения;
— локализация источника радиации;
— оказание медицинской помощи пораженному персоналу станции и личному составу пожарных частей;
— безаварийная остановка первого, второго и третьего блоков.

Проведение указанных мероприятий дало возможность спланировать и приступить к выполнению целого ряда крупномасштабных работ по ликвидации последствий аварии.
Учитывая сложность создавшейся обстановки, Министерству автомобильного транспорта и Управлению железной дороги было поручено сосредоточить в районе г. Припять необходимые транспортные средства для возможной эвакуации населения, а эвакоофициацию области — пройти подготовку к вывозу людей. К вечеру 26 апреля в районе г. Припять находилось 1200 автобусов и около 200 бортовых автомобилей. На ж/д ст. Янов подготовлено два дизельных поезда на 1500 мест.

Прибывшая на место аварии к исходу дня 26 апреля союзная правительственная комиссия, проанализировав представленную информацию, и основываясь на заключении специалистов Минздрава СССР, не сочла необходимым приступить к немедленной эвакуации людей (нормы Минздрава СССР согласованы с МАГАТЭ в 1971 г.).

И только к утру 27 апреля в связи с осложнениями положения было принято решение об эвакуации с 14 часов населения г. Припять и ж/д ст. Янов. В течение 3-х часов почти 50 тыс. человек были вывезены и временно размещены в безопасных районах.

Справочная информация.

В соответствии с "Временными методическими указаниями Минздрава СССР, для разработки мероприятий по защите населения в случае аварии ядерных реакторов" (созданными на основании рекомендаций МАГАТЭ), эвакуация в случае аварии на АЭС (и др.) обязательна при условии возможности формирования у населения суммарных доз облучения за 1-4 сут. равных или более 750 мЗв. Мы приняли 250 мЗв. Этот критерий при решении вопроса об эвакуации населения г. Припять и 10-км зоны превышен не был. В соответствии с прогнозом и расчетом, жители не получили доз, превышающих 200 мЗв.

По мере уточнения радиационной обстановки правительственной комиссией было принято решение и проведена эвакуация населения десятикилометровой зоны, а также ограждена ее территория. В дальнейшем были отселены и жители тридцатикилометровой зоны (которая стала называться зоной отселения) (табл. 1).

Эвакуация населения проводилась в три этапа, организовано и в сжатые сроки (с 27 апреля по 7 мая 1986 года). За это время вывезено 92 тыс. человек. Население было размещено с обоюдного согласия у жителей близлежащих районов, у знакомых и родственников. Одновременно было перемещено 66 тыс. голов скота.

Переселение большого количества людей потребовало от правительства республики оперативного решения целого комплекса задач, связанных с их размещением, организацией жизни и быта, трудоустройством и компенсацией понесенного материального ущерба.
ТАБЛИЦА I. ПОСЛЕДОВАТЕЛЬНОСТЬ ЭВАКУАЦИИ ПРИ АВАРИИ НА ЧАЭС И ЧИСЛЕННОСТЬ ЭВАКУИРОВАННОГО НАСЕЛЕНИЯ

<table>
<thead>
<tr>
<th>Время проведения</th>
<th>Наименование районов (населенных пунктов)</th>
<th>Количество эвакуированных населенных пунктов</th>
<th>Численность эвакуированных (тыс.чел.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>14.00—18.00</td>
<td>г. Припять (Киевская область, УССР)</td>
<td>1</td>
<td>49 360</td>
</tr>
<tr>
<td>27.04.86 г.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>10.00—02.05.86 г.</td>
<td>Чернобыльский район — 10-ти-километровая зона вокруг ЧАЭС (Киевская обл., УССР)</td>
<td>15</td>
<td>10 779</td>
</tr>
<tr>
<td>18.00—03.05.86 г.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>09.00—04.05.86 г.</td>
<td>Чернобыльский район — 30-ти-километровая зона вокруг ЧАЭС (Киевская обл., УССР)</td>
<td>33</td>
<td>30 136</td>
</tr>
<tr>
<td>18.00—07.05.86 г.</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Итого 49 90 275

Для оказания медицинской помощи было привлечено более 2 тыс. врачей, 2,5 тыс. средних медицинских работников, переведены на круглосуточное дежурство медицинские учреждения. Особое внимание уделялось предупреждению вспышек инфекционных заболеваний, для чего было привлечено 11 санэпидбригад, развернуто 42 передвижных и стационарных санитарно-обмывочных пункта. Для обеспечения населения продуктами питания и товарами первой необходимости задействовано около 700 предприятий торговли и общественного питания.

Остро стал вопрос снабжения питьевой водой, так как часть открытых источников водоснабжения была загрязнена. Усилия правительства республики и местных органов власти направлялись на увеличение прежде всего мощностей водопроводов из подземных источников, бурение и ввод дополнительных артезианских скважин, очистку и герметизацию шахтных колодцев, а также закрытие непригодных.

Правительство республики проявило особую заботу о детях, беременных женщинах и матери с новорожденными. Лучшие санатории, профилактории и пансионаты, пионерские лагеря на берегу Черного моря, начиная со знаменитого "Артека", летом 1986 г. были предоставлены в их распоряжение. Всего было оздоровлено более 200 тыс. школьников и свыше 300 тыс. матерей с детьми из зоны отселения, ряда прилегающих областей и города Киева.
Эвакуированное население в первые дни бесплатно обеспечивалось одеждой и постельными принадлежностями, питанием. Была оказана единовременная денежная помощь, выплачена компенсация за оставленные строения и имущество. Всего на эти цели израсходовано свыше одного миллиарда рублей. Таким образом, первоочередные задачи по обеспечению эвакуированного населения в районах размещения в основном были решены. Значительную роль в решении этих вопросов сыграли силы и средства гражданской обороны республики, соответствующих подразделений министерств и ведомств. Имевшиеся у них планы на случай чрезвычайного положения позволяли быстрее принимать необходимые решения.

В июне месяце стало ясно, что проживание в зоне отселения будет невозможно в течение длительного времени, поэтому перед правительством Украины стал вопрос размещения людей на постоянное жительство и трудоустройства. Для этого в республике было выделено 39 тыс. квартир, и уже к ноябрю 1986 года, в основном, все семьи были обеспечены благоустроенным жильем и работой.

В последующем в местах расселения эвакуированных, а также населенных пунктах, прилегающих к 30-километровой зоне (зона жесткого контроля), возведено около 600 объектов социально-культурного назначения (магазины, столовые, детские сады, школы, коммунально-бытовые объекты). Построено и капитально отремонтировано около тысячи километров автомобильных дорог с твердым покрытием, заасфальтировано 40 млн. кв.м территории.

Кроме решения вопросов, связанных с эвакуацией людей, правительство Украины совместно с учеными определило главные направления дальнейшей работы по ликвидации последствий аварии:

— дезактивационные работы;
— защита водных ресурсов;
— осуществление дозиметрического контроля;
— организация медико-санитарного обеспечения;
— агропромышленные мероприятия;
— меры, связанные с возобновлением работы АЭС.

Начиная с первых дней после аварии, с целью снижения возможных дозовых нагрузок на людей, были развернуты дезактивационные работы на территории Чернобыльской АЭС и в населенных пунктах зоны жесткого контроля. Для этих целей привлекались спецвойска химзащиты, строительные подразделения министерств и ведомств республики.

Многие приемы и способы дезактивации применялись впервые в практике. Так, предложенная сотрудниками Академии наук СССР и УССР обработка развала четвертого энергоблока и территории станции
специальными пылеподавляющими составами привела к снижению концентрации активных аэрозолей в воздухе промплощадки в 100–1000 раз. Химико-биологический способ закрепления пылящих территорий с успехом применялся также и в 10-километровой зоне.

**Справочная информация.**

Пылеподавляющими составами (представляющими собой отходы производства) в 1986–1987 гг. обработано около 7 тыс. гектаров пылящих территорий, а также до 1,5 тыс. км грунтовых и обочин шоссейных дорог. На эту работу израсходовано более 100 тыс. т указанных составов.

Способом химико-биологического закрепления обрабатывали в первую очередь обширные песчаные территории в районе г. Припять, а также территории, лишенные растительного покрова в результате дезактивационных работ. Всего в 10-километровой зоне Чернобыльской АЭС закреплено с помощью этого способа более 800 гектаров.

С мая 1986 г. не прекращаются работы по дезактивации населенных пунктов. При этом дезактивировано 7,5 тыс. зданий и помещений, очищено и дооборудовано 25 тыс. колодцев, снято и вывезено 540 тыс. куб. м зараженного грунта, проведена санитарная очистка территории на полутора миллионах квадратных метров.

Вопрос защиты от радиационного загрязнения источников водоснабжения для нас был одним из главных, т. к. вынос радиоактивности в р. Днепр, загрязнение других водных источников представляло огромную опасность для 32 млн. человек, проживающих в бассейне реки. Особенно остро он стал осенью и весной 1986–1987 гг. Поэтому правительством были приняты экстренные меры по защите водных ресурсов (рис. 2).

В нижнем течении р. Припять, притоках Днепра и Киевском водохранилище было построено и введено в эксплуатацию 131 гидротехническое сооружение типа фильтрующих и глухих дамб общей протяженностью около 18 км, препятствующих выносу радиоактивных веществ с наиболее загрязненной территории, сооружены 4 донные ловушки и 5 подводных дамб, выполнены земляные работы в объеме свыше 5 млн. куб. метров. Постоянные наблюдения за работой этих сооружений свидетельствуют о их высокой эффективности. По оценкам специалистов комплекс водоохранных мероприятий позволил снизить загрязнение воды в устье реки Припять и Киевском водохранилище в 5–7 раз.

Был осуществлен ряд дополнительных мер по созданию резервного водоснабжения в случае ухудшения качества питьевой воды. В Киеве и других городах, находящихся ниже по течению реки Днепр, пробурено
РИС.2. Водоохранные мероприятия.
дополнительно около 1 тыс. артезианских скважин, проложено свыше 1 тыс. км водопроводов и водопроводных сетей.

На всех водопроводах, использующих днепровскую воду, по рекомендации Академии наук республики были перезаряжены фильтры водопроводных сооружений с использованием активированного угля и цеолита. В итоге выполнение водоохранных мероприятий, строительство водопроводов от подземных источников позволили обеспечить потребность населения и народного хозяйства качественной питьевой водой. Уровень радиоактивного загрязнения ее даже в самый напряженный период не превышал допустимых нормативов.

Следует отметить, что в те напряженные дни прорабатывались и другие варианты по защите водных ресурсов. Некоторыми специалистами предлагалось, в частности, очистить пруд-охладитель АЭС и Киевское водохранилище от радиоактивного загрязнения путем сброса воды вниз по течению Днепра. Но тщательная проработка этого варианта с учеными показала его несостоятельность, помогла избежать загрязнения вод Днепра и Черного моря.

Важным направлением работы для нас было предотвращение распространения радиоактивных веществ за пределы тридцатикилометровой зоны. С этой целью были развернуты 56 постов дозиметрического контроля и построены три комплексных пункта перегрузки грузов, дезактивации техники и санитарной обработки людей.

На основных маршрутах, ведущих в г. Киев, и в пригородной зоне города было организовано свыше 60 постов и пунктов, на которых ежесуточно контролировалось до 40 тыс. единиц транспорта и обрабатывалось 200–300 автомашин, проходило санитарную обработку до 10 тыс. населения.

Одной из сложных и ответственных задач, которую пришлось решать правительственным органам республики при ликвидации последствий аварии, была организация и осуществление оперативного дозиметрического контроля на территории всей Украины. Для чего было привлечено 1130 учреждений сети наблюдения и лабораторного контроля, около 15 тыс. постов радиационного наблюдения и создано 94 подвижных лаборатории.

Нужно сказать, что в первый период остро ощущался недостаток всех видов дозиметрических приборов, специальной одежды и средств защиты, особенно в зоне, непосредственно прилегающей к АЭС.

По поручению Совета Министров УССР в сжатые сроки была определена номенклатура необходимых приборов и приняты меры по резкому увеличению их производства. Это позволило быстро обеспечить первоочередные потребности в дозиметрической аппаратуре и средствах защиты.
Правительство Украины работало в тесном контакте с ученными. В короткий срок была создана система мониторинга по радиометрическому контролю окружающей среды, в разработке которой активное участие приняли Академия наук УССР, министерства и ведомства республики. Автоматизированный сбор и отработка данных о радиационной обстановке в тридцатикилометровой зоне осуществляется с помощью электронно-вычислительной техники (рис.3).

Созданный банк информации позволил прогнозировать и моделировать складывающуюся радиоэкологическую ситуацию. На основании этого правительство делало необходимые выводы, принимало обоснованные решения.

Одним из важных аспектов мониторинга было выделение так называемой зоны жесткого контроля, в которую вошли 77 населенных пунктов, где продолжают проживать люди. Этому населению уделяется особое внимание.

С целью ограничения потребления продуктов питания местного производства и личных подсобных хозяйств организован завоз "чистых" продуктов и компенсируются затраты на их потребление. В школах и детских дошкольных учреждениях осуществляется трехразовое бесплатное питание.

В связи с необходимостью обеспечения безопасного проживания людей в этих районах в течение длительного времени, перед учеными страны стал вопрос о величине предельной индивидуальной дозовой нагрузки.

Была разработана концепция, согласно которой максимальная доза облучения человека за всю его жизнь не должна превысить 350 мЗв. С учетом этой концепции и реально сложившейся радиационной обстановки, правительством республики принято решение о дополнительном отселении в 1989-1991 гг. жителей еще из 14 наиболее загрязненных населенных пунктов, в которых нет гарантии обеспечения указанного дозового предела.

Много сделано правительством и по организации медико-санитарного обеспечения населения, подверженного воздействию повышенной радиации в результате Чернобыльской аварии.

Всего в республике было проведено свыше 440 тыс. осмотров с обследованием взрослых и детей, йодной профилактикой было охвачено 4,5 млн человек, что уменьшало нагрузку на щитовидную железу в 2–4 раза. Для приема пораженных участников ликвидации аварии на Чернобыльской АЭС были созданы 5 специализированных клиник в научно-исследовательских институтах и медицинских вузах Киева. Для эвакуированных организовывались специализированные отделения в больницах республики.
Плотности загрязнения почв Cs-137
Киевской области.

Рис. 3. Радиационная обстановка в Киевской области (по состоянию на 01.08.89 г.).
В 1986 г. в Киеве был создан Всесоюзный научный центр радиационной медицины Академии медицинских наук СССР. В настоящее время этот центр, будучи оснащенным новейшим оборудованием и хорошо укомплектованным, осуществляет координацию всей деятельности научного, профилактического и лечебного характера.

Широкий комплекс научнообоснованных мероприятий осуществляется в агропромышленном секторе.

Проведено сплошное радиологическое обследование сельскохозяйственных угодий, расположенных на загрязненных территориях. В каждом хозяйстве имеются планы землепользования, на которых указаны плотности загрязнения всех полей, сенокосов, пастбищ, составлены радиологические паспорта, выведены из сельскохозяйственного оборота наиболее загрязненные земли.

С начала работ произвестковано 75 тыс. гектаров земли в общественном секторе и полностью выполнена эта работа на приусадебных участках. На площади 33 тыс. гектаров внесены повышенные дозы удобрений.

Выполненные мелиоративные мероприятия и применение высоких доз минеральных и органических удобрений, а также перепрофилирование хозяйств обеспечивает в настоящее время получение продукции растениеводства в пределах временных допустимых норм на землях с уровнями загрязнения до 40 Ки/км².

В течение 1986 г. вследствие значительного аэрального загрязнения кормовых культур и пастбищ в трех областях УССР радиоактивное загрязнение молока и мяса в 3-10 раз превышали допустимый уровень.

По рекомендациям ученых было принято решение о переработке молока, получаемого в загрязненных районах на масло, что позволило решить проблему его хозяйственного использования так как более 90% радионуклидов уходит с обратом. Разработаны экспресс-методики прижизненного определения содержания цезия в мясе животных и проведения их предубойного откорма на чистых кормах.

Указанные мероприятия позволили исключить непосредственную опасность здоровью сотен тысяч людей.

Одним из важных вопросов нашей деятельности являлось создание условий для возобновления работы атомной станции вахтовым методом, поскольку в энергосистемах республики ощущалась острая нехватка генерирующих мощностей.

В короткие сроки для персонала АЭС был сооружен поселок Зеленый Мы с полным комплексом коммунально-бытовых услуг, построена скоростная шоссейная дорога длиной 40 км и речной порт. Выполнение этих и других работ позволило уже в октябре-ноябре 1986 г. возобновить работу I и II, а в декабре 1987 г. III блоков АЭС.
РИС.4. Загрязненность территории УССР.
Параллельно для коллектива Чернобыльской АЭС и специалистов, работающих в зоне отселения, с участием строительных коллективов других республик построен современный город Славутич с необходимыми объектами социально-бытового назначения. В настоящее время здесь уже проживает более 10 тыс. человек. (проект рассчитан на 20 тыс. жителей, с перспективой развития до 30 тыс. человек.).

После Чернобыльской аварии прошло более трех лет, но последствия радиоактивного загрязнения природной среды на значительной территории республики продолжает оставаться острой социальной проблемой (рис. 4). Поэтому вопросы ликвидации последствий аварии постоянно находятся в центре внимания правительства республики и комиссии по чрезвычайным ситуациям.

На 1989–1990 гг. и на период до 2000 г. в республике разработан проект комплексной программы, основным содержанием которой является:

— обеспечение безопасности проживания населения в зонах так называемого "жесткого контроля";
— дальнейшее осуществление работ по дезактивации местности и проведению агромелиоративных мероприятий, направленных на уменьшение дозы облучения (внешней и внутренней) до установленных пределов;
— радиационный и дозиметрический контроль в опасных зонах;
— проведение и координация научно-исследовательских работ и других мер.

Ориентировочная стоимость всего комплекса работ, предусмотренных программой, составит около 11 млрд. рублей.

В развитие этой программы по инициативе Правительства республики Совет Министров СССР и Центральный Совет Профсоюзов 20 октября этого года приняли совместное постановление, которым предусмотрен ряд дополнительных мер по усилению охраны здоровья и улучшению материального положения населения, проживающего на загрязненных в результате аварии территориях.

В целом по ликвидации последствий аварии проведена огромная работа, но успокоения быть не должно, потому что потребуются еще долгие годы, прежде чем трагедия уйдет в прошлое. Это хорошо понимает Правительство республики и делает все от него зависящее, чтобы как можно быстрее и надежнее решить эту проблему.
L’ORGANISATION NATIONALE FRANÇAISE EN CAS D’URGENCE RADIOLOGIQUE

Y. MOURÈS
Comité interministériel de la sécurité nucléaire,
Paris, France

Abstract–Résumé

ORGANIZATIONAL ARRANGEMENTS TO DEAL WITH RADIOLOGICAL EMERGENCIES IN FRANCE.

The French nuclear safety regulations provide for measures to be taken in the event of an incident or accident which could entail a radiation risk, either at a nuclear facility or during the transport of nuclear material. The organization set up for this purpose must be able to ensure that the technical measures taken by the responsible authorities are effective in terms of nuclear safety and radiation protection as well as civil defence and public order. Specific emergency plans lay down measures in order to ensure the immediate protection of the population which must be taken in the area around a nuclear facility where an accident has occurred. The Interministerial Nuclear Safety Committee, under the authority of the Prime Minister, co-ordinates the actions taken by the public authorities. This organization has been improved since the Chernobyl accident. Among other things, it has proved necessary to prepare post-accident plans going well beyond the immediate emergency intervention programme — medium and long term plans for medical monitoring of the population and for rehabilitation of the environment. A draft post-accident plan prepared by the Ministry of the Interior is currently being studied at interministerial level and tested during radiological emergency exercises in which the operators and ministries concerned take part.

L’ORGANISATION NATIONALE FRANÇAISE EN CAS D’URGENCE RADIOLOGIQUE.

La réglementation française en matière de sûreté nucléaire prévoit les mesures à prendre en cas d’incident ou d’accident pouvant impliquer un risque radiologique dans une installation nucléaire ou pendant le transport de matières nucléaires. L’organisation mise en place dans ce but doit être à même de s’assurer de l’efficacité des mesures techniques prises par les autorités responsables tant de la sûreté nucléaire et de la protection radiologique que de la défense civile et de l’ordre public. Des plans particuliers d’intervention prévoient les mesures à prendre autour d’une installation nucléaire où se produit un accident afin d’assurer la protection immédiate de la population. Le Comité interministériel de la sécurité nucléaire, placé sous l’autorité du Premier ministre, coordonne l’action des pouvoirs publics. Cette organisation a été améliorée depuis l’accident de Tchernobyl. Il s’est avéré notamment nécessaire de préparer des plans post-accidentels concernant, au-delà de la période pendant laquelle s’applique un plan particulier d’intervention, les actions à moyen et long terme portant sur le contrôle médical des personnes et la restauration de l’environnement. Un projet de plan post-accidentel préparé par le Ministère de l’intérieur est actuellement à l’étude au niveau interministériel et est mis à l’épreuve au cours d’exercices d’urgence radiologique auxquels participent les exploitants et les ministères concernés.
En France, la sécurité nucléaire est harmonisée dans le cadre du Comité interministériel de la sécurité nucléaire (CISN). Le CISN est présidé par le Premier ministre et réunit tous les ministres qui ont une compétence en matière de sécurité nucléaire. Les problèmes relatifs à la sécurité nucléaire ne relèvent pas en effet d’un seul ministère et peuvent toucher tous les ministères concernés par un aspect de ce domaine complexe. Le Secrétariat général du CISN est constitué par une équipe placée auprès du Premier ministre et assure le suivi de l’action des différentes administrations et organismes pour veiller à leur harmonie et susciter éventuellement les initiatives nécessaires. Le CISN n’a, par contre, pas de rôle opérationnel ni de gestion, un tel rôle incombant aux services qui ont une responsabilité précise établie par les lois et règlements.

C’est dans ce cadre que se situe l’action conçue pour le rétablissement d’une situation normale après un accident majeur. Le retour à une situation normale en pareille circonstance implique, bien évidemment, la mise en œuvre de techniques de pointe, qui sont essentielles. Il n’en reste pas moins que, dans toute lutte, il ne suffit pas d’utiliser les moyens les plus complets de la technologie d’avant-garde. Il faut encore que l’action porte sur tous les aspects du problème, qu’elle soit coordonnée, qu’elle mobilise tous les moyens nécessaires et qu’elle se déroule en harmonie avec la population et ses représentants. Il faut donc un dispositif complet et efficace.

Pour comprendre l’ensemble de ce dispositif, il faut relever l’importance du problème pour la France. C’est dans notre pays, notamment avec Pierre Curie et son épouse Marie Sklodowska, qu’a été découvert le radium. Depuis plus d’un siècle, la radioactivité est un centre d’intérêt majeur tant au niveau de la recherche qu’à celui des applications médicales et industrielles. Dès que le nucléaire est entré dans l’ère industrielle, la France a eu pour souci de développer les technologies relevant de ce domaine et elle a lancé, dès la fin des années 50, les premières centrales expérimentales. C’est dans ce contexte qu’elle a été confrontée au premier choc pétrolier.

Comme tous les pays industrialisés, la France est un gros consommateur d’énergie. Ses besoins, même maîtrisés, augmentent constamment. Mais elle ne possède pratiquement pas de pétrole et son charbon, presque totalement épuisé, est d’un coût très élevé. Quant à son domaine hydraulique, qui est important, il a fait l’objet d’un aménagement très soutenu et précoce. Le résultat est que les sites économiquement aménageables sont presque tous utilisés depuis le début des années 70. Par ailleurs, la composition des échanges internationaux ne permet pas à la France de supporter une charge déficitaire trop importante en matière énergétique. Le choix a donc été fait de fonder le développement de l’énergie électrique sur l’équipement nucléaire.

Ce choix a été suivi d’effets. Actuellement, la France possède 52 500 MWe de production nucléaire installée, sur un parc thermique de 69 200 MW, avec 55 réacteurs répartis sur 18 sites. Elle occupe le deuxième rang mondial pour la production totale d’énergie nucléaire et le premier rang mondial pour la part du nucléaire dans la production d’électricité (70%).
Parallèlement, l’effort de maîtrise de la production nucléaire a conduit à s’assurer de la maîtrise complète du cycle du combustible, depuis le minerai jusqu’au retraitement du combustible et au stockage des déchets.

La sécurité nucléaire, corollaire de l’activité nucléaire, est donc en France une priorité nationale reconnue, dont les pouvoirs publics et les citoyens, conscients des risques encourus, tirent les conséquences. Ces risques, aussi importants qu’ils soient, ne sont d’ailleurs pas les seuls dans une économie moderne.


On se trouve donc en présence d’un exploitant industriel unique, EDF, doté de moyens puissants et d’équipes très diversifiées et capable de développer une technologie homogène facilitant l’étude des mesures de sûreté industrielle, l’exploitation du retour d’expérience et la mise en œuvre des mesures de sécurité.

D’une façon générale, la responsabilité de l’ensemble des mesures de sécurité concernant le territoire et la population est assumée par l’État, tout spécialement lorsqu’il s’agit de risques «technologiques ou industriels majeurs» pour lesquels, en matière de prévention et de traitement, sa compétence est établie par la loi. La très large décentralisation opérée depuis 1982 a conduit à associer plus étroitement les collectivités locales à l’action de l’État sans pour autant dessaisir celui-ci de ses responsabilités.

On se trouve donc en présence de deux autorités pleinement responsables, chacune dans son domaine, dans le cas d’un accident entraînant une situation de crise suivie de conséquences radiologiques sérieuses.

Le premier responsable est l’exploitant, et plus particulièrement le chef de l’installation nucléaire. Avec des soutiens techniques et scientifiques au niveau national, il a pour responsabilités:

— de tout mettre en œuvre pour que la menace décelée ne débouche pas sur un accident;
— de combattre les conséquences de cet accident, s’il vient à se produire, dans l’enceinte de la centrale, ce pourquoi il doit appliquer un plan d’urgence interne à l’installation (PUI);
— d’informer les populations;
— d’informer le représentant de l’État des conséquences prévisibles à l’extérieur du site et de le mettre en mesure de préparer les actions nécessaires pour faire face aux débordements éventuels de l’accident hors du domaine de la centrale.
En effet, dès que les conséquences, notamment les rejets de matières radioactives, ont franchi l’enceinte de l’installation nucléaire, la responsabilité de toutes les mesures incombe au représentant local de l’État, le préfet, compétent dans le département où se trouve la centrale électrique en difficulté.

Pour faire face à l’accident, les mesures à prendre s’inscrivent dans un plan systématique, préparé à l’avance, qui envisage les parades et les moyens à employer et, d’une façon générale, qui prévoit les éléments de la décision incombant au préfet.

Deux plans sont appliqués successivement.

Le plan particulier d'intervention (PPI) est celui de la première phase d'urgence, c'est-à-dire du moment où l'accident vient de se produire et où l'on doit prendre d'urgence un certain nombre de mesures de protection alors même que l'on ne possède sur la situation que des données qui peuvent être fragmentaires ou incertaines. Cette phase est dite «phase réflexe». Le plan vise essentiellement les secours et les précautions à prendre pour limiter les effets.

Le plan post-accidentel (PPA) se substitue progressivement au PPI; il intéresse la période suivante où l'on connaît les résultats constatés de l'accident, où l'on peut réunir des éléments d'information fiables et où l'on doit envisager les mesures de longue échéance pour organiser la vie et, si possible, la réhabilitation des régions qui ont été touchées.

La description de ces deux plans fait l'objet d'un mémoire publié dans les présents comptes rendus.

Ces plans ne sont pas des supputations imaginées pour les seuls accidents nucléaires.

En effet, depuis une trentaine d'années, la France s’est progressivement dotée de façon systématique d’un schéma général pour toutes les situations d'urgence. Il a été testé dans un certain nombre de circonstances pour faire face à des catastrophes. Ce sont les plans ORSEC dont sont tirés les plans plus spécialisés comme ceux qui nous occupent ici.

Le principe sur lequel reposent ces plans est que la responsabilité de la mise en œuvre des moyens appartient à l’autorité localement compétente sur le lieu du sinistre, c'est-à-dire dans l’organisation française, le préfet du département.

Mais si cette autorité a été choisie parce qu’elle est la plus proche du problème et parce qu’elle est de ce fait mieux à même d’apprécier, elle ne dispose pas moins de tous les éléments que l’appareil d’État a aménagés sur l’ensemble du territoire national pour faire face à une crise.

En ce qui concerne le suivi de l’incident, des cellules de crise gouvernementales se mettent en place immédiatement pour une veille permanente:

1) Auprès du Premier ministre, la cellule de crise est animée par le Secrétariat général du CISN. Parmi ses tâches figure celle de recueillir l’ensemble des éléments d’information qui doivent être transmis aux instances internationales pour remplir les engagements de la France, notamment à l’égard de l’AIEA.
2) Auprès du Ministre de l'industrie, le Service centrale de sûreté des installations nucléaires (SCSIN) analyse techniquement les évaluations, les prévisions et les mesures à prendre, est en consultation permanente avec l'exploitant et le représentant local de l'État et dispose du soutien de l'Institut de protection et de sûreté nucléaire qui, fonctionnant dans le cadre du CEA, a précisément pour mission de poursuivre des recherches et des analyses approfondies dans ce domaine.

3) Auprès du Ministre de l'intérieur, la cellule de crise coordonne la mise en place et l'organisation des secours, notamment à travers le Centre opérationnel de la Direction de la sécurité civile. Ce département ministériel a également la charge de la lutte contre les actes de malveillance.

4) Auprès du Ministre de la santé publique, la cellule de crise est animée par le Service central de protection contre les rayonnements ionisants (SCPRI) qui est saisi de toutes les questions ayant trait à la santé publique et à la radiologie.

5) Une cellule de crise peut être constituée auprès d'autres ministres, parmi lesquels le Ministre des affaires étrangères si l'accident est susceptible d'avoir des répercussions en dehors du territoire national. C'est en effet à ce département ministériel qu'il appartient de transmettre à l'AIEA, à EURATOM et aux États les éléments d'information recueillis par la cellule de crise du Premier ministre.

Ces cellules ont un rôle d'expertise, de conseil et de coordination des moyens dont les autorités pourraient avoir besoin pour faire face à la situation. Certains moyens d'intervention existent en permanence dans les départements. Il existe également des moyens nationaux, trop importants pour pouvoir être démultipliés en autant d'unités qu'il y a de sites mais qui peuvent être mis à la disposition du site concerné dans de très brefs délais. C'est le cas notamment des équipes du SCPRI, des cellules mobiles d'intervention radiologique des zones de défense et des équipes mobiles issues des établissements du CEA.

Sur le plan opérationnel, l'ensemble des services d'incendie et de secours répartis sur tout le territoire national peuvent être détachés dans de très brefs délais, sur décision du Ministre de l'intérieur, afin de renforcer des unités locales.

Parmi les autres services dont l'intervention serait requise, citons la Météorologie nationale dont le Service central d'exploitation et les services locaux devraient transmettre à tous les intervenants les éléments météorologiques nécessaires à la prévision du transport atmosphérique.

Ce dispositif doit concilier deux impératifs: maintenir la prise de décisions le plus près possible du lieu de l'accident et être capable, en permanence, de mobiliser l'ensemble des moyens dont la France dispose. Il est le fruit de l'expérience acquise en matière de lutte contre les catastrophes.

En effet, dans le cas particulier du nucléaire, et fort heureusement, aucune application de ce dispositif n'a été faite qui permette d'en tester la fiabilité dans une situation d'urgence réelle. On ne saurait cependant se satisfaire indéfiniment d'un
travail d'état-major conçu dans des bureaux. C'est pourquoi, pour tester les plans d'intervention de façon aussi réaliste que possible sur le terrain, on a généralisé la procédure des exercices qui est employée également dans d'autres domaines.

Ces exercices, dont la conception varie avec les installations concernées, sont organisés par les organismes responsables. C'est ainsi, par exemple, qu'EDF en organise dans chacune de ses centrales. Pour toute installation nucléaire donnant lieu à l'établissement d'un PPI, les autorités locales responsables, c'est-à-dire les préfets, sont amenées à organiser des exercices mettant en œuvre les moyens de leur département.

Enfin, au plan national, le Secrétariat général du CISN organise régulièrement des exercices impliquant tous les acteurs: ministères, exploitants, services locaux de l'Etat, autorités locales, population. Ces exercices sont planifiés dans le temps en tenant compte d'autres exercices relatifs à des risques de nature différente. Leur ampleur même contraint à ne pas les multiplier, afin de ne pas gêner le travail quotidien de ceux qui ont à intervenir et qui sont ceux-là mêmes qui ont la responsabilité de la vie collective de la population. Hors du site d'origine, la sécurité nucléaire ne dépend pas en effet des seules autorités spécialisées mais relève de l'ensemble des services publics.

Un exercice complet suppose la préparation d'un scénario et d'une animation complexes visant à amener les services administratifs et techniques ainsi que l'autorité investie du pouvoir de décision à prendre les mesures qui s'imposent en cas d'accident réel.

Les exercices mettent également à l'épreuve un certain nombre de moyens concernant, notamment, les services de secours, les transmissions, les services sanitaires etc. Ces services sont destinés à manœuvrer réellement sur le terrain.

Cet entraînement poursuit plusieurs objectifs dont les principaux peuvent être ainsi résumés:

— contribuer à la formation des acteurs, à tous les niveaux, qui seraient amenés à participer à la mise en œuvre des plans;
— tester les procédures de répartition de responsabilités, d'information réciproque et de coordination pour apprécier leur efficacité dans toute la mesure possible;
— faire apparaître, en créant une situation de crise, les problèmes et les difficultés qui n'auraient pas été pris en compte dans l'élaboration du document.

Cette pratique a jusqu'à présent fait progresser assez largement l'examen critique des mesures à prendre.

Dans les exercices eux-mêmes, tout comme dans les plans, les problèmes d'information des autorités locales et des populations par les voies nécessaires et notamment par les médias prennent une place grandissante; car on ne doit pas se dissimuler que, si les mesures techniquement les plus adaptées ne sont pas annoncées aux populations et à leurs représentants et expliquées attentivement, la mise en œuvre des dispositions les plus techniquement fiables et sûres demeure aléatoire et leurs résultats
incertains. On ne doit pas oublier que, dans la réalité, les plans de lutte ne seraient pas mis en œuvre par les seuls services administratifs mais aussi par tous les acteurs de la vie publique, sociale et économique. Toute l'action menée devra donc être expliquée et représenter une synthèse des exigences techniques et des réactions humaines possibles.

L'ensemble de ces dispositions s'insère dans le cadre des échanges internationaux. La France ne prétend pas régler seule tous les problèmes auxquels elle pourrait être confrontée. De même, elle est prête à tout moment à apporter son concours pour répondre à ses engagements internationaux ou à une demande expresse qui lui serait adressée à l'occasion d'une crise.
EFFECT OF ADMINISTERING STABLE IODINE TO THE WARSAW POPULATION TO REDUCE THYROID CONTENT OF IODINE-131 AFTER THE CHERNOBYL ACCIDENT

P. KRAJEWSKI
Central Laboratory for Radiological Protection,
Warsaw,
Poland

Abstract

EFFECT OF ADMINISTERING STABLE IODINE TO THE WARSAW POPULATION TO REDUCE THYROID CONTENT OF IODINE-131 AFTER THE CHERNOBYL ACCIDENT.

An assessment of the effectiveness of administering stable iodine to reduce $^{131}$I doses to the thyroid has been performed. The five compartment model of iodine metabolism developed by Johnson was used to evaluate predicted levels of $^{131}$I content in the thyroid and integrated activity for different doses of stable iodine and various age and sex groups within the population. A computer program, MODELTHR, was designed to perform the calculations. The data obtained have been compared with results of measurements of $^{131}$I in the thyroid of Warsaw residents. The measured values of $^{131}$I concentration in air and in milk and standard values for milk and food consumption and inhalation rate as well as metabolic parameters were used. The theoretical calculation showed that the administration of stable iodine on 1986-04-28 could reduce the committed dose to the thyroid from inhalation by about 80% and from inhalation and ingestion by 28% whereas the administration of stable iodine in the two following days 1986-04-30 and 1986-05-01 could reduce the committed dose to the thyroid from the inhalation pathway by about 35% and 7% and from the inhalation and ingestion pathway of about 25% and 10% respectively.

1. INTRODUCTION

Exposure to radioactive isotopes of iodine, in particular $^{131}$I, and the resulting irradiation of the thyroid presents possibly the most serious radiological risk from accidental release of fission products. The early hazard would partially arise from the inhalation and rapid remedial action would be desirable to minimize the uptake to the thyroid. The longer term hazard from ingestion of contaminated milk and foodstuffs is due to the fact that $^{131}$I deposits on the vegetation and thus may be transferred to humans through the pasture-grazing-animal-milk pathway.
Because of its rapid uptake by the thyroid, $^{131}$I delivers a relatively high dose to this organ in a short period of time.

The reactor accident at the Chernobyl power plant Unit 4 in the USSR released a plume of radioactive materials which reached the northeastern part of Poland sometime during the night of 1986-04-27 [1]. The highest air concentration of $^{131}$I in Warsaw lasted from 29 to 30 April, reaching a value of about 200 Bq/m$^3$, later decreasing to 2 Bq/m$^3$, rising again from 7 to 9 May up to around 10 Bq/m$^3$ than dropping after 10 May to below any measurable level (Fig. 1) [2].

Similarly, the highest concentration of $^{131}$I in Warsaw milk (about 500 Bq/L) occurred during the period 29-30 April, then decreased (to about 100 Bq/L), increased again on 12 May (to about 450 Bq/L) and then decreased exponentially with a half-life of approximately 8 days (Fig. 2).

Also, in some cases, elevated concentrations of $^{131}$I in milk occurred in the northeastern region of Poland, reaching values up to 3 kBq/L [1]. This level was higher than the intervention level (1 kBq/L) recommended by the Polish National Atomic Energy Agency. The high concentration in air and milk and lack of information concerning further releases at the Chernobyl reactor suggested the possibility of...
a significant radiological risk of radioiodine intake for infants and children. Therefore the Government Commission for Assessment of Irradiation and Resources decided on 1986-04-29 to distribute over eleven districts prophylactic doses of stable iodine, the dose being: 15 mg for infants younger than 1 year, 30 mg for children 1–10 years old and 60 mg for young people up to 16 years old. Additionally diet restriction and limitations on cows grazing were introduced [3]. Also, a number of adults voluntarily took prophylactic doses of stable iodine.

The use of stable iodine to reduce or prevent the uptake of radioiodine is widespread in diagnostic medicine and has already been the subject of many papers [4, 5]. In a prolonged incident it might become necessary to distribute repeated doses of stable iodine to people at risk.

In Poland, this method was utilized for the general public for the first time. Most of the controversy concerning the use of stable iodine has arisen primarily from the lack of verified information on adverse reactions to a single high dose of KI. The toxicity of stable iodine in such small doses is low, although some authors report that taking 130 mg KI as a single dose or as 10 daily doses could cause 0 to 1 life-threatening events per million doses [4].

**FIG. 2.** Iodine-131 concentration in milk measured in Warsaw.
In addition, the benefits of a single dose of stable iodine seem to be uncertain because chronic (repeated uptake of radioiodine), rather than acute exposure (single intake) occurred in Poland due to the contamination being spread over time.

For this reason the present study focuses on an assessment of effectiveness of administering stable iodine to reduce $^{131}$I doses to the thyroid.

2. THE METABOLIC MODEL FOR IODINE IN MAN

Risk assessment in the specific situation described above requires a model of the biological kinetic of iodine which includes all age groups (especially children and infants). This model should include the effects of stable iodine intakes on radioiodine uptake by the thyroid and the recovery of the thyroid from its blocked state.

The non-recycling model for iodine metabolism recommended by the ICRP [6] for calculating annual limits on intake contains no information on either the uptake of iodine by the gland in children or its turnover. Therefore it is of limited use for population risk assessment.

The Stather-Greenhalgh [7] dosimetric model was developed for different age groups and provided a basis for calculating doses to the UK population after the release of radioactive isotopes of iodine into the environment. However, this model does not permit calculation of the reduction in the committed thyroid dose following oral administration of stable iodine. In addition, these dosimetric models consider only a less complicated pattern of intake, e.g. only acute or chronic exposure with constant airborne iodine concentrations. So it could not be applied for the more complicated pattern of intakes which occurred in Poland.

For this reason the five compartment model of biological kinetic of iodine developed by Johnson [8] seems to be the most appropriate for our assessments.

The model was adopted with the following assumptions:

(a) the rate of stable iodine uptake by the thyroid is directly proportional to the mass of the subject and is independent of the daily intake and of the blocking dose of stable iodine

(b) the rate constant for iodine leaving the organic compartment is $0.0535 \text{ d}^{-1}$

(c) the rate constant for excretion of iodine from the inorganic compartment does not vary with the subject's age, weight or dietary intake of iodine and is $1.92 \text{ d}^{-1}$

(d) the amount of iodine in the thyroid is equal to the product of thyroid mass and iodine concentrations

(e) the rate constant for iodine leaving the thyroid compartment is $s_2/M_t$, where $M_t$ is the mass of total iodine in the thyroid.

The biological kinetics of iodine is determined by a set of four differential equations describing its behaviour in the following compartments: gut or lung, inor-
### TABLE I. INHALATION RATES FOR VARIOUS AGE GROUPS [10]

<table>
<thead>
<tr>
<th>Age group</th>
<th>Inhalation rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Adult male</td>
<td>0.98 m(^3)/h</td>
</tr>
<tr>
<td>Adult female</td>
<td>0.88 m(^3)/h</td>
</tr>
<tr>
<td>10 a</td>
<td>0.616 m(^3)/h</td>
</tr>
<tr>
<td>5 a</td>
<td>0.4 m(^3)/h</td>
</tr>
<tr>
<td>1 a</td>
<td>0.158 m(^3)/h</td>
</tr>
</tbody>
</table>

### TABLE II. MILK AND FOOD CONSUMPTION FOR VARIOUS AGE GROUPS [7]

<table>
<thead>
<tr>
<th>Age group</th>
<th>Intake (kg/d)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0-1 a</td>
<td>0.64</td>
</tr>
<tr>
<td>2-6 a</td>
<td>0.64</td>
</tr>
<tr>
<td>7-15 a</td>
<td>0.74</td>
</tr>
<tr>
<td>Adult</td>
<td>0.90</td>
</tr>
</tbody>
</table>

### TABLE III. FACTORS \(S_{ij}\) TO CONVERT INTEGRATED ACTIVITY (Bq·d) IN THE THYROID TO THE COMMITTED THYROID DOSE (\(\mu\)Sv/Bq·d)

<table>
<thead>
<tr>
<th>Age</th>
<th>0</th>
<th>1</th>
<th>2</th>
<th>5</th>
<th>10</th>
<th>15</th>
<th>Adult female</th>
<th>Adult male</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thyroid</td>
<td>1.63</td>
<td>1.29</td>
<td>1.04</td>
<td>0.627</td>
<td>0.359</td>
<td>0.238</td>
<td>0.170</td>
<td>0.142</td>
</tr>
<tr>
<td>Blood and tissues</td>
<td>(6.408 \times 10^{-5})</td>
<td>(7.978 \times 10^{-3})</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
TABLE IV. MODEL EQUILIBRIUM VALUES FOR STABLE IODINE AS A FUNCTION OF AGE

<table>
<thead>
<tr>
<th>Age (a)</th>
<th>0</th>
<th>1</th>
<th>5</th>
<th>10</th>
<th>Ref. female</th>
<th>Ref. male</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass of body (kg)</td>
<td>3.51</td>
<td>7.2</td>
<td>22.0</td>
<td>40.5</td>
<td>58.9</td>
<td>70.0</td>
</tr>
<tr>
<td>Mass of thyroid (g)</td>
<td>1.63</td>
<td>2.12</td>
<td>4.39</td>
<td>7.87</td>
<td>17</td>
<td>20</td>
</tr>
<tr>
<td>Daily iodine intake (µg/d)</td>
<td>10.0</td>
<td>20.6</td>
<td>62.8</td>
<td>116</td>
<td>166</td>
<td>200</td>
</tr>
<tr>
<td>Mass of stable iodine (µg) in compartment:</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>a) Inorganic</td>
<td>5.1</td>
<td>10</td>
<td>32</td>
<td>59</td>
<td>84</td>
<td>100</td>
</tr>
<tr>
<td>b) Thyroid</td>
<td>300</td>
<td>300</td>
<td>990</td>
<td>3700</td>
<td>10 000</td>
<td>12 000</td>
</tr>
<tr>
<td>c) Organic</td>
<td>56</td>
<td>120</td>
<td>350</td>
<td>650</td>
<td>930</td>
<td>1100</td>
</tr>
<tr>
<td>Iodine taken up by thyroid per day (µg)</td>
<td>3.25</td>
<td>6.68</td>
<td>20.4</td>
<td>37.6</td>
<td>53.9</td>
<td>65</td>
</tr>
<tr>
<td>Biological half-life of release of iodine from the thyroid (d)</td>
<td>64</td>
<td>31</td>
<td>33.6</td>
<td>68.2</td>
<td>128.6</td>
<td>128</td>
</tr>
</tbody>
</table>

ganic, organic, thyroid. The kinetics of radioactive iodine is described by five differential equations related to the above compartments and additionally an equation describing bladder contents. They form a set of coupled linear first order equations, with weak coupling between stable iodine and radioactive iodine concentration by the rate of uptake of radioactive iodine by the thyroid. This uptake is determined by the ratio of the concentration of radioactive and stable iodine in the inorganic compartment.

\[ r_2 = s_2 \left( \frac{Y^r_2}{Y^s_2} \right) \]

where \( r_2 \) is the thyroid uptake rate of radioiodine, \( s_2 \) is the thyroid uptake rate of stable iodine, \( Y^r_2 \) is the concentration of radioactive iodine in the inorganic compartment, and \( Y^s_2 \) is the concentration of stable iodine in the inorganic compartment.

These equations can be solved numerically with satisfactory precision by a Runge–Kutta algorithm of order four [9]. The MODELTHR program was designed to perform the calculation on an IBM AT PC. The program evaluates predicted \(^{131}\text{I} \) contents in the thyroid as a function of time for different doses of stable iodine and
various age groups and calculates integrated activity. As the program uses the integrated package Symphony 2.1, graphic presentation and printout of the results is easy.

3. COMPUTER SIMULATION

Using MODELTHR the computer calculations of the $^{131}$I thyroid contents and committed doses for the population of Warsaw were evaluated. The program took into account measured values of $^{131}$I activity concentrations in air in Warsaw and milk (Figs 1, 2). Standard values for inhalation rates, food consumption and also $S_{ii}$ conversion factors were used (Tables I—III). Calculations were performed for four age groups: children 1, 5 and 10 years old and for standard man. Metabolic parameters used in the calculations are presented in Table IV. The concentration of stable iodine in the gut, inorganic, organic and thyroid compartment plotted against time for a standard man are shown in Fig. 3. It can be seen from Fig. 3 that the

![Graph](image)

**FIG. 3.** Concentration of stable iodine in particular compartments as a function of time for standard man (blocking dose was taken on 1986-04-30).
FIG. 4. Predicted $^{131}$I activity in particular compartments for standard man (inhabitant of Warsaw) from inhalation pathway (60 mg of stable iodine was taken on 1986-04-30).

FIG. 5. Predicted $^{131}$I activity in particular compartments for standard man (inhabitant of Warsaw) from inhalation with ingestion pathway.
<table>
<thead>
<tr>
<th>Age</th>
<th>Day of blocking</th>
<th>Dose of stable iodine</th>
<th>Integrated activity</th>
<th>Inhalation reduction (Bq · d)</th>
<th>Committed absorbed dose (mSv)</th>
<th>Ingestion with inhalation</th>
<th>Committed absorbed dose (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Adult</td>
<td>—</td>
<td>0 mg</td>
<td>12 925</td>
<td>0%</td>
<td>1.8</td>
<td>38 165</td>
<td>0%</td>
</tr>
<tr>
<td></td>
<td>1986-05-01</td>
<td>60 mg</td>
<td>12 108</td>
<td>6%</td>
<td>1.7</td>
<td>34 948</td>
<td>10%</td>
</tr>
<tr>
<td></td>
<td>1986-04-30</td>
<td>60 mg</td>
<td>8 523</td>
<td>35%</td>
<td>1.2</td>
<td>28 629</td>
<td>25%</td>
</tr>
<tr>
<td></td>
<td>1986-04-28</td>
<td>60 mg</td>
<td>2 607</td>
<td>80%</td>
<td>0.4</td>
<td>27 478</td>
<td>28%</td>
</tr>
<tr>
<td>10 a</td>
<td>—</td>
<td>0 mg</td>
<td>7 715</td>
<td>0%</td>
<td>2.8</td>
<td>—</td>
<td>—</td>
</tr>
<tr>
<td></td>
<td>1986-04-28</td>
<td>60 mg</td>
<td>1 374</td>
<td>80%</td>
<td>0.5</td>
<td>—</td>
<td>—</td>
</tr>
<tr>
<td>5 a</td>
<td>—</td>
<td>0 mg</td>
<td>4 543</td>
<td>0%</td>
<td>2.8</td>
<td>16 470</td>
<td>0%</td>
</tr>
<tr>
<td></td>
<td>1986-05-01</td>
<td>30 mg</td>
<td>4 247</td>
<td>7%</td>
<td>2.7</td>
<td>14 823</td>
<td>10%</td>
</tr>
<tr>
<td></td>
<td>1986-04-30</td>
<td>30 mg</td>
<td>2 965</td>
<td>35%</td>
<td>1.9</td>
<td>12 380</td>
<td>25%</td>
</tr>
<tr>
<td></td>
<td>1986-04-28</td>
<td>30 mg</td>
<td>833</td>
<td>80%</td>
<td>0.5</td>
<td>11 860</td>
<td>28%</td>
</tr>
<tr>
<td>1 a</td>
<td>—</td>
<td>0 mg</td>
<td>1 803</td>
<td>0%</td>
<td>2.3</td>
<td>13 500</td>
<td>0%</td>
</tr>
<tr>
<td></td>
<td>1986-05-01</td>
<td>15 mg</td>
<td>1 682</td>
<td>7%</td>
<td>2.1</td>
<td>12 150</td>
<td>10%</td>
</tr>
<tr>
<td></td>
<td>1986-04-30</td>
<td>15 mg</td>
<td>1 177</td>
<td>35%</td>
<td>1.5</td>
<td>10 131</td>
<td>25%</td>
</tr>
<tr>
<td></td>
<td>1986-04-28</td>
<td>15 mg</td>
<td>291</td>
<td>80%</td>
<td>0.4</td>
<td>9724</td>
<td>28%</td>
</tr>
<tr>
<td></td>
<td>1986-04-30</td>
<td>60 mg</td>
<td>1 140</td>
<td>3%*</td>
<td>1.5</td>
<td>—</td>
<td>—</td>
</tr>
</tbody>
</table>


The concentration of stable iodine in the inorganic compartment after just a few minutes reaches a maximal value approximately equal to the dose of stable iodine taken, then decreases exponentially with a half-time of 6 hours. After about five days it reaches a settled level before thyroid blocking. This level depends on the daily intake of stable iodine and the metabolic parameters of the subject.

As the rate of uptake of radioactive iodine is reciprocal to the concentration of stable iodine in the inorganic compartment, five days after the thyroid blocking dose has been taken, reduction of the thyroid uptake of radioactive iodine is equal to zero. Consequently there is no thyroid blocking after this period.
Figure 4 gives an example of the activity of $^{131}$I in particular organs for a Warsaw inhabitant (standard man) as predicted by the model, calculated for intakes through inhalation. Figure 5 presents similar calculations for intakes through inhalation and ingestion. In both cases the recommended stable iodine dose of 60 mg was taken on 1986-04-30. The radioiodine concentrations in the lung and gut, inorganic, organic and thyroid compartments are plotted against time and reflect the different pattern of intake depending on $^{131}$I contamination in air and in milk (Fig. 1, 2). The $^{131}$I thyroid contents calculated for inhalation and ingestion is a factor of two higher than the contents calculated for inhalation only. Table V gives data calculated for integrated activities and committed doses depending on the way internal contamination arose and the day of thyroid blocking.

The comparison of values in Table V shows that the administration of stable iodine on 1986-04-28 could reduce the thyroid dose due to inhalation by 80%, whereas the administration of stable iodine two days later (on 1986-04-30) produced reductions of 35% and three days later (1986-05-01) of 7%.

For the ingestion and inhalation pathway the reduction factors are 28%, 25% and 10% respectively. To explain this it can be noted that the high concentration of $^{131}$I in air lasted a relatively short time compared with $^{131}$I in milk and other food products. The uptake of radioiodine by thyroid was effectively blocked only for a short period of time and the thyroid dose from further intakes was not reduced. Therefore the pathway of ingestion made a much higher contribution to the thyroid doses than the pathway of inhalation.

However a fourfold increase of the prophylactic dose of stable iodine (e.g. up to 60 mg) for a child one year old causes a reduction of only 3%, of integrated activity in relation to an administered dose of 15 mg.

4. THE COMPARISON OF PREDICTED LEVELS OF $^{131}$I THYROID CONTENT WITH MEASUREMENT DATA

Theoretical calculations were compared to results of measuring the $^{131}$I content of the thyroid of 578 inhabitants of Warsaw. These measurements were undertaken by the Central Laboratory for Radiological Protection from 1986-04-05 to 1986-06-27 (Figs 6–8). Three statistical groups were taken into account: men, women, and children (1–15 a). Each group was divided into subgroups characterized by the date of administration of stable iodine, e.g. 1986-04-29, 1986-04-30, 1986-05-01 and a subgroup that did not take stable iodine. The parameters of the exponential function $F(t) = A \cdot \exp(-\ln2/T_{1/2} \cdot t)$ were fitted to the data of each subgroup using the least squares method. Given the most complicated pattern of intakes, it is only valid to assume a simple exponential decrease of radioiodine in the thyroid with time in cases of acute exposure or after decline of the contamination.
FIG. 6. The measured $^{131}$I thyroid contents for men (inhabitants of Warsaw) plotted together with predicted levels of $^{131}$I in thyroid (curve 1 — pathway of inhalation with ingestion for date of blocking 1986-04-30; curve 2 — pathway of inhalation for date of thyroid blocking 1986-04-30; curve 3 — pathway of inhalation for date of blocking 1986-04-29).

FIG. 7. The measured $^{131}$I thyroid contents for women (inhabitants of Warsaw) plotted together with predicted levels of $^{131}$I in thyroid (curve numbers as in Fig. 6).
However, in our opinion, the calculated values of constant A and $T_{131}$ could reflect the effects of stable iodine on reducing $^{131}$I uptake in the thyroid assuming similar conditions for each statistical subgroup, e.g. the same diet, air contamination, metabolic parameters and habitation. Table VI gives the results of the calculations.

The remarkable differences in $T_{131}$ of $^{131}$I in the thyroid that were obtained probably reflect the accidental ingestion of $^{131}$I by some members of particular subgroups. Consumption of contaminated food products, especially contaminated milk, dairy products and vegetables, quite often occurred despite the ban. The values of A constant calculated for the subgroups which blocked the thyroid on 29 April and for subgroups which did not use stable iodine differ by a factor of about 2 for men and children and by a factor of about 5 for women. The results do not confirm the effectiveness of administering prophylactic doses of stable iodine but rather indicate a significant statistical scatter. More detailed regression analysis of the data is

\[ \text{FIG. 8. The measured } ^{131}\text{I thyroid contents for children 1-15 a (inhabitants of Warsaw) plotted together with predicted levels of } ^{131}\text{I in thyroid (curve 1 — pathway of inhalation for child 1 year old at the date of thyroid blocking, 1986-04-30; curve 2 — pathway of inhalation with ingestion for child 5 years old at the date of blocking, 1986-04-30; curve 3 — pathway of inhalation for child 5 years old at the date of blocking, 1986-04-30; curve 4 — pathway of inhalation with ingestion for teenager 15 years old at the date of blocking, 1986-04-30).} \]
difficult. The size of some of the subgroups does not permit valid subdivisions according to the above criteria. In addition, there was a lack of credible information about these conditions. To recapitulate, data derived from the administration of stable iodine in Poland is not statistically significant.

To verify this conclusion another analysis was performed. The theoretically evaluated levels of $^{131}$I thyroid content were plotted together with the data of measurements (Figs 6–8). The levels were calculated, using MODELTHR, for male and female inhabitants of Warsaw, taking into account the date of thyroid blocking and the pathway of intakes.

Figures 6 and 7 show the measured $^{131}$I thyroid contents for men and women respectively with predicted levels of $^{131}$I in the thyroid as follows:

- curve 1 — pathway of inhalation with ingestion with thyroid blocking on 1986-04-30
- curve 2 — pathway of inhalation with thyroid blocking on 1986-04-30
- curve 3 — pathway of inhalation with thyroid blocking on 1986-04-29.

Figure 8 shows the measured $^{131}$I thyroid contents for children, considering different age groups (marked by letters) with predicted levels of $^{131}$I as follows:

- curve 1 — pathway of inhalation of a one-year-old with thyroid blocking on 1986-04-30
- curve 2 — pathway of inhalation with ingestion for a five-year-old with thyroid blocking on 1986-04-30
- curve 3 — pathway of inhalation for a five-year-old with thyroid blocking on 1986-04-30
- curve 4 — pathway of inhalation with ingestion for a teenager 15 years old with thyroid blocking on 1986-04-30.

**TABLE VI. PARAMETERS OF REGRESSION ANALYSIS OF THE THYROID I-131 CONTENTS FOR WARSAW INHABITANTS**

<table>
<thead>
<tr>
<th>Statistical Date of administration of prophylactic dose of stable iodine</th>
<th>1986-04-29</th>
<th>1986-04-30</th>
<th>1986-05-01</th>
<th>No blocking</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>A</td>
<td>$T_{1/2}$</td>
<td>N</td>
<td>A</td>
</tr>
<tr>
<td>Men</td>
<td>706</td>
<td>31</td>
<td>34</td>
<td>655</td>
</tr>
<tr>
<td>Women</td>
<td>208</td>
<td>108</td>
<td>58</td>
<td>1154</td>
</tr>
<tr>
<td>Children 1-15 a</td>
<td>689</td>
<td>32</td>
<td>40</td>
<td>986</td>
</tr>
</tbody>
</table>
Comparing the results given in Figs 6 and 7 it may be noted that most of the measured values cover the ranges determined by the curves calculated for the pathway of inhalation with ingestion — curve 1 (upper limit) and for inhalation — curve 2 (lower limit). About 20% of the measured data are below curve 2 and above curve 3. Curve 3 determines the level of $^{131}$I contents in thyroid calculated for administration of stable iodine on 1986-04-29. Although there are some persons whose $^{131}$I thyroid content fit the criteria (between curves 2 and 3) and who took stable iodine on 29 April, no correlation was found because there are also some persons who did not take the stable iodine and whose $^{131}$I thyroid content is found in this range.

Figure 8 gives examples of the group of children and teenagers at ages ranging from 1 year to 15 years. Because in this age range the metabolic parameters change dramatically, the scatter of the results of measurements is greater compared with the group of adults. The measured values of $^{131}$I thyroid content for children under 5 years (marked by letter a) lie between curves 1 and 3. These curves determine the predicted levels of $^{131}$I thyroid content calculated for a one-year-old and a five-year-old respectively for the pathway of inhalation. It is reasonable because most of the children between one and five were restrained from drinking fresh milk and consuming other fresh food products. It is not clear why some of the measured values (mainly for children aged five to ten) lie above the upper limit of the predicted level of $^{131}$I content in the thyroid of a teenager for the inhalation with ingestion pathway. Probably these children consumed food products delivered from locally more contaminated areas.

In conclusion, the theoretical calculations and analysis of the measurements presented in Figs 6–8 confirm the assumption that the contribution to the $^{131}$I in the thyroid from the intakes of $^{131}$I through consumption of contaminated food is much higher than inhalation. As appeared from interviews of the people monitored, most of them took a prophylactic dose of stable iodine on 30 April and did not refrain from consuming contaminated food, including milk (with the exception of infants). For those who took a prophylactic dose on 30 April the reduction of integrated activity of $^{131}$I in the thyroid after taking stable iodine is only 25%. In the authors’ opinion regression analysis of the measured data would not furnish any proof.

5. CONCLUSIONS

Theoretical calculations of $^{131}$I thyroid content performed on the basis of the $^{131}$I concentration in air and milk in Warsaw showed that administration of stable iodine on 1986-04-28, 1986-04-30 and 1986-05-01 could have reduced the committed dose to the thyroid by 28%, 25% and 10%, respectively.

The ingestion pathway was responsible for about 70% of the total dose for adults and teenagers and over 80% for one-year-olds.
Data of measured $^{131}$I contents in the thyroid of Warsaw inhabitants demonstrate significant scatter, probably caused by consumption of contaminated foodstuffs. Thus, the 25% reduction of integrated activity of $^{131}$I in the thyroid attributed to a single dose of stable iodine could not be proved using regression analysis.

In a case of prolonged contamination, administering a single dose of stable iodine seems to be less effective than diet restriction.

ACKNOWLEDGEMENTS

This work is in part supported by grant MZ XVII from the Polish Ministry of Health and Social Welfare.

REFERENCES


PLANNING AND EMERGENCY MEASURES ESTABLISHED IN TURKEY AFTER THE CHERNOBYL ACCIDENT

Ö.A. SOYBERK
Çekmece Nuclear Research and Training Centre,
Istanbul,
Turkey

Abstract

PLANNING AND EMERGENCY MEASURES ESTABLISHED IN TURKEY AFTER THE CHERNOBYL ACCIDENT.

Considerable knowledge was gained from the Chernobyl accident with regard to techniques for protecting the public. In order to mitigate the effects of any undue radiation hazard, monitoring teams continued to operate for a long time during the recovery phase. Critical areas to be surveyed were those of higher fallout, particularly in Thrace and the eastern Black Sea coast of Turkey. Contaminated agricultural products such as vegetables, tea, meat, fish, milk and dairy products were surveyed by measuring the activity of a large number of samples. Contaminated tea and nuts were a significant problem in Turkey. Some protective measures have been taken during the recovery phase in the Edirne province of Thrace. The dose estimates were complicated due to local or regional variations in agricultural practices. Since recovery may take months to years, establishment of intervention levels is very important and should have appropriate accuracy. The Emergency Committee for Radiation Security of Turkey (ECRST) has established intervention levels for tea and other foodstuffs, adopting the values already issued by international bodies. There are already a nuclear plant operating and two nuclear research centres in Turkey. The Turkish Atomic Energy Authority (TAEA) has decided to set up a general off-site emergency planning and preparedness programme for radiological accidents, whether industrial or medical. The paper describes the emergency planning and necessary measures in the event of an accident in Turkey.

1. INTRODUCTION

The severe nuclear power plant accident at Chernobyl had serious radiological effects on the unprotected population. Various protective actions were taken to alleviate these radiological effects. Accident consequences to the general public following the release from Chernobyl were evaluated. In order to protect the public from undue radiation hazards, monitoring teams continued to operate for a long time during the recovery phase.

The Turkish Atomic Energy Authority (TAEA) took the necessary actions continuously from the beginning of the accident. The radiation exposure rate from the
background cumulative doses, contamination of the air and ground were measured and evaluated. A large number of samples of contaminated agricultural products such as vegetables, tea, nuts, fish, milk and dairy products were monitored. Contaminated tea and nuts were a significant problem in Turkey. Tea, which is consumed in large amounts, was the main contributor to the dose of the public.

Some protective measures were taken by the Emergency Committee for Radiation Security of Turkey (ECRST) during the recovery phase in the Edirne province of Thrace. The countermeasures in Turkey were principally aimed at restricting public exposures from consumption of foodstuffs (such as milk, meat, vegetables, fruit, etc.) or from the environment, assessing the stochastic effects of doses to the population and evaluating the cost-benefit of protective measures.

In order to evaluate the individual and collective doses, data such as weighted averages of yearly gamma dose rate, ground deposition levels and food activity concentrations have been collected every year since Chernobyl.

To minimize the radiation effects the ECRST has established derived intervention levels for some foodstuffs, including tea, by adopting the values already issued by international bodies.

Because of the Chernobyl accident, nuclear plants operating around the Black Sea and some accidents in industrial and medical fields in the past, TAEA decided to set up a general off-site emergency planning and preparedness plan in Turkey, covering both nuclear and non-nuclear applications of radioactivity.

This paper describes in detail radiological consequences, monitoring, individual and collective doses and intervention levels during the recovery phase following the Chernobyl accident, in conjunction with the above mentioned planning.

2. PLANNING AND MONITORING RELATED TO EMERGENCY ACTIONS

A large amount of radioactive material was released and almost all Europe as well as Turkey was contaminated by the Chernobyl accident. In order to minimize the radiological consequences to the population living in Turkey, emergency actions were taken by the TAEA. Emergency monitoring teams began to work immediately according to the emergency plan.

Before the accident at Chernobyl all matters related to radiation protection were under the jurisdiction of the TAEA. However at the end of May 1986 a decision was taken by the Council of Ministers to establish an Emergency Committee for Radiation Security of Turkey, the ECRST, under the chairmanship of the Minister of Industry and Commerce. This committee consisted of twenty members, including the president of the TAEA and his deputy and the representatives of related departments of some Ministries.
In Turkey, there are two nuclear research and training centres, one in Istanbul and one in Ankara, each with TAEA operating environmental monitoring stations. These stations have been provided with the necessary equipment, including high pressure ion chambers, low background beta counters, gamma spectrometers with high purity Ge and NaI detectors and whole body counters.

Immediately after the Chernobyl accident the mobile monitoring teams were organized and started to operate in addition to the environmental monitoring stations. In line with the emergency planning, the teams consisted of a radiation protection expert and a monitoring technician, and were equipped with portable radiation monitors capable of measuring low radiation levels and sampling devices to collect air, water and soil samples.

Initially they tried to detect any abnormal conditions and assess the magnitude of contamination. As a second step they monitored large areas from Thrace to the eastern part of Turkey to help decide on remedial actions to prevent the harmful effects of radiation to the public. The third step of monitoring was the same as in the second step, and was carried out after the remedial actions had been taken, i.e. in the recovery phase.

Since the consequences of a nuclear accident extend beyond national boundaries the measurements were important for convincing the public about their safety despite the accident having happened very far from Turkey. As the country has a nuclear power reactor in operation, continuous monitoring of the environment and relevant foodstuffs is a major concern of nuclear safety.

3. RECOVERY MONITORING

3.1. Environmental monitoring

In order to protect the public from any undue radiation hazard, monitoring teams continued to operate for a long period during the recovery phase. Radiation rates measured 1 m from the ground, cumulative doses and radioactivity on the ground were monitored continuously. The following measurements were made in this step: (a) beta and gamma dose rates in air; (b) air contamination; (c) ground deposition; (d) food control (milk and dairy products, vegetables, meat and fish).

Critical areas to be surveyed were those which had higher fallout, particularly in Thrace and on the eastern Black Sea coast of Turkey. Radiation exposure rates from the background, cumulative doses and contamination of the air and ground were measured and evaluated. Contaminated agricultural products such as vegetables, tea, meat, nuts, fish, milk and yoghurt were surveyed by measuring the activity levels of a large number of samples.
3.1.1. Air gamma dose rate

The first effect of the Chernobyl accident observed was the increase in the gamma radiation levels in the northwestern region (Thrace) and the Black Sea coast of Turkey. The environmental gamma dose rate levels in these regions increased from normal values of 8–10 μR/h to 30–50 μR/h on 4–5 May 1986. From June 1986 onward levels decreased slowly to the normal values. The gamma dose rates measured in 1987 were between 29 μR/h and 50 μR/h 1 m above the ground level of tea plantations. At ground level the figures varied from 20 μR/h to 41 μR/h.

3.1.2. Airborne radioactivity

Airborne samples were collected 1 m above the ground in various areas by using portable air samplers and were measured with low background beta counters. The airborne samples evaluated showed considerable variation with respect to region and time. The air activity concentration increased up to 120 Bq/m$^3$ and then decreased quickly to 1.2 Bq/m$^3$ on 7 May and to normal values after that date.

The significant radionuclides observed by spectrometric analyses in the airborne samples were: $^{131}$I, $^{132}$Te, $^{134}$Cs, $^{137}$Cs, $^{140}$La, $^{140}$Ba, $^{103}$Ru, $^{106}$Ru, $^{141}$Ce, $^{144}$Ce.

3.1.3. Rain and drinking water activity

The activity of the rain water collected on 4 May was found to be 8911 Bq/L in Kapikule-Edirne and 3503 Bq/L on 10 May in Istanbul.

Spectral analysis of the rain water collected showed that approximately 59% of the activity measured on 4 May was due to $^{131}$I and on 10 May 40% was from $^{103}$Ru. The rain water activity concentration decreased quickly to normal values in the following months.

Although the concentrations in the rain water at Edirne were found to be highest at the beginning of May, the underground waters were not affected by radioactive contamination.

The activity of drinking water was analysed continuously starting from 1 May. The highest radioactivity level, 26 Bq/L, was found on 5 May 1986. Levels then decreased to normal values during the course of the year.

The activity of the radionuclides in various barrage waters changed during the year as follows: $^{131}$I: 15–19 Bq/L, $^{137}$Cs: 4–6 Bq/L, $^{134}$Cs: 2–3 Bq/L.

3.1.4. Ground deposition and soil activity concentration

The deposition on the ground was, for total radiiodine, 694 TBq and for $^{137}$Cs plus $^{134}$Cs, 63 TBq during the first year. The national average was 890 Bq/m$^2$ for iodine and 80 Bq/m$^2$ for caesium.
The activity deposition on the soil samples increased in the Edirne region and the Black Sea coast during the first year. The highest gross beta activity was 2915 Bq/kg in Edirne and 1070 Bq/kg in Istanbul in May 1986. The survey of the Radionuclide dispersion in the soil showed the significant contaminant was caesium.

The gross beta activity of the soil samples taken in 1987 from depths of 0–20 cm and 20–40 cm varied from 5 Bq/kg to 400 Bq/kg and 10 Bq/kg to 420 Bq/kg, respectively.

Unfortunately the measured data are problematic in nature since information about the location and type of soil samples is not available. It is often unknown whether the samples were taken from a ploughed field or from unploughed natural soil.

3.2. Foodstuffs monitoring

3.2.1. Milk and dairy products

Milk samples from selected farms were collected and monitored continuously from the beginning of the accident. Goat’s, cow’s and sheep’s milk and fresh cheeses were also monitored. The highest activity was found in the milk samples collected from Kapikule and Eskikadin village region (20 000 Bq/L) near Edirne. The average milk activity concentration in the Edirne region was determined to be only 520 Bq/L. Milk activity decreased from May to June 1986 to 1000 Bq/L for milk, 500 Bq/L for $^{137}\text{Cs}$ and 200 Bq/L for $^{134}\text{Cs}$.

Table I shows the radionuclide concentration in milk from 1986–05–04 to 1986–05–16. It decreased to normal values during the course of the year.

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>$^{131}\text{I}$</th>
<th>$^{137}\text{Cs}$</th>
<th>$^{134}\text{Cs}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Date</td>
<td>1986–05–04</td>
<td>1986–05–16</td>
<td>1986–05–16</td>
</tr>
<tr>
<td>I-131</td>
<td>21 000</td>
<td>1 000</td>
<td></td>
</tr>
<tr>
<td>Cs-137</td>
<td>800</td>
<td>500</td>
<td></td>
</tr>
<tr>
<td>Cs-134</td>
<td>500</td>
<td>200</td>
<td></td>
</tr>
</tbody>
</table>
3.2.2. Meat and fish radioactivity

The two most important radionuclides were $^{137}\text{Cs}$ and $^{134}\text{Cs}$, deposited primarily in muscular tissues. The periodically measured activity in 1987 in sheep, cattle and lamb meat varied from 5 Bq/kg to 350 Bq/kg according to the region. The $^{137}\text{Cs}$ concentration in the sheep and cattle meat was 69 Bq/kg and 7 Bq/kg respectively in February 1987. The highest activity measured in beef and mutton was 1000 Bq/kg in Thrace, but in other areas the average activity did not exceed 80 Bq/kg during 1986. Usually mutton showed a higher activity than beef.

Starting from 5 May, fish and mussels caught in the Black Sea were systematically analysed. Measured values were surprisingly low. The highest activity in mussels was 100 Bq/kg at the beginning of May 1986; it dropped quickly in October 1986 and no activity was found in 1987.

3.2.3. Tea and nuts

The first, second and third shoots of the tea bushes planted in Giresun, Trabzon and Rize, which are located on the eastern Black Sea Coast were collected and analysed in 1986. The highest radioactivity was found in the first shoots and the lowest concentration was in the third. The significant radionuclides were $^{137}\text{Cs}$ and $^{134}\text{Cs}$.

The activity of the dry teas produced in 1986 is shown in Table II.

The average concentration in the first shoots of the dried tea product varied between 500 Bq/kg and 1500 Bq/kg for $^{137}\text{Cs}$, and 200 Bq/kg and 600 Bq/kg for $^{134}\text{Cs}$ in 1987.

The highest activity levels in hazel nuts were found in the eastern Black Sea region (Rize, Findikli — 12800 Bq/kg) and the lowest level in the western Black Sea region (Bolu, Akçakoca — 150 Bq/kg).

<table>
<thead>
<tr>
<th>Shoots</th>
<th>Highest (Bq/kg)</th>
<th>Lowest (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>First</td>
<td>31 000</td>
<td>14 000</td>
</tr>
<tr>
<td>Second</td>
<td>13 000</td>
<td>7 500</td>
</tr>
<tr>
<td>Third</td>
<td>8 500</td>
<td>2 000</td>
</tr>
</tbody>
</table>
4. PROTECTIVE MEASURES

Some protective measures were taken by the ECRST during the recovery phase in Edirne, eastern Thrace. Countermeasures in Turkey were principally aimed at restricting public exposures arising from the environment and from the consumption of foodstuffs such as milk and meat; assessing the stochastic effects of doses to the population and evaluating the cost-benefit of protective measures. The following protective measures, decisions and recommendations were implemented by the ECRST:

— Restrictions were put on the food consumption of people in the Edirne province.
— Outdoor grazing of dairy cattle was prohibited; in a small district of Edirne they were kept indoors and fed on stored fodder.
— Production and distribution of milk in some regions of Thrace were prohibited. People were warned about drinking milk or consuming dairy products.
— In some cases contaminated milk taken from cattle was converted into cheese and butter and then stored until the short lived radionuclides had sufficiently decayed.
— Highly contaminated areas around the border station of Kapikule in Edirne were decontaminated and contaminated soils collected.
— Highly contaminated dried tea products with activities of more than 10 000 Bq/kg were collected and stored as radioactive waste.

5. INTERVENTION LEVELS

Since recovery may take from months to years, the establishment of the intervention levels was very important: these should have appropriate accuracy. As is well known, in this phase some risk may still arise from consumption of contaminated foodstuffs and direct radiation of the contaminated areas. The ECRST has established derived intervention levels for some foodstuffs, including tea, by adapting those already issued by international bodies. Relatively low values were selected for radiocaesium in milk, 370 Bq/L, and 600 Bq/L for other foodstuffs. Highly contaminated tea and other products above the intervention levels were stored as low level waste or destroyed. For establishing the contamination levels of the areas involved and determining the intervention level, practices recommended by the international bodies are closely followed in Turkey in order to diminish the prolonged effects of the contamination.
### TABLE III. INDIVIDUAL AND COLLECTIVE DOSES RECEIVED BY CRITICAL GROUPS AND THE GENERAL PUBLIC IN TURKEY DURING THE FIRST YEAR AFTER THE CHERNOBYL ACCIDENT (mSv)

<table>
<thead>
<tr>
<th>Exposure pathway</th>
<th>Critical groups</th>
<th>General public</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Infants</td>
<td>Adults</td>
</tr>
<tr>
<td>Clouds</td>
<td>(0.26 \times 10^{-2})</td>
<td>(0.26 \times 10^{-2})</td>
</tr>
<tr>
<td>Ground</td>
<td>(2.64 \times 10^{-2})</td>
<td>(2.64 \times 10^{-2})</td>
</tr>
<tr>
<td>Inhalation</td>
<td>(20.33 \times 10^{-2})</td>
<td>(19.74 \times 10^{-2})</td>
</tr>
<tr>
<td>Milk and dairy</td>
<td>(9.57 \times 10^{-2})</td>
<td>(6.84 \times 10^{-2})</td>
</tr>
<tr>
<td>products</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Vegetables</td>
<td>(0.63 \times 10^{-2})</td>
<td>(5.92 \times 10^{-2})</td>
</tr>
<tr>
<td>and fruits</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bread, etc.</td>
<td>(0.03 \times 10^{-2})</td>
<td>(1.92 \times 10^{-2})</td>
</tr>
<tr>
<td>Meat</td>
<td>(0.09 \times 10^{-2})</td>
<td>(1.99 \times 10^{-2})</td>
</tr>
<tr>
<td>Tea</td>
<td>—</td>
<td>(20.10 \times 10^{-2})</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td>(0.350)</td>
<td>(0.594)</td>
</tr>
</tbody>
</table>

### 6. INDIVIDUAL AND COLLECTIVE DOSES

In order to evaluate the individual and collective doses, data such as weighted averages of yearly gamma dose rates, ground deposition levels and food activity concentrations have been collected in the years since the Chernobyl accident. Data on the eating habits and daily consumption of food by various age groups were obtained from statistics published by various institutes. By considering dose conversion factors and filtering effects of buildings, the effective doses from ingestion, inhalation and dose rates from ground and clouds were estimated for individuals and critical groups as well as for the public. However, the dose estimates were complicated by local or regional variations in agricultural practices. Different values for release rates from ground or air are obtained in the growing season or harvest time. In order to obtain an accurate assessment of potential doses due to ingestion, appropriate recommendations are needed for regional and local foodstuffs.
Individual effective committed dose equivalents have been calculated for the critical group, chosen as 100,000 inhabitants, and for the general public in Turkey. Individual doses of critical groups have been estimated as 0.350 mSv for children 0-1 years old and 0.594 mSv for adults; individual doses of the general public have been estimated as 0.147 mSv for adults and 0.500 mSv for children 0-1 years old.

Table II shows the individual and collective doses received by critical groups and the general public of Turkey during the first year after the accident (May 1986-April 1987).

7. CONCLUSION

Contaminated tea and nuts were significant problems in Turkey. Tea, which is consumed in large amounts, was the main contributor to the dose of the public. Highly contaminated tea above derived intervention levels is being temporarily stored in the provinces of the northeast Black Sea region. However, difficult problems exist as regards permanent storage. The people and the local authorities are not willing to co-operate in solving the problem because of the difficulties in communication between the common people and scientists regarding the effects of radiation and protective measures.

In addition to this, certain protective countermeasures such as prohibiting outdoor grazing and keeping cattle indoors, where intervention levels were stated, were often misconstrued by the public.

The confusion concerning the doses received by the public, speeches of politicians on scientific matters and news and articles in the press almost demoralized the public.

The accident in Chernobyl and some other incidents involving industrial and medical applications of radioactive sources have provided a valuable experience in Turkey. There are already several nuclear plants around the Black Sea. Turkey and her neighbours have plans to build nuclear plants in the near future in their territories. Therefore, the TAEA decided to set up a general off-site emergency plan in Turkey to deal with nuclear, industrial or medical radiological accidents. In order to optimize the protection of the public this plan includes: (a) organization of responsibilities; (b) implementation of emergency planning and recovery actions; (c) assessment of the emergency organization required; (d) establishment of an early warning system.

We are considering using the existing computer facilities at Çekmece Nuclear Research and Training Centre for data storage and dose calculations during the recovery phases as well as in other phases of emergency conditions.
BIBLIOGRAPHY


TURKISH ATOMIC ENERGY AUTHORITY, Consequences of the Measurements of Radiation and Radioactivity after Chernobyl in Turkey, TAEC (1988) (in Turkish).

Invited Paper

CLEANUP OF AREAS CONTAMINATED AS A RESULT OF A NUCLEAR ACCIDENT

M.A. FERADAY
M.A. Feraday & Associates Ltd,
Toronto, Canada

Abstract

CLEANUP OF AREAS CONTAMINATED AS A RESULT OF A NUCLEAR ACCIDENT.

In spite of all precautions, the possibility of an accident at a nuclear power plant which would result in the release of unacceptable amounts of radioactive material and cause serious contamination of surrounding areas cannot be excluded. One protective measure which may be required during the intermediate and late phases of an accident is the cleanup of contaminated areas. The term cleanup includes decontamination, stabilization or isolation of the contamination, along with the transport and disposal of the wastes arising from cleanup. If the Emergency Director decides to implement cleanup after a serious accident, all reasonable means should be used to minimize the huge costs and the detriment to humans. To ensure that the cleanup can be quickly and efficiently carried out requires good preliminary planning, clear strategies, a good managerial team, well trained workers and suitable equipment. To assist Member States in the development of their emergency preparedness plans for cleanup, the IAEA is preparing publications which provide an integrated overview of the overall operational planning for cleanup, methods and equipment to carry out these actions and the means to transport safely and dispose of the large volumes of waste. The paper provides an overview of the information in those reports.

1. INTRODUCTION

The development of nuclear power has in general been associated with an excellent record of safety. Nuclear facilities are sited, designed, constructed, and operated according to strict regulations to protect the workers and the public. In spite of these precautions, the possibility of an accident which would result in the release of unacceptable amounts of radioactive material and cause serious contamination of surrounding areas cannot be excluded. Therefore it is desirable to plan in advance the measures which are required to protect both the public and the facility staff.

The International Atomic Energy Agency (IAEA) and the International Commission on Radiological Protection (ICRP) have published general guidance and recommendations on emergency planning and preparedness for situations where an
accident at a nuclear power plant may involve the need for off-site remedial actions and implementation of protective measures [1-4]. One protective measure which could be implemented in the intermediate phase (days to weeks after the accident) and the late phase (weeks to years) is the cleanup of areas.

The term cleanup includes decontamination, stabilization or isolation of contamination and the transport and disposal of wastes arising from cleanup. Decontamination is the removal of radioactive contaminants with the objective of reducing the residual radioactivity level in or on materials, persons or the environment. Decontamination can occur as a result of actions by humans or as a result of natural processes such as precipitation. Stabilization of the radioactivity means fixing it so that it no longer is a detriment to the environment. Radioactivity can be isolated by covering the materials affected with a layer of clean material such as concrete or clean soil, or by deep ploughing to remove the contamination from the upper layer of soil. Also, there may be subregions within the affected area where alternatives to cleanup, such as ‘do nothing’ or interdiction of the area, would be preferable.

In normal situations, before the planned cleanup of nuclear facilities or sites commences, detailed plans would have been made and approved by regulatory authorities, teams of trained people and special equipment gathered and the sites and contamination well characterized.

In contrast, in the event of a serious accident at a nuclear facility which results in widespread contamination of the environment, the uncertainties are much greater and the action plans cannot be as well defined as for planned events [5, 6]. For example, contaminated areas could include hundreds of square kilometres, the affected area could be highly populated and contain many buildings, and skilled workers and suitable cleanup/monitoring equipment might not be readily available. These and many other uncontrollable factors complicate the planning and implementation of the cleanup after a serious accident at a nuclear facility.

If the Emergency Director decides to implement cleanup, all reasonable means should then be used to minimize costs and the detriment to humans of such a cleanup. To ensure that cleanup can be quickly and efficiently implemented requires good planning, clear strategies, a good managerial team, well trained workers and suitable equipment and techniques. To assist Member States in the development of their emergency preparedness plans for cleanup, the IAEA is preparing three reports which provide an integrated overview of the overall operational planning, implementation and management aspects of such large-scale operations [5]; methods and equipment available to carry out these actions [6]; and the means to transport and dispose of large volumes of waste [7]. This paper provides an overview of the information in those three reports.

For this paper it is assumed that the Emergency Director has declared that the area is a restricted zone not suitable for public use, that the public has been evacuated from the affected area and that measures must be taken to avoid the spread of the contamination and/or reduce doses to acceptable levels.
2. PLANNING, IMPLEMENTING AND MANAGING THE CLEANUP

2.1. Responsibilities

The actions taken off the site after an accident, including the cleanup of contaminated areas, are the responsibility of the public authorities [3]. They should designate an Emergency Director who has the political, administrative and legal power to control all personnel directly associated with the response to the emergency, including the cleanup team; and to implement such protective measures as might be deemed necessary.

If large scale cleanup is required, a separate Cleanup Director should be appointed. The Cleanup Director and his staff would, under the supervision of the Emergency Director, define the broad strategy for cleanup and clearly outline key decisions required e.g. first actions, critical facilities in the area, division of zones, etc. To implement cleanup successfully, teams for each zone must be aware of the equipment and methods which have been effective in the past and are readily available (Section 6). They must then decide which are most applicable for their particular situation.

If the cleanup continues for many months or years, the Public Authorities Emergency Organization [3] would probably revert to its normal stand-by status and some of its responsibilities could be transferred to the cleanup organization.

2.2. Planning the cleanup

To minimize the potential detriment to the environment of a nuclear accident which results in serious contamination of large areas, national and local governments should be encouraged to do emergency cleanup planning for each major nuclear facility. For example, a programme in France [8] called RESSAC (rehabilitation of soils and subsurfaces, following an accident) is being used to determine land use patterns around the 21 nuclear sites in France and develop the planning and techniques required for cleanup after a serious accident.

The degree of preliminary planning should be in relation to the probability of occurrence of such an accident and the potential detriment to the public if it does happen. Since the probability of occurrence of a serious nuclear accident such as Chernobyl is expected to be very low, it is unlikely that local governments around nuclear facilities will be willing to spend large amounts of money in preparing detailed preliminary plans for the cleanup of large areas. However, since the potential consequences of such an accident are very serious, especially in an urban area, some emergency preparedness planning for cleanup should be done.

Ideally, preparations for cleanup should be divided into two phases: preliminary planning which is done as part of normal emergency preparedness for each major nuclear facility and the final detailed planning and assessment which would
be initiated at the onset of the accident and would take into account site specific and accident specific information. These plans are complementary and both should contain an overall operational plan and technical information on suitable equipment and procedures required to carry out the cleanup safely, efficiently and with minimum costs.

2.2.1. Preliminary planning

A preliminary cleanup plan will contain generic information which is applicable to all similar facilities and site specific information which is applicable to a particular facility [5-7].

Examples of the generic information that would be relevant to all or many facilities in a country include:

(a) Technical information on: characterizing (Section 4) and stabilizing (Section 5) the contamination; cleanup equipment and techniques (Section 6); and means of disposing of the radioactive wastes (Section 8).

(b) Criteria: three basic criteria are required for the successful implementation and completion of a cleanup campaign i.e. derived intervention levels (DILs), cleanup criteria and final release criteria. The decision to implement the cleanup of a contaminated area is made, to a large extent, on the basis of the DIL [4] for this protective measure. Once the decision to implement such a cleanup has been made, then cleanup criteria must be available to define the specific radionuclide concentration or gamma exposure levels which must be achieved by the workers doing the remedial action. In addition, final release criteria must be available for the release of all or part of the area for unrestricted or restricted use to allow the return of the population, reuse of the land for agriculture, etc. The development of these criteria is the responsibility of the competent national authorities. However, the IAEA and other organizations give guidance on setting such criteria [4, 9].

(c) Details of the subplans which are required to carry out a large scale radiological cleanup (see Section 2.2.4).

(d) An assessment of equipment, facilities and other resources which are readily available locally, nationally or internationally and the contact person.

In addition, site and/or facility specific information should be available in the preliminary plan for the area surrounding a particular facility. This information should include:

— source term: the radionuclides and source term which would be released to the environment and the physical/chemical form of such releases can be estimated for various accident scenarios.

— characteristics of the affected area (Section 3).
— a radiological survey plan: Fig. 1 shows the basic elements of such a plan and some protocols required for implementation.

— a geographical co-ordinate grid system for the areas around each facility is required so that the location of each sampling or data point is known precisely and unambiguously.

— a data management system (Section 2.2.4).

The cost of preliminary planning for cleanup should not be large since much of the data required will already be available and only need to be collected and correlated. Also it should be possible to develop generic information which is applicable to many nuclear facilities on a national or international level.

Table I gives examples of the kind of information which might be collected during preliminary planning.

2.2.2. Final planning

If an incident at a nuclear facility should result in the release of significant amounts of contamination and it is decided that cleanup is required, the preliminary plan should be reviewed and updated in the light of new information. The resources needed for the cleanup would have to be assembled.

2.2.3. Operational planning

Preliminary and final plans should both contain an operational plan which shows how subplans (Section 2.2.4) interface and interact with each other and with other emergency programmes [5]. The operational plan would outline the objectives, broad strategic and tactical approach to cleanup, management structure (Section 2.3) and other key requirements to ensure that cleanup can be performed safely, efficiently and as quickly as possible under adverse conditions.

2.2.4. Important subplans

To implement and complete cleanup effectively requires that important operational details be clearly and unambiguously defined in subplans which should be formulated during preliminary planning. Since the information in many of the subplans is generic in nature, they could be developed by national authorities or international bodies and adapted to local situations. This section briefly reviews some important subplans which are discussed in more detail in Refs [5, 6]:

(a) Radiological survey plan: this plan has two basic components. As part of normal emergency preparedness, the concentration of natural and man-made radionuclides in the area around a nuclear facility should be mapped. Also, post-accident radiological surveys should be planned. Figure 1 shows how the
FIG. 1. Details of the radiological survey plan and grid layout.

Output to database

Radiological survey plan

Emergency preparedness:
Determine the background concentration of natural and man-made radionuclides in the designated areas

Post-accident radiological surveys

Comprehensive radiological characterization of the contamination

Control monitoring for excavation and other remedial action

General radiological survey to verify that areas can be released for reuse

Post cleanup radiological survey

Geographical co-ordinate system for the area

(1) Select permanent benchmarks — national grid
(2) Decide grid layout and modes for urban, rural, forest, etc. areas
(3) Survey in sample grid using professional surveyors

Laboratory support for environmental sampling

Field sampling of air, soil, water, vegetation, building material, etc.

Radiological monitoring surveys

(a) Aerial surveys
(b) Ground-level surveys for lands and buildings
(c) Subsurface monitoring

Protocols required

- Instrument — sensitivities — types
- Calibration procedures
- Sample control and data recording
- Analytical procedures
- QA requirements
- Sample treatment procedures

Protocols required

- Sampling procedures
- Statistical sample planning for each zone
- Type and number of samples for each zone
- Data recording procedures in the field
- QA procedures
- Sample shipping

Protocols required

- Instrument — sensitivities — types
- Calibration procedures
- Survey methodology
- QA requirements
- Verification procedures
Examples of the type of information which should be gathered during preliminary planning include [6]:

(a) demographic data: population size/distribution, etc.
(b) layout of cities and towns, including identification of areas/facilities requiring priority cleanup
(c) land classes; land uses; etc.
(d) geology and hydrogeology of the area: soil types; permeability; groundwater depth and direction; etc.
(e) accident scenarios and possible release source terms
(f) geographical co-ordinate grid system for the area
(g) background monitoring data base for the area
(h) sampling and monitoring plans for different zones
(i) location of critical rivers/lakes which could be susceptible to inflow of large volumes of contaminated water from the accident or cleanup
(j) the location of downstream drinking water supplies
(k) actions which may have to be taken shortly after the accident, e.g. immobilizing the contaminants; the installation of in-ground bypasses to reduce groundwater flow through the contaminated area; construction of diversion ditches or dams to prevent contaminated water from reaching clean water systems
(l) criteria required for the cleanup and other regulatory and radiation protection information
(m) management structure for cleanup
(n) analysis of potential cleanup options for each area based on an assessment of soil type, land use, available equipment, etc.
(o) a list of facilities and equipment required for cleanup and the location of available items, including potential disposal sites
(p) personnel requirements for various scenarios including a list and telephone numbers of key staff members and details of the notification system
(q) an outline of quality control requirements
(r) cost/risk analysis
(s) international contacts on various aspects of cleanup

Elements of the radiological survey plan interact with each other and how this plan interfaces with the grid and database.

Data management: a massive amount of basic data and calculated results will be generated during cleanup. Although some information will arise during preliminary planning (Table I), most will be generated as a result of the cleanup actions after the accident. Well organized databases and a database management system are required so that managers and operators can use this
FIG. 2. Management structure for cleanup of contaminated areas.
diverse and multidisciplinary information effectively [5, 10]. The system should be designed and made operational during preliminary planning so as to avoid the extreme pressures inherent in an actual radiological emergency.

(c) Quality assurance plan: all cleanup activities must be covered by suitable quality assurance (QA) programmes [5, 11].

(d) Compliance with criteria: just as important as the cleanup and release criteria are the protocols required to ensure compliance with these criteria [5].

(e) Radiation protection and safety of the workers: as part of normal emergency response planning, a radiation protection and safety plan should be formulated. If a serious accident should occur, this plan should be tailored to meet specific accident situations [5, 6].

2.3. Management structure for cleanup

The management organization for cleanup would consist of the Cleanup Director assisted by advisers and teams for such things as: zonal control, logistic support, quality assurance and control of common facilities and functions. Figure 2 outlines one possible management structure for cleanup showing the interaction between elements of the cleanup organization and the interface with the overall emergency plan. The organization shown is only for illustrative purposes. The final structure and the number and makeup of field teams could vary considerably depending on factors such as: site specific and accident specific conditions, scale of the accident, availability of trained staff and equipment, local infrastructure, urgency of cleanup, etc.

2.4. Cleanup strategies and tactics

The Cleanup Director and his advisers, under the supervision of the Emergency Director, would define the overall cleanup strategy as well as the strategies and tactics related to zones and generic types of areas [5, 6].

In defining the overall strategy, the Cleanup Director must consider a wide variety of factors and determine:

(a) actions to be taken as soon as the accident is brought under control: e.g. diversion of surface streams and groundwater streams around the damaged facility; construction of ditches to direct badly contaminated water to holding areas.

(b) which areas are to be cleaned up or interdicted.

(c) if cleanup should be started immediately or delayed to take advantage of decay and natural decontamination effects.

(d) which off-site facilities should get cleanup priority.

(e) which criteria should be used.

(f) facilities needed to implement cleanup, e.g. decontamination and disposal facilities, analytical laboratories, laundries, etc.
Within the overall cleanup strategy, the Zone Manager must determine the best strategy to control and clean the zone which is his responsibility. The strategy, composition of the cleanup teams and types of equipment would be specifically designed for various zones.

3. CHARACTERIZING THE AFFECTED AREA

During emergency preparedness, areas around major nuclear facilities should be characterized to obtain the basic data required to enable cleanup teams to understand the area and its interrelation with surrounding or downstream areas. These data, which would include many of the items shown in Table I, would assist teams to select the best cleanup methods for individual zones, potential disposal sites and transportation routes.

Sources of information include siting, environmental assessment and licensing documents for major facilities, e.g. power plants, dams, municipal waste disposal areas; land use forecasts; agricultural, forestry and mining industry data; demographic forecasts; aerial photographs; maps etc. In many countries, much of this information is readily available but it needs to be collected and collated by those responsible for planning the cleanup.

4. CHARACTERIZING THE CONTAMINATION

Before cleanup can be initiated, the type, mix, concentration and spatial distribution of radionuclides released as a result of the accident must be determined. Characterization efforts should aim to provide information to determine which, if any, cleanup actions should be initiated in various zones.

The techniques available to characterize the distribution and inventory of released material include: airborne and vehicle-borne monitors, semiportable and hand-held instruments, air and soil (surface and subsurface cores) sampling, etc. [6, 12, 13].

Remote gamma sensing from aircraft is an effective way of rapidly locating, monitoring and mapping gamma activity [12]. Helicopters and fixed-wing aircraft can be used as platforms for sensitive NaI or germanium detectors to measure the total gamma count rate. Helicopters are used for low level work where maximum sensitivity is required. The aircraft is positioned during surveys using microwave locating systems which provide a signal to indicators to guide the pilot accurately along preselected routes. Gamma signal, flight path, altitude and meteorological data are fed into an on-board data acquisition system for post-flight analysis.

For more detailed analysis of the contamination, especially during cleanup, vehicle-borne and hand-held alpha, beta and gamma detectors are available [6, 13].
Information on the type of remotely operated monitoring devices used at Chernobyl is given in Appendix A of Ref. [6].

In addition to understanding where contamination is located, cleanup teams must have a good understanding of its chemical and physical forms, the conditions under which the contamination was deposited and the physical and chemical state of the surface, since these factors will probably have an important bearing on the potential for subsequent decontamination.

5. STABILIZING THE CONTAMINATION

Following a nuclear accident which results in widespread contamination, the detriment to humans from the radioactive contaminants can be reduced by the decontamination methods described in Section 6, by interdiction of the contaminated area (Section 7) or by using coatings to stabilize the contamination.

The objectives of using coatings on soils, buildings, roads and equipment are to reduce the spread of contamination, airborne inhalation hazards and the volume of waste generated; to decontaminate surfaces by incorporating the contamination into a removable coating; and to reinforce the upper layer of soil.

In many cases, the application of stabilizers is a short term corrective action which would be followed by further remedial actions. A large number of stabilizers are commercially available [6, 14] and these are generally classified as chemical, mechanical, physical or chemical with mechanical characteristics. Chemical stabilizers are additives which alter the physical properties of the treated surface. Mechanical stabilizers include concrete and asphalt covers, polyvinyl films, sandbags, etc.

Following the Chernobyl accident, rapid polymerizing solutions were sprayed from a helicopter onto the site, the roof of the turbine buildings and sides of roads to stabilize the contamination, reinforce the upper layers of the soil and prevent the spread of dust.

6. CLEANUP METHODS AND EQUIPMENT

Decontamination of materials, equipment, buildings and sites to permit operation, maintenance and decommissioning to be done safely has been an integral function of the nuclear industry since its inception. A wide variety of chemical and mechanical decontamination techniques and equipment has been developed over the last 30 years to assist in removing contamination from all kinds of surfaces [15, 16].
In selecting the most effective method, cleanup teams must take account of many factors, such as:

- material type: metal, asphalt, concrete, soil
- surface finish: rough, porous, coated, smooth
- chemical and physical form of the contaminant
- weather conditions during deposition
- the proven efficiency of the process
- availability of equipment and trained staff
- the need to condition the secondary wastes generated
- occupational doses resulting from the process.

In the following sections, methods available to decontaminate buildings, equipment, paved surfaces and land areas are examined. These basic components will apply to both urban and rural areas.

6.1. Decontamination of buildings, equipment and pavement

Although there exists a great deal of experience in the decontamination of nuclear facilities, less attention has been paid to the development of methods suitable for large scale application to the wide variety of construction materials in urban areas. Many of the techniques suitable for nuclear plants may be too expensive for application on the scale required in an urban environment or too aggressive to be acceptable. Recently, more work has been done to help understand factors which are important in achieving good decontamination of urban surfaces [6, 17, 18].

It is beyond the scope of this paper to give details of the various kinds of equipment used for the cleanup of buildings, equipment and paved surfaces. However a list of the more common techniques and equipment is given in Table II.

6.2. Decontamination of large land areas

The selection of methods suitable for cleaning contaminated land and restoring it to productive use is complicated by:

- the large number of possible ecosystems, land uses and vegetation types;
- the large variation in the characteristics of soil classes;
- the complex behaviour of radionuclides with different soils;
- the impact that different cleanup techniques have on land ecosystems and restoration.

A generic assessment of the ecological impact of cleanup techniques on land types and land use classes in the USA is given in Ref. [14].

In selecting cleanup methods, the following factors must be considered: the type and level of contamination, how it was deposited, soil types, value of the land,
TABLE II. SUMMARY OF TECHNIQUES AND EQUIPMENT AVAILABLE TO CLEANUP/DECONTAMINATE DIFFERENT TYPES OF AREAS [6]

1. DECONTAMINATION OF BUILDINGS, EQUIPMENT AND PAVED SURFACES

Precipitation runoff, washoff and weathering
Motorized sweeping and vacuum sweeping
Firehosing
High pressure water jetting (hydrolasing)
Steam cleaning
Aqueous methods incorporating chemical additives
Abrasive jet cleaning
Road planing/grinding
Spalling
Gels and foams
Strippable coatings
Decontamination centres for equipment

2. DECONTAMINATION/CLEANUP OF LAND AREAS

Physical/chemical separation of radionuclides from the soil
Ploughing
Removal of vegetation
Removal of surface soil

alternative land use, population distribution, size of the affected area, equipment and staff availability, ecological impact, cost, safety, etc. Many techniques and types of equipment will be required for cleanup after any serious accident.

In general, the methods used to clean land areas can be classified as physical, chemical and agricultural or some combination of all of these.

The physical/chemical methods are the ones most frequently used. They include [6]: separating the particles containing radionuclides from the soil matrix; deep ploughing to remove the contamination from the surface; or removing the top layer of soil containing the contaminants. The volume of waste arising from the cleanup would be smallest for deep ploughing and largest for layer removal. The volume of waste from separation techniques would depend on how well the separation could be done.

Separation methods are applicable only to coarser grained soil or gravel in which radionuclides are associated with fine-grained particles which can be separated
easily. Experiments using washing, scrubbing and screening with water or chemical solutions have demonstrated the feasibility of some separation techniques [6, 19]. Many of the processes are similar to those used in conventional mineral ore treatment. The economics and practicality of these techniques for different radionuclides and soil types need to be demonstrated.

Deep ploughing has been investigated as an alternative to the removal of the contaminated soil layer [20–22]. Typically, a tractor drawn trenching plough is used to invert completely a thick layer of soil, placing the active top layer 50 to 70 cm below the surface and moving the deep clean layers to the top. In theory, most of the activity would be placed well below the lower boundary of the roots of the crop. However ploughing does not result in the perfect turnover of layers and some mixing occurs. The effectiveness of deep ploughing in reducing long term pickup of radionuclides in plants needs to be examined further. One study concluded that this method was costly and ineffective in reducing the uptake of radioactivity for deep rooted crops [20]. However it was concluded that deep ploughing should be effective in reducing direct radiation from surface contamination, airborne radioactivity and pickup by shallow-rooted plants. Another study [21] concluded that deep ploughing was the most effective cleanup method for the type of agricultural soils that were being dealt with. A reduction in the level of contamination by a factor of 10 was reported for deep ploughing to about 70 cm. Deep ploughing to 70 cm followed by the application of chemicals which prevented root intrusion achieved a reduction in the uptake of $^{90}$Sr by soya beans by a factor of 1000 [21].

The removal of vegetation should be an effective method of cleaning up certain areas since it can intercept almost all the fallout [6, 20, 21, 23]. The effectiveness of the technique depends on the density and type of vegetation, the nature of the contaminant and the method of deposition, i.e. wet or dry. Forage harvesters have been examined as a method of removing all kinds of crops [23]. Other machines have been used to reduce underbrush and all sizes of trees to chips for easy disposal. Removal of sod can be done by machines or manually. However, radiation hazards can be significantly higher for manual removal.

Layers of contaminated soil can be removed very effectively using many common types of earth-moving equipment such as bulldozers, graders and scrapers [6, 20–22]. These machines can remove layers of sod or soil as thin as 15 cm and transport the soil distances of 150 metres without reloading or stopping. This type of cleanup is most effective in flat relatively large areas having fine-grained compact earth. The efficiency of removal of the surface layer is affected by surface unevenness, soil texture, moisture content and vegetation cover.

Equipment for soil removal can be operated remotely or with shielding for cleanup of highly contaminated areas. Appendix A of Ref. [6] describes some of the equipment used at Chernobyl for removal of badly contaminated soil. For cleanup of relatively flat areas, the use of high capacity machines which load the soil directly into trucks may be desirable.
6.3. Management of contaminated forests

The management of contaminated forests is a complex problem which cannot be effectively dealt with in a short paper. The reader is referred to Ref. [6], which reviews the topic, and to other references, e.g. [24, 25], which deal with specific problems related to the management of contaminated forests.

7. INTERDICTION OF AREAS

Interdiction means the complete or partial restriction on the use of land or property for a period of time. During the late stage recovery operations, the competent authorities may decide to interdict all or part of the restricted area for long periods if it is determined that it is not practical or economic to clean these areas.

The cost to interdict economically important areas could be very high and in general it probably would be less expensive to cleanup such areas rather than interdicting them for long periods of time, unless the area is badly contaminated or release criteria are set too low. On the other hand, the interdiction of limited-use land such as certain forests, mountainous areas and marshes could involve relatively small economic penalties.

8. TRANSPORTATION AND DISPOSAL OF WASTES

The costs of loading, transporting and disposing of wastes arising from the cleanup of large contaminated areas will be a significant fraction of the total costs. The removal of a thin layer of contaminated material in the most active area around a damaged facility after a Chernobyl type accident could result in several million cubic metres of waste. The main pollutants of concern by the time cleanup starts after a reactor accident are probably $^{90}$Sr and $^{137}$Cs. The transportation and disposal of such large volumes of waste is time consuming and expensive. However operational experience on the procedures, techniques and equipment to transport and dispose of such wastes is available in some countries [5-7, 26].

Loading of the contaminated soils can be done using conventional construction equipment such as tracked loaders or force feed loaders with conveyors for the low level wastes which would make up most of the material. For higher activity wastes, the equipment would have to be modified for remote operation or operation with shielded cabs.

For the transportation of the low level wastes bulldozers could be utilized to move the wastes directly into trenches or dump trucks or rail cars for longer distances. The means of effectively managing the movement of large volumes of wastes is discussed in further detail in Ref. [7].
A variety of generic designs is available for the safe storage/disposal of the large volumes of contaminated soil and other material arising from the cleanup after a major reactor accident [7, 27]. These designs include natural basins or valleys; specially dug trenches; mined out quarries or open pit mines; underground mines; piling the wastes into large mounds, etc. If necessary, the impoundment facility could be lined with clay or other impermeable material barriers to minimize leakage. The wastes would also have to have suitably designed covers to prevent intrusion and the ingress of water into the wastes.

9. SUMMARY

In spite of all precautions, the possibility of an accident at a nuclear power plant which would result in the release of unacceptable amounts of radioactive material and cause serious contamination of surrounding areas cannot be excluded. One protective measure which may be required during the intermediate and late phases of an accident is the cleanup of areas.

Since the probability of such an accident occurring is low, it is unlikely that local governments around nuclear facilities would be willing to spend large amounts of money in preparing the detailed plans required for the safe and efficient cleanup of contaminated areas. However, since the potential consequences of such an accident are very large, especially in urban areas, some emergency planning for cleanup should be done to minimize the huge costs of such a cleanup and the potential detriment to humans.

To assist Member States to develop emergency preparedness plans for cleanup, the IAEA is preparing three reports which provide an integrated overview of the overall operational planning, implementation and management aspects of such large operations; methods and equipment available to carry out these actions; and the means to transport and dispose of large volumes of waste. The information in those three reports shows that a lot of equipment, techniques and experience relevant to the cleanup of large areas have been developed. However, further information is required on certain aspects including urban area decontamination, planning and managing such cleanups and the overall impact of the accident and cleanup on the affected area. Member States should be encouraged to prepare preliminary plans for cleanup of large areas. The cost of preparing preliminary plans need not be large.
REFERENCES


USO DEL ANALISIS COSTO-BENEFICIO
EN LA TOMA DE DECISION
DURANTE LA ETAPA DE RECUPERACION
DE ACCIDENTES NUCLEARES

E. PALACIOS, H. BRUNO, J.J. KUNST
Gerencia de Protección Radiológica y Seguridad,
Comisión Nacional de Energía Atómica,
Buenos Aires, Argentina

Abstract — Resumen

USE OF COST—BENEFIT ANALYSIS IN THE DECISION MAKING PROCESS DURING THE RECOVERY STAGE AFTER NUCLEAR ACCIDENTS.

During the recovery stage after an emergency, when the decisions being taken are aimed at re-establishing normal living conditions in the affected zone, technical considerations must be weighed against social and economic considerations. The Argentine licensing authority has made use of differential cost-benefit analysis to specify appropriate intervention levels, and on this basis an intervention level of $10^{-2}$ Sv/month was selected for the return of evacuees to the area; levels for contaminated foodstuffs have also been specified below which no restrictions on consumption would apply. The paper extends the application of these criteria to the selection of intervention levels for the removal of restrictions on land use.

INTRODUCCION

Los criterios utilizados en Argentina para seleccionar niveles de intervención relacionados con accidentes nucleares son consistentes con las recomendaciones internacionales [1, 2]: a) deben evitarse en lo posible los efectos no estocásticos;
(b) el riesgo de efectos estocásticos debe limitarse por la aplicación de contramedidas que produzcan un beneficio neto positivo en los individuos involucrados; y c) la incidencia total de efectos estocásticos debe ser reducida tanto como sea razonablemente lograble, reduciendo la dosis colectiva. Con relación a esta última recomendación, el análisis diferencial costo-beneficio ha permitido seleccionar niveles de intervención para ser aplicados en la última fase de accidentes nucleares, durante la recuperación.

**PASAJE DE LA NUBE Y DEPOSITO**

La autoridad licenciante seleccionó diversas contramedidas a ser aplicadas en la fase temprana de un accidente, durante el pasaje de la nube (es decir, mantenerse a cubierto, ingerir pastillas de iódurro de potasio, etc.) y en la fase siguiente, cuando la exposición se debe al depósito de material radiactivo. Así, los siguientes niveles de intervención fueron seleccionados: a) si la dosis equivalente efectiva integrada en las primeras 24 h no excede 0,1 Sv, la evacuación no es recomendada; b) si la dosis equivalente efectiva supera 0,1 Sv en 24 h pero es inferior a 0,1 Sv en las primeras 6 h, se evacúan todos aquellos casos de fácil implementación (es decir, no se recomienda la contramedida para la evacuación de hospitales o asilos de ancianos); y c) si la dosis equivalente efectiva supera 0,1 Sv en las primeras 6 h, se recomienda la evacuación de todos los casos involucrados.

**FASE FINAL**

En la fase final, cuando el material radiactivo depositado sobre el terreno ha decaído significativamente, es necesario decidir respecto del reingreso de las personas evacuadas. La aplicación del análisis diferencial costo-beneficio ha permitido seleccionar la tasa de dosis que optimiza la protección radiológica del público [3]. El valor óptimo de la tasa de dosis, $H_0$, resulta de relacionar el costo de mantener la contramedida (evacuación) y el valor monetario asignado a la unidad de dosis colectiva, $\alpha$:

$$H_0 = \frac{C}{\alpha}$$

El costo promedio de mantener evacuada a una persona, como costo adicional al costo normal de vida, fue estimado en dól. 100 por persona y por mes. El valor monetario asignado por la autoridad argentina a la unidad de dosis colectiva, $\alpha$, para fines de optimización, es de dól. 10 000/Sv·hombre. Por consiguiente, el nivel de intervención para permitir el reingreso de personas evacuadas es $10^{-2}$ Sv/mes.
CUADRO I. NIVELES DE INTERVENCION PARA ALIMENTOS CONTAMINADOS (Bq/kg)

<table>
<thead>
<tr>
<th>Grupo</th>
<th>Cereales</th>
<th>Tubérculos</th>
<th>Vegetales</th>
<th>Frutas</th>
<th>Carnes y pescados</th>
<th>Leche</th>
</tr>
</thead>
<tbody>
<tr>
<td>1&lt;sup&gt;a&lt;/sup&gt;</td>
<td>10</td>
<td>10</td>
<td>15</td>
<td>15</td>
<td>50</td>
<td>10</td>
</tr>
<tr>
<td>2&lt;sup&gt;b&lt;/sup&gt;</td>
<td>1000</td>
<td>1000</td>
<td>1500</td>
<td>1500</td>
<td>5000</td>
<td>1000</td>
</tr>
</tbody>
</table>

<sup>a</sup> Nucleidos de alta toxicidad radiológica (Pu-238, Pu-239, Pu-240, Am-241, Cm-242, Cm-244, Np-239).

<sup>b</sup> Nucleidos de baja toxicidad radiológica (Cs-134, Cs-137, I-131, Sr-89, Sr-90, Zr-95, Ru-103, Ru-106, Ba-140, Ce-144).

CONTROL DE ALIMENTOS

Mediante un análisis diferencial costo-beneficio, la autoridad licenciante estableció también niveles de intervención para alimentos, en términos de concentración, de manera que, si no se superan, no se considera la aplicación de restricciones al consumo [4]. El valor óptimo, $A_i$, resulta de dividir el costo del alimento en cuestión por unidad de masa o volumen, $b$, por el producto del factor dosimétrico por unidad de actividad ingerida, $F_d$, y el valor monetario asignado a la unidad de dosis colectiva, $\alpha$:

$$A_i = \frac{b}{\alpha \cdot F_d}$$

Los distintos nucleidos considerados se agruparon en dos grupos, según su toxicidad radiológica. Para el grupo 1, de alta toxicidad, se adoptó un valor de $F_d$ de $10^{-6}$ Sv/Bq y para el grupo 2, de baja toxicidad, se seleccionó el valor de $F_d = 10^{-8}$ Sv/Bq. Se consideraron los siguientes alimentos: leche, carne, vegetales, tubérculos, frutas y cereales. En el Cuadro I se presentan los valores de $A_i$ adoptados en la República Argentina.

USOS DEL SUELO

El nivel de intervención para el reingreso de personas evacuadas es $10^{-2}$ Sv/mes y corresponde a un depósito de $^{137}$Cs de $7 \times 10^{-6}$ Bq/m$^2$ o más. En todos los casos en que este valor se supere, el suelo será previamente arado en los primeros 30 cm de profundidad.
CUADRO II. NIVELES DE INTERVENCIÓN PARA USOS DEL SUELO

<table>
<thead>
<tr>
<th>Alimentos</th>
<th>Concentración de Cs-137 en suelo (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tubérculos</td>
<td>$10^5$</td>
</tr>
<tr>
<td>Cereales</td>
<td>$10^5$</td>
</tr>
<tr>
<td>Leche</td>
<td>$5 \times 10^4$</td>
</tr>
<tr>
<td>Vegetales</td>
<td>$5 \times 10^4$</td>
</tr>
<tr>
<td>Carne</td>
<td>$5 \times 10^4$</td>
</tr>
</tbody>
</table>

Alcanzado el nivel de intervención que permitiría el reingreso de personas evacuadas, automáticamente el uso del suelo quedaría liberado para cualquier propósito agrícola-ganadero. Sin embargo, zonas contaminadas podrían ser utilizadas para usos específicos mucho antes que las personas evacuadas sean autorizadas a reingresar en forma permanente. En el Cuadro II se presentan los niveles de intervención seleccionados para permitir el uso del suelo con fines específicos. Fueron determinados utilizando parámetros de transferencia medidos en Argentina, como consecuencia de “fall out” radiactivo [5].

REFERENCIAS

RADIATION PROTECTION STANDARDS AND MONITORING IN THE EVENT OF AN ACCIDENT AT A NUCLEAR FACILITY

G.M. AVETISOV, L.A. BULDAKOV, O.A. KOCHETKOV, D.P. OSANOV, A.V. BARABANOVA
Institute of Biophysics,
USSR Ministry of Health,
Moscow,
Union of Soviet Socialist Republics

Abstract

RADIATION PROTECTION STANDARDS AND MONITORING IN THE EVENT OF AN ACCIDENT AT A NUCLEAR FACILITY.

To regulate and limit irradiation of the population in the event of a nuclear accident, two levels of radiation exposure have been established. Depending on the level of exposure either emergency measures, including evacuation, or limited measures taking account of particular situation and local conditions, may be implemented. Evacuation of the population outside the 30 km zone was carried out after the Chernobyl accident and a wide range of protection arrangements were applied in adjacent areas. Temporary permissible contamination levels of the main types of foodstuff were established, including the admissible contamination levels of environmental objects (roads, clothes, etc.) and radioactive products monitored after the Chernobyl NPP accident. For effective protection, these levels must constantly be decreased. Emergency workers involved in dealing with the consequences of a serious radiation accident as a rule should not be exposed at doses exceeding the maximum permissible dose (1 MPD). As an exception, exposure of 2 MPD up to 5 MPD may be tolerated if it is followed up and compensated for under constant medical supervision. Other standards and limits should apply to the irradiation of a person under reproductive age. For men the dose accumulated by the age of 30 years should not exceed 12 MPD, for women aged up to 40 years planned heightened exposure should be forbidden. To diminish the occupational exposure of the personnel in the allocation of work in the area of the accident, the possibility of identifying zones with different levels of exposure should be considered, with appropriate health measures for each. Monitoring and radiation control are becoming increasingly important, as is the setting of standards for triggering emergency actions. As the Chernobyl accident has shown, this control should provide for continuous collection of information on the dynamics of the radiation situation over a wide area, including the monitoring of exposure to personnel.
Standardization of exposure and radiation parameters in the USSR is based on the Standards of Radiation Protection (SRP-76/87) [1].

Depending on the exposure of a human, three categories of individuals exposed are established: category A (personnel), category B (limited part of the population) and category C (the population of a district, territory, republic, country). Two classes of standards are specified for categories A and B: these cover basic dose limits and permissible levels. For category C (population) no standards are established and limitation of exposure is realized through regulation or control of radioactivity from environmental objects or technological processes, medical exposure doses and increased radiation background due to technology.

To regulate the irradiation of the population following an accident, two levels of radiation exposure have been established. Figures are given in Table I.

When nuclides other than those listed in Table I [2] are released to the environment, the values for levels A and B are in every case established by the Ministry of Health.

If the predicted exposure or contamination levels reach level B, emergency measures (e.g. immediate sheltering indoors, iodine prophylaxis, exclusion or limitation of ingestion of contaminated food, transfer of milk, transfer of productive livestock to uncontaminated pastures, provision of forage fodder, evacuation) should be taken.

<table>
<thead>
<tr>
<th>Source of radiation</th>
<th>Unit</th>
<th>Level of exposure</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>A</td>
</tr>
<tr>
<td>External gamma radiation</td>
<td>Gy</td>
<td>0.25</td>
</tr>
<tr>
<td>Thyroid exposure due to iodine-137 intake</td>
<td>Gy</td>
<td>0.25</td>
</tr>
<tr>
<td>Integrated airborne concentration of iodine-131</td>
<td>kBq·s·L⁻¹</td>
<td>Children</td>
</tr>
<tr>
<td>Total ingestion of iodine-131</td>
<td>kBq</td>
<td>55</td>
</tr>
<tr>
<td>Maximum contamination of new milk or daily ration with iodine-131</td>
<td>kBq/day</td>
<td>3.7</td>
</tr>
<tr>
<td>Initial density of iodine-131 fallout on a pasture</td>
<td>kBq/m²</td>
<td>26</td>
</tr>
</tbody>
</table>
### Table II. Provisional Permissible Limits (PPL) of Contamination for Basic Food Products (Bq/L, kg)

<table>
<thead>
<tr>
<th>Name of product</th>
<th>PPL-86</th>
<th>PPL-88</th>
</tr>
</thead>
<tbody>
<tr>
<td>Drinking water</td>
<td>370</td>
<td>18.5</td>
</tr>
<tr>
<td>Milk</td>
<td>370</td>
<td>370</td>
</tr>
<tr>
<td>Condensed milk</td>
<td>18 500</td>
<td>1 100</td>
</tr>
<tr>
<td>Dried milk</td>
<td>3 700</td>
<td>1 800</td>
</tr>
<tr>
<td>Curds</td>
<td>3 700</td>
<td>370</td>
</tr>
<tr>
<td>Cheese</td>
<td>7 400</td>
<td>370</td>
</tr>
<tr>
<td>Butter</td>
<td>7 400</td>
<td>1 100</td>
</tr>
<tr>
<td>Sour cream</td>
<td>3 700</td>
<td>370</td>
</tr>
<tr>
<td>Vegetable oil, margarine</td>
<td>7 400</td>
<td>370</td>
</tr>
<tr>
<td>Meat and meat products: beef</td>
<td>3 700</td>
<td>3 000</td>
</tr>
<tr>
<td>Pork, lamb, fat</td>
<td>3 700</td>
<td>1 800</td>
</tr>
<tr>
<td>Poultry</td>
<td>3 700</td>
<td>1 800</td>
</tr>
<tr>
<td>Eggs (Bq/piece)</td>
<td>1 850</td>
<td>1 800</td>
</tr>
<tr>
<td>Fish</td>
<td>3 700</td>
<td>1 800</td>
</tr>
<tr>
<td>Vegetables, table greens, potatoes, fruits, berries, fruit juices</td>
<td>3 700</td>
<td>740</td>
</tr>
<tr>
<td>Bread and bread products, grain, grain products, cereals</td>
<td>370</td>
<td>370</td>
</tr>
<tr>
<td>Sugar</td>
<td>1 800</td>
<td>370</td>
</tr>
<tr>
<td>Mushrooms (fresh and dried), wild berries</td>
<td>18 500</td>
<td>1 800</td>
</tr>
</tbody>
</table>

If the predicted exposure or contamination does not exceed level A, no emergency measures need be taken. However, measures for limiting exposure are needed. In this case the Ministry of Health establishes the provisional basic dose limits and permissible limits of exposure as well as developing the health regulations to assure economic activity on the contaminated areas. It is primarily concerned with setting standards for radioactivity concentration in agricultural products and foodstuffs.

Iodine-131, $^{134}$Cs and $^{137}$Cs are the major dose-delivering nuclides acting as a result of oral intakes of radioactivity. Most countries with developed nuclear power regulate only the annual limits on intake (ALIs) by ingestion and inhalation for an
individual member of the population. In the USSR, permissible concentrations of nuclides in air and drinking water have been established [1]. The radionuclide content of certain foodstuffs is not regulated. A decision standard in the event of an accident is specified for a critical product (cow milk) as is a radionuclide of highest significance for the accident.

The need for operational decision making on sorting and prohibition of use of certain food products varies after the accident. As at first $^{131}$I is most hazardous, intensively introduced to the human body in spring and summer by both milk and leafy vegetables, it is of importance to establish standards for the permissible $^{131}$I content of milk, dairy products (curds, sour cream, cheese, butter) and leafy vegetables.

The exposure of child thyroid (the critical organ for $^{131}$I) should not exceed 0.3 Gy, so the permissible $^{131}$I content of milk is limited to 3.7 kBq/L.

In connection with the subsequent growth of the contribution of $^{137}$Cs and $^{134}$Cs to dose formation and with the presence of rare earths in foodstuffs, a problem inevitably arises as regards the control and sorting of these by means of the simplest equipment, which requires the setting of standards for the total beta activity. In the USSR such provisional standards were introduced in 1986 and then revised in 1988 (see Table II) [3].

A considerable area is contaminated after an accident, leading to the necessity of organizing large scale radiation monitoring of roads, buildings, vehicles, households, clothes, human skin, etc., to avoid or minimize the propagation of contamination to adjacent areas and the further irradiation of the population.

To arrange radiation monitoring on such a large scale it is necessary primarily to have standards really reflecting the radiation situation, to assure a sufficient supply of radiometric equipment and to settle a number of organizational questions. It is reasonable to rescind the established standards for the permissible levels of contamination in beta particles per square centimetre per minute and to pass to setting standards for the dose rate for gamma radiation from the objects to be measured for the following reasons:

— there are rather more operative measurements of gamma dose rate relative to the measurement of contamination in particles per cm$^2$ per min;
— it is difficult to measure contamination in beta particles per cm$^2$ per min when the gamma radiation background is high.

In developing the permissible levels of radioactive contamination it is possible to start from the established relationship according to which the level of contamination equal to 10 000 beta particles per cm$^2$ per min corresponds to a gamma dose rate of 1 mR/h as measured near a contaminated flat surface. For example, the level of contamination of 0.1 mR/h corresponds approximately to the value established by SRP-76/87 for the permissible contamination of personnel’s working clothes, equivalent to 800 beta particles per cm$^2$ per min.
Dosimetric radiation monitoring is of great importance along with the setting of standards. It should include the comprehensive monitoring of radiation and the monitoring of personnel and population exposure, surface contamination of various objects, volume activity, environmental parameters and foodstuffs. In this complex of interrelated control activities, the monitoring of the parameters of radiation (dose equivalent rate, volume airborne radioactivity of major dose-forming nuclides and, above all, iodine and land fallout of such nuclides) with the aid of radioactivity surveillance stations is of primary significance. Such a monitoring system must provide continuous data on the dynamics of the developing radiation situation over a sufficiently wide area around an NPP to allow the estimation of the possible levels of exposure of the population around it.

The next level of control should include the monitoring of surface contamination, both in units of dose equivalent rate and in beta particles per cm$^2$ per min, of various objects and household articles, including the monitoring of contamination of farm products and the main foodstuffs.

Finally, the third level of control must provide for the direct monitoring of personnel and population exposures by various means. Among such means are, in the first place, wide range external gamma dosimeters, skin beta dosimeters (they are needed to control the exposure of the emergency workers), means of measuring the $^{131}$I doses to the thyroid, whole body counters and methods of excreta sampling for bio-assay to estimate mainly the internal exposures due to incorporated alpha nuclides. It should be noted that the above means of monitoring are sufficiently developed and involve a wide variety of techniques and procedures. The monitoring of skin beta exposures perhaps deserves special mention. The release of great amounts of fission products in the event of an NPP accident leads to the formation of a massive surface beta radiation source. Its role in the irradiation of humans is of particular significance in the first hours and days after the accident. Therefore, in the planning of accident damage recovery, special measures should be taken as regards protection of open areas of skin and control of beta radiation doses by multilayer beta dosimeters.

Limiting the irradiation of the emergency workers assumes a special significance. It should be noted, first of all, that all persons employed in emergency and rescue operations are given the same status as personnel of category A and they fall under provisions regulating the permit to work and the levels of exposure of persons in this category. As a rule, the work must be arranged so that the exposure would not exceed the maximum permissible dose (1 MPD) of 5 rem or 5 cGy. If necessary, with the permission of a competent authority (the Ministry of Health) the exposure of up to 2 MPD may be admitted only once a calendar year provided that such an exposure will be compensated for over the next five years of occupational activity. Exceptionally, exposure doses above 2 MPD are admitted, but not above 5 MPD. Such an increased exposure may be allowed only once in the entire working lifetime of a worker and must be compensated for over the next ten years of occupational
work. The dose of 5 MPD is considered to be potentially dangerous to health and all persons receiving such doses must be helped out of the zone of irradiation and directed to medical inspection. Further exposure can be allowed only in the absence of contra-indications. Irradiation of a person under reproductive age should be considered separately. For men, the dose accumulated by the age of 30 years should not exceed 12 MPD in any case; for women up to 40 years old the planned increased irradiation should be forbidden.

The above provisions should be consolidated by appropriate national legislation.

To diminish the occupational exposure when planning the arrangement of work in the area of the accident, the possibility of establishing several zones with different levels of exposure should be considered. The first zone includes the accident unit and the adjacent area where the basic recovery work is carried out. The levels of radiation (dose equivalent rate) in this zone can be very high, therefore, a strict dosimetric monitoring of both the radiation situation and personnel must be realized. Access to the zone is through a sanitary locker room with a total change of clothes. The second zone is an area where the preliminary work is done, that is the productive capacity needed to perform accident recovery is generated. In this zone the individual external gamma radiation dose is monitored and when the work is finished there is a need for more detailed studies by whole body counting.

The third zone is a zone where access is allowed only for the workers responsible for the emergency work. The main management organizations, possibly temporary staff, and other auxiliary services are sited in this zone. The passage of people and vehicles between the zones is strictly controlled. Therefore, means of washing people and vehicles should be provided for at the control posts. Strictly keeping to the regime of passage through the zones will make it possible to minimize or avoid the transfer of contamination to a pure zone.

The experience gained in setting standards of hygiene and management of radiation monitoring has shown that in the implementation of the whole complex of measures directed to a further improvement of the radiation safety of humans it is possible to achieve an appreciable decrease in individual and collective personnel and population doses. However, in some cases accidents may occur which result in uncontrolled exposure of the workers occupied with nuclear facilities, including the personnel involved (firemen, ambulance crews, guides, etc.).

About 300 persons, for instance, were taken to hospital in the first three days after the Chernobyl NPP accident. Of these, 145 were diagnosed to have acute radiation sickness. Thirty patients are known to have died. Obviously, both the radiation levels and effective radiation factors mainly depended on the location where the person was working at the moment of the accident and after it.

The workers and personnel who were indoors, at the industrial site, on the roof and in other workplaces were exposed to the combined action of various radiation factors, predominantly gamma and beta radiation due to the gas–steam cloud
### TABLE III. ESTIMATED DOSES FOR THE
**CHERNOBYL ACCIDENT DEATHS (Gy)**

<table>
<thead>
<tr>
<th>External</th>
<th>Internal</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Thyroid</td>
</tr>
<tr>
<td>Occupational workers</td>
<td></td>
</tr>
<tr>
<td>12.4</td>
<td>1.8</td>
</tr>
<tr>
<td>11.8</td>
<td>1.2</td>
</tr>
<tr>
<td>10.9</td>
<td>0.4</td>
</tr>
<tr>
<td>10.1</td>
<td>1.3</td>
</tr>
<tr>
<td>9.3</td>
<td>0.7</td>
</tr>
<tr>
<td>8.9</td>
<td>1.0</td>
</tr>
<tr>
<td>8.5</td>
<td>1.3</td>
</tr>
<tr>
<td>8.2</td>
<td>7.6</td>
</tr>
<tr>
<td>7.5</td>
<td>1.5</td>
</tr>
<tr>
<td>6.7</td>
<td>1.0</td>
</tr>
<tr>
<td>6.6</td>
<td>1.1</td>
</tr>
<tr>
<td>6.5</td>
<td>6.0</td>
</tr>
<tr>
<td>6.1</td>
<td>7.9</td>
</tr>
<tr>
<td>5.8</td>
<td>9.2</td>
</tr>
<tr>
<td>5.5</td>
<td>1.28</td>
</tr>
<tr>
<td>5.3</td>
<td>1.2</td>
</tr>
<tr>
<td>4.7</td>
<td>8.5</td>
</tr>
<tr>
<td>4.4</td>
<td>0.1</td>
</tr>
<tr>
<td>3.7</td>
<td>30.0</td>
</tr>
<tr>
<td>Firemen</td>
<td></td>
</tr>
<tr>
<td>13.7</td>
<td>0.3</td>
</tr>
<tr>
<td>13.7</td>
<td>16.7</td>
</tr>
<tr>
<td>12.7</td>
<td>0.5</td>
</tr>
<tr>
<td>11.1</td>
<td>1.1</td>
</tr>
<tr>
<td>10</td>
<td>1.6</td>
</tr>
<tr>
<td>9.4</td>
<td>1.0</td>
</tr>
<tr>
<td>Other personnel</td>
<td></td>
</tr>
<tr>
<td>11.8</td>
<td>6.0</td>
</tr>
<tr>
<td>4.1</td>
<td>1.9</td>
</tr>
</tbody>
</table>
(particularly for the staff in the area of the accident at that moment), and also gamma and beta exposure, both due to the fragments of the damaged reactor core and dispersed fuel (especially for the firemen and ambulance crew), as well as fission products on the skin, wet clothing and shoes. There was no major exposure due to internal contamination of the body caused by the fission products and fuel.

Data concerning the exposure doses of the patients who died of acute radiation syndrome (ARS) are given in Table III.

The patients in the gravest condition were among those working in the NPP because of concomitant damage factors such as thermic skin burns. Six of the firemen died and about 23 had ARS of various degrees of gravity. Some casualties were also registered among the workers involved (e.g. a guide outside).

None of the ambulance crews, who were the first to assist the affected workers, received exposure doses sufficient to cause ARS. Finally, the medical staff in the hospitals treating the affected workers received some exposure through contact with the patients. However, these doses, as a check determined, were much lower than 1 rem (100 Sv), excluding the radiation levels of the two most severely affected physicians which were higher. Therefore, in a very severe accident such as the one at Chernobyl it is usually the occupational workers who receive the highest doses of exposure. However, the emergency personnel may be also exposed to doses dangerous for health. This is why measures for the reduction of irradiation levels should be developed. These must primarily be technological, constructive and engineering measures providing safety and limiting the accident consequences (automatic antifire system, mitigation system, etc.). All workers directed into the zone of the accident should possess the full range of equipment for mitigating accident consequences, including robot systems, individual protection equipment and radiation monitoring gear. The radiation monitoring and medical assistance must function perfectly. Persons who have to work in controlled areas shall be given specific training for the work to be performed and adequately informed on the radiation situation, conditions and consequences of the work performed.

REFERENCES

RECOVERY OPERATIONS AFTER THE CHERNOBYL ACCIDENT: THE INTERVENTION CRITERIA OF THE USSR’S NATIONAL COMMISSION ON RADIATION PROTECTION

A.J. GONZALEZ
Division of Nuclear Safety,
International Atomic Energy Agency,
Vienna

Abstract

RECOVERY OPERATIONS AFTER THE CHERNOBYL ACCIDENT: THE INTERVENTION CRITERIA OF THE USSR’S NATIONAL COMMISSION ON RADIATION PROTECTION.

An informal meeting was arranged by the Secretariat of the International Atomic Energy Agency to discuss the policy on intervention criteria recommended by the USSR’s National Commission on Radiation Protection (NKRZ). The paper presents a summary of the criteria presented and the discussions and conclusions of the meeting. The meeting took place in Vienna on 12 May 1989 and was attended by nearly 100 experts from 20 countries. Many of the experts were participating in the meeting of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) being held the same week. The meeting dealt with possible problems arising from long term contamination by radioactive substances after major radiation accidents in general, with particular consideration of the post-Chernobyl situation. The NKRZ’s Chairman informed the participants of the contamination situation after the Chernobyl accident, of the remedial actions which had been taken, and of the intervention criteria that had been used and recommended for the future. Special attention was given to the problems remaining after the first years, during which the remedial actions were consistent with a globally accepted policy. There was little previous experience, however, of the long term effects of a nuclear accident causing large contamination. The policy proposed by NKRZ is to limit the total dose received from the accident by individuals in the critical groups in the USSR to 350 mSv over their lifetimes; such a level met with general acceptance by the participants. It was agreed, however, that the dose limitation for such purposes would have to be decided by national authorities, because it would depend on the local situation and on the severity of the accident.

1. PURPOSE

The purpose of this paper is to present a summary overview of an informal meeting arranged by the Secretariat of the International Atomic Energy Agency to discuss the post-Chernobyl intervention criteria recommended by the National Commission on Radiation Protection (NKRZ) of the Ministry of Health of the USSR. The
meeting took place in Vienna on 12 May 1989 and was attended by nearly 100 experts from 20 countries. Many were participating at the meeting of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) being held the same week. The informal meeting was chaired by B. Lindell. The NKRZ criteria were extensively described by the NKRZ’s Chairman, L.A. Il’in, and substantiated in the document ‘The Policy of the USSR National Commission on Radiation Protection on the Substantiation of Temporary Annual Dose Limits for Public Exposure due to the Chernobyl Accident’. A limited distribution of this document, coded UNSCEAR/XXXVIII/10 (8 May 1989), was made at the 38th Session of UNSCEAR held in Vienna from 8 to 12 May 1989.

The accident at Unit 4 of the Chernobyl Nuclear Power Plant in 1986 is unique in the history of the peaceful use of nuclear energy in its scope, character and potential effects on people and the environment. Radiation protection specialists have learned and will continue to learn many lessons from the Chernobyl accident. The accident has shown that, although a very sophisticated system of radiation protection standards and recommendations has been established for dealing with anticipatable, planned situations for which it is assumed that radiation exposure will be delivered with certainty, only very limited guidance was available for unanticipated, unplanned for, de facto problems such as the long term contamination resulting from an accidental release of radioactive materials into the environment. The most applicable guidance would be that relating to the problem of high levels of radon in existing dwellings; this guidance, however, is also very limited.

After a radiation accident a typical problem is how to deal with contaminated areas where the radiation level has been found to be higher than ‘normal’, and what criteria to use for deciding whether people in such areas must be evacuated or not, or for permitting evacuees to re-enter the areas. Several international bodies, including the International Atomic Energy Agency, are working at this moment to fill the vacuum of international guidance for dealing with this problem.

The NKRZ experts had to deal with a unique de facto problem: the unprecedented post-Chernobyl contamination. They have been quite successful, particularly at the earlier stage for which there was some applicable international guidance and for which there was also preparedness in the USSR. But, in the longer

---

run, there appeared to be problems which no national or international competent body had really anticipated well enough to prepare a policy for. The Soviet authorities had to face (and they are still facing) the problem in reality and as a result they have gained a great deal of experience that they are willing to share. The ultimate purpose of the informal meeting organized by the IAEA Secretariat was to foster information exchange among specialists in order to obtain the full benefit of such unique experience.

Additionally, there were two main impetuses for the meeting: the first being the sincere wish of the NKRZ to share their first-hand and extremely important experience; and the second was the NKRZ's wish to hear the experts' views on its approach, bearing in mind that as the Chernobyl accident developed the NKRZ found itself faced with many problems which had previously never been contemplated.

The meeting intended to discuss only the purely scientific aspects of the matter, but it should be emphasized that the political, economic and, particularly, the socio-psychological aspects of the Chernobyl accident have played such a significant part in the NKRZ's work that it was impossible to exclude them from the discussion.

2. THE NKRZ INTERVENTION CRITERIA
(as presented at the meeting)

2.1. The early stage

Immediately after the Chernobyl accident the NKRZ had to face a radiological situation without precedent in the history of radiation protection. In dealing with it, the NKRZ's main objectives were to avoid public exposures which might cause acute radiation injury and to keep as low as reasonably achievable the possible late detrimental effects on those present in contaminated areas and their progeny. In fact, in the first hours following the accident the NKRZ's first and main problem was that of protecting the staff and general public from the possible non-stochastic effects of irradiation.

Long before the Chernobyl accident the Soviet Union had, like other countries, prepared decision aiding criteria for use in the event of an accident involving a nuclear reactor: the Criteria for Decision Making on Measures to Protect the Population in the Event of a Reactor Accident, which were established by the USSR Ministry of Health in 1983. The criteria specify countermeasures to be taken at the

---

2 The basis of this criteria had already been discussed internationally at the IAEA Symposium on Handling of Radiation Accidents which was held in Vienna from 19 to 23 May 1969 (see: Dikobes, I.K.; Il'in, L.A.; Kozlov, V.M.; Moiseev, A.A.; Konstantinov, Yu.O.; Tarasov, S.I.; and Shamov, V.P.; "The adoption of urgent measures to protect the population in the event of an accidental release of radioactivity into the environment", Handling of Radiation Accidents (Proc. IAEA/WHO Symp. Vienna), IAEA, Vienna (1969).
TABLE I. DOSE CRITERIA FOR THE EVACUATION OF THE POPULATION AT THE EARLY STAGE OF AN ACCIDENT

<table>
<thead>
<tr>
<th>Country or international agency</th>
<th>Projected dose for first week</th>
<th>Year adopted</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Whole body (mGy)</td>
<td>Separate organs (mGy)</td>
</tr>
<tr>
<td>ICRP</td>
<td>50–500</td>
<td>500–5 000</td>
</tr>
<tr>
<td>IAEA</td>
<td>50–500</td>
<td>500–5 000</td>
</tr>
<tr>
<td>WHO</td>
<td>50–500</td>
<td>500–5 000</td>
</tr>
<tr>
<td>CEC</td>
<td>100–500</td>
<td>300–1 500a</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>100–500</td>
<td>300–1 500a</td>
</tr>
<tr>
<td>Netherlands</td>
<td>50–500</td>
<td>250–1 500b</td>
</tr>
<tr>
<td>Federal Republic of Germany</td>
<td>100–500</td>
<td>300–1 500</td>
</tr>
<tr>
<td>USSR</td>
<td>250–750</td>
<td>300–2 500b</td>
</tr>
</tbody>
</table>

* Skin value 1000–5000 mGy.
* Thyroid only.

early stage of an accident. They provided for the implementation of a series of measures aimed at protecting the public. A dose range of 250–750 mSv for a short period of time — literally a matter of a few days following an accident — was recommended as a trigger for decisions on extreme radiation protection measures, such as the evacuation of the general public. The level at which evacuation became compulsory was a forecast total radiation dose of 750 mSv and above.

The early stage is assumed to last from the beginning of the accident (initial release and plume formation) until the occurrence of ground deposition of radioactive material from the plume. However, the Chernobyl accident overturned all earlier ideas concerning the time factor and the dynamics of accident development: in the space of few days the exposed reactor core and the graphite cladding fire resulted in the release into the environment of an unprecedent amount of radioactive materials. At Chernobyl the length of the first stage was about ten days.

Besides preventing acute somatic effects, a secondary objective in the early stage was to reduce public exposures as much as possible by applying prompt and sometimes simple measures such as sheltering, respiratory protection, administration of stable iodine and evacuation. (At this early stage, the problem of limiting the late
stochastic effects, including radiation induced cancers and hereditary defects, was of lesser importance). The compulsory action levels corresponded to doses which could cause either acute radiation sickness or functional disturbance of specific organs, primarily of the thyroid. The criteria used for evacuation at the early stage of the accident are shown in Table I in comparison with criteria recommended by international organizations and adopted in some countries.

2.2. The intermediate stage

The main radiation protection requirement during the second, intermediate stage of the accident was to keep the stochastic effects as low as reasonably achievable. This was effected by the introduction of what were termed temporary annual dose limits (TADLs) for public exposures. It was initially assumed that this intermediate stage would last no longer than a year, but in practice it has been extended until 1989.

The TADLs were applied to the whole Soviet territory contaminated by the accident. Their enforcement was assisted by protective measures, such as withholding and bans on foodstuffs and drinking water contaminated in excess of temporary permissible levels derived from the TADLs; decontamination of buildings, roads, household articles, transport facilities and other environmental objects; and a number of agrotechnical measures, such as feeding the dairy cattle on stored fodder and deep ploughing of agricultural land. It should be noted that unless the above measures could ensure compliance with the TADL in any particular population centre, the population was relocated to areas where the constraints could be observed.

The IAEA and the International Commission on Radiological Protection (ICRP) recommend that decisions on action levels for introducing countermeasures at the intermediate stage should be based on the principle of optimization of protection. This principle may be implemented by using cost–benefit analysis or any other appropriate decision aiding techniques. However, at least two problems made the use of quantitative optimization techniques difficult in the aftermath of the Chernobyl accident:

(a) The use of quantitative optimization techniques is facilitated by dealing with objective radiation detriments alone. However, as experience from the Three Mile Island accident had already shown, psychological factors, such as radiophobia and stress, played a governing role in the post-accident decision making processes.

(b) The inaccuracies in factors in the implementation of optimization in practice, such as the value assigned to the unit collective dose, may lead to undesirable decisions. In the literature this value has been reported to range from US$ 1000 to 100 000 per man-Sv approximately. In the USSR such a monetary value had not been officially established and had never been used in practice.
### TABLE II. DOSE CRITERIA TO DECIDE UPON THE RELOCATION AT THE INTERMEDIATE STAGE

<table>
<thead>
<tr>
<th>Country or international agency</th>
<th>Projected dose for the first year (mSv/a)</th>
<th>Year adopted</th>
</tr>
</thead>
<tbody>
<tr>
<td>ICRP</td>
<td>50–500</td>
<td>1984</td>
</tr>
<tr>
<td>IAEA</td>
<td>50–500</td>
<td>1985</td>
</tr>
<tr>
<td>WHO</td>
<td>50–500</td>
<td>1985</td>
</tr>
<tr>
<td>Federal Republic of Germany</td>
<td>50–250</td>
<td></td>
</tr>
<tr>
<td>USSR</td>
<td>100</td>
<td></td>
</tr>
</tbody>
</table>

Thus, taking into account the scope and character of the Chernobyl accident and the uncertainties in the radiological conditions the NKRZ decided to follow ad hoc assessments of experts for setting TADL values.

The following TADLs were established for the population within the accident zone: 100 mSv in the first year after the accident, 30 mSv in the year 1987 and 25 mSv per year for 1988–1989. The observance of these dose constraints, which were approved by the USSR Ministry of Health, is verified by assessing the average dose to the critical groups in every population centre of the accident zone.

The scale of the Chernobyl accident was such that a number of Soviet districts, with a total population of some 273,000 people, had to be included within the areas of strict control (ASC). In these territories (a total of 789 population centres) major restrictions had to be introduced in order to comply with the TADL of 100 mSv in 1986–87, after which the NKRZ recommended the new TADLs for the following years up to 1 January 1990.

Table II presents the TADL for relocation adopted by the NKRZ for the first year on the basis of experts’ ad hoc evaluation, in comparison with other national and international criteria.

As indicated, in the second year the NKRZ introduced a TADL of 30 mSv for people living not in these areas only, but throughout the contaminated territory. The TADL was reduced to 25 mSv in each of the following two years. Thus, over the entire intermediate stage period, the conclusion of which can be tentatively estimated as 1990, the cumulative permitted exposure of the public is expected to be around 170 mSv.

As the NKRZ had to take extensive measures in order to keep contamination and resulting doses below the established levels, the result has been that the normal
pattern of human life in the affected areas has been disturbed because, among other reasons, the consumption of locally produced food products had to be banned. The inhabitants continue to receive food products imported into the area, principally milk, since contaminated milk is a main path for caesium intake. The consumption of milk on private farms was also forbidden, which means in effect that normal life has been considerably disrupted.

The large number of strict health measures caused great changes in practices in the social and economic structure. One of the primary measures for controlling internal contamination was the adoption of derived intervention levels (DILs) for drinking water and food. During the first three months after the accident, iodine radioisotopes present in drinking water and food were the most important internal dose contributors. Table III presents DILs for $^{131}$I adopted in the USSR and other countries and those recommended by international organizations.

Later, caesium radioisotopes became the important radiation source. Table IV shows the established DILs for contamination with $^{134}$Cs plus $^{137}$Cs in drinking water and food.

The wide range of DILs in Tables III and IV may be the result of differences in the local situation, including national differences in administrative and public health systems. Thus, in spite of similarity in the common principles forming the basis for remedial measures taken in various countries, DILs differed even in countries with similar contamination levels and social and economic structures. These differences caused concern and anxiety among the public, perplexity among experts and difficulties on the official level, including loss of public credibility.

The Nuclear Energy Agency of the OECD, in its report on The Radiological Impact of the Chernobyl Accident in OECD Countries (OECD, Paris, 1987), singled out the following reasons for countries' different responses to the Chernobyl contamination:

— "the large emphasis given to non-radiological, non-objective criteria in the decision making process;
— differing levels of uncertainty on impacts;
— the use of different methodologies in assessing the potential impact; and
— the use of different assumptions and values of parameters related to environmental transfer modelling, dosimetry modelling and characteristics of the affected population groups."

2.3. The long term stage

Before the Chernobyl accident there had been no accident in the world that had resulted in heavy, long term contamination of rural areas populated with hundreds of thousands of people. Not surprisingly, there was no available international guidance on radiation protection criteria for dealing with such a situation. The only
<table>
<thead>
<tr>
<th>Country</th>
<th>Drinking water (Bq/L)</th>
<th>Milk/dairy products (Bq/L or Bq/kg)</th>
<th>Vegetables (Bq/kg)</th>
<th>Meat (Bq/kg)</th>
<th>Other (Bq/L or Bq/kg)</th>
<th>Date adopted</th>
</tr>
</thead>
<tbody>
<tr>
<td>Austria</td>
<td>370</td>
<td></td>
<td>185</td>
<td></td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>Norway</td>
<td>1 000</td>
<td></td>
<td>1 000</td>
<td>1 000</td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>Finland</td>
<td>2 000</td>
<td>2 000</td>
<td></td>
<td></td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>Sweden</td>
<td>2 000</td>
<td>300; 50 000</td>
<td>300</td>
<td></td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>Italy</td>
<td>560</td>
<td></td>
<td>560</td>
<td>560</td>
<td></td>
<td>1971</td>
</tr>
<tr>
<td>Greece</td>
<td>125</td>
<td></td>
<td>90</td>
<td></td>
<td></td>
<td>26 May 1986 (CEC)</td>
</tr>
<tr>
<td>Netherlands</td>
<td>500</td>
<td></td>
<td>1 000</td>
<td></td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>Luxembourg</td>
<td>500</td>
<td></td>
<td>250</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Germany, Fed.Rep.</td>
<td>500</td>
<td></td>
<td>250</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>United Kingdom</td>
<td>11 000</td>
<td>2 000</td>
<td>110 000</td>
<td>160 000</td>
<td></td>
<td>March 1986 (NRPB-DL10)</td>
</tr>
<tr>
<td>Belgium</td>
<td>500</td>
<td></td>
<td>1 000</td>
<td></td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>Japan</td>
<td>110</td>
<td></td>
<td>220</td>
<td>7 400</td>
<td></td>
<td>2 May 1986</td>
</tr>
<tr>
<td>United States of America</td>
<td>1.5</td>
<td></td>
<td>560</td>
<td>1 850</td>
<td></td>
<td>1982 (forage)</td>
</tr>
<tr>
<td>Country</td>
<td>Drinking water (Bq/L)</td>
<td>Milk/dairy products (Bq/L or Bq/kg)</td>
<td>Vegetables (Bq/kg)</td>
<td>Meat (Bq/kg)</td>
<td>Other (Bq/L or Bq/kg)</td>
<td>Date adopted</td>
</tr>
<tr>
<td>-----------</td>
<td>-----------------------</td>
<td>-------------------------------------</td>
<td>--------------------</td>
<td>--------------</td>
<td>-----------------------</td>
<td>--------------</td>
</tr>
<tr>
<td>Canada</td>
<td>10</td>
<td>10; 40</td>
<td>70</td>
<td>70</td>
<td>70</td>
<td>May 1986</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(all but water)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Spain</td>
<td>125</td>
<td>90</td>
<td></td>
<td>90</td>
<td></td>
<td>26 May 1986</td>
</tr>
<tr>
<td>Portugal</td>
<td>125</td>
<td>90</td>
<td></td>
<td>90</td>
<td></td>
<td>26 May 1986</td>
</tr>
<tr>
<td>USSR</td>
<td>3 700(^a)</td>
<td>3 700–370 000(^a)</td>
<td>37 000(^a)</td>
<td>370–3 700</td>
<td>370</td>
<td>8 May 1986</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>16 May 1986</td>
</tr>
</tbody>
</table>

\(^a\) Permissible level of \(^{131}\)I ten times less in food supplied to children’s homes.
<table>
<thead>
<tr>
<th>Country</th>
<th>Drinking water (Bq/L)</th>
<th>Milk/dairy products (Bq/L or Bq/kg)</th>
<th>Vegetables (Bq/kg)</th>
<th>Meat (Bq/kg)</th>
<th>Other (Bq/L or Bq/kg)</th>
<th>Date adopted</th>
</tr>
</thead>
<tbody>
<tr>
<td>Austria</td>
<td>185; 300</td>
<td>110; 175</td>
<td>185; 300</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Norway</td>
<td>370</td>
<td>600</td>
<td>600; 6000</td>
<td>370</td>
<td></td>
<td>20 June 1986</td>
</tr>
<tr>
<td>Finland</td>
<td>1 000</td>
<td>1 000</td>
<td>1 000</td>
<td>1 000</td>
<td></td>
<td>22 May 1986</td>
</tr>
<tr>
<td>Sweden</td>
<td>300</td>
<td>300; 10 000</td>
<td>300</td>
<td>300</td>
<td></td>
<td>15 May 1986</td>
</tr>
<tr>
<td>Switzerland</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td>600</td>
<td></td>
<td>8 Sep 1986</td>
</tr>
<tr>
<td>Italy</td>
<td>250; 370</td>
<td>250; 600</td>
<td>250; 600</td>
<td></td>
<td></td>
<td>31 May 1986</td>
</tr>
<tr>
<td>Germany, Fed.Rep.</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td></td>
<td></td>
<td>31 May 1986 (CEC)</td>
</tr>
<tr>
<td>Greece</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td></td>
<td></td>
<td>31 May 1986 (CEC)</td>
</tr>
<tr>
<td>Ireland</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td></td>
<td></td>
<td>31 May 1986 (CEC)</td>
</tr>
<tr>
<td>Luxembourg</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td></td>
<td></td>
<td>31 May 1986 (CEC)</td>
</tr>
<tr>
<td>Netherlands</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td></td>
<td></td>
<td>31 May 1986 (CEC)</td>
</tr>
<tr>
<td>France</td>
<td>370</td>
<td>600</td>
<td>600</td>
<td></td>
<td></td>
<td>31 May 1986 (CEC)</td>
</tr>
<tr>
<td>Country</td>
<td>Pathway</td>
<td>Date adopted</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>------------------------</td>
<td>----------------------------------------</td>
<td>-----------------</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Drinking water (Bq/L)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Denmark</td>
<td>370</td>
<td>31 May 1986 (CEC)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>United Kingdom</td>
<td>3600; 370</td>
<td>March 1986 (DERL)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Belgium</td>
<td>370</td>
<td>31 May 1986 (CEC)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Turkey</td>
<td>370</td>
<td>31 May 1986</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>United States of America</td>
<td>90</td>
<td>(milk) 1982</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Canada</td>
<td>50; 100</td>
<td>May 1986</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Spain</td>
<td>370</td>
<td>31 May 1986 (CEC)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Portugal</td>
<td>370</td>
<td>31 May 1986 (CEC)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Australia</td>
<td>100</td>
<td>May 1986</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>USSR</td>
<td>18.5</td>
<td>6 Oct 1986</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(Diet of children)
available guidance was limited to criteria for the protection of the population in the initial and, eventually, intermediate phases of an accident, including time restrictions for outdoor residence, provision of iodine prophylaxis and evacuation. Precedents for solutions were lacking for dealing with the protracted lifetime exposures due to the contamination of large areas with long lived radionuclides. Thus, the competent bodies of the USSR, particularly the NKRZ, found themselves faced with a lack of relevant international guidance and the immensely critical responsibility of deciding the fate of thousands of people living in the affected regions.
The NKRZ’s approach was to introduce a radically new standard which was called a ‘lifetime dose limit’ (LDL). Provided the LDLs were not exceeded, all restrictive measures, including the ban on the consumption of certain products and on their substitution for imported products, could be removed. In parallel extensive measures would be taken to decontaminate these areas and improve radiological conditions in the population centres.

TABLE V. SIZE OF POPULATION AND NUMBER OF SETTLEMENTS IN THE AREAS OF STRICT CONTROL (ASCs)

<table>
<thead>
<tr>
<th>Region</th>
<th>District (No. of ASC settlements)</th>
<th>Total population (10³)</th>
<th>ASC population (10³)</th>
<th>Percentage of total population in ASCs</th>
<th>No. of settlements</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bryansk</td>
<td>Gordeyevskij (55)</td>
<td>153.6</td>
<td>111.8</td>
<td>72.7</td>
<td>274</td>
</tr>
<tr>
<td></td>
<td>Klimovskij (1)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Klintsovskij (41)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Krasnogorskij (46)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Novozybkovskij (131)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Kiev</td>
<td>Polesskij</td>
<td>35.8</td>
<td>20.8</td>
<td>58.0</td>
<td>28</td>
</tr>
<tr>
<td>Zhitomir</td>
<td>Luginskij (3)</td>
<td>107.8</td>
<td>31.2</td>
<td>28.9</td>
<td>45</td>
</tr>
<tr>
<td></td>
<td>Narodicheskij (37)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Obruchskij (5)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Mogilev</td>
<td>Bykhovskij (1)</td>
<td>135.4</td>
<td>23.3</td>
<td>17.2</td>
<td>193</td>
</tr>
<tr>
<td></td>
<td>Klimovichskij (17)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Kostyukovichskij (46)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Krasnopolskij (76)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Slavgorodskij (28)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cherikovskij (25)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gomel</td>
<td>Braginskij (7)</td>
<td>339.5</td>
<td>85.7</td>
<td>25.2</td>
<td>246</td>
</tr>
<tr>
<td></td>
<td>Budakoshelevskij (21)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Vetkovskij (69)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Dobrushskij (22)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Yelskij (2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Lelchitskij (1)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Loyevskij (1)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Kormyanskij (27)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Narovlyanskij (22)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Khoynijskij (26)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Checherskij (48)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>26 Districts</td>
<td>772.1</td>
<td>272.8</td>
<td>35.3</td>
<td>786</td>
</tr>
</tbody>
</table>
Figure 1 shows a map, which was produced jointly by the USSR’s Ministry of Public Health and the State Committees on Meteorology and Agroindustrial Complex, which shows the so-called ‘caesium spots’ in the contaminated areas. The total area of 137Cs contamination amounts to 10 000 km² for radioactive contamination in excess of 5.55 10⁵ Bq/m² (15 Ci/km²); the areas with a contamination higher than 1.85 10⁵ Bq/m² (5 Ci/km²) total 21 000 km². Furthermore, in the north of Rovno province and the south of Brest province, and in zones of Kaluga and Tula, radioactive spots of a lower density, from 0.74 to 1.85 10⁵ Bq/m² (2 to 5 Ci/km²), were detected, in soils characterized by an extremely low content of humus substances, with the result that the rates of caesium migration through the food chain are fairly high.

Table V gives specific data relating to the Bryansk, Kiev, Zhitomir, Mogilev and Gomel provinces, with a total of 26 districts, including the number of population centres in each district and the total number of people living in areas of strict control (ASC).
Table VI presents data prepared by the NKRZ and is based on information from measurements of external irradiation doses, in vivo measurements, and data relating to caesium concentrations in milk and throughout the entire chain. It includes the distribution of individual doses received by the public from 26 April 1986 to 1 January 1990 inclusive (in the case of the year 1990, forecast data are given). It is estimated that, whereas the 'permitted' dose (sum of the relevant TADLs) was in the order of 170 mSv, the actual average exposure of the public in these areas amounted to 60 mSv. It could therefore be concluded that the whole complex of measures that were put into effect 'saved' approximately 65% of the permissible dose laid down by the NKRZ. There were, however, a total of six population centres, with a combined population of some 800 inhabitants, in which these dose levels were exceeded. The NKRZ found that the people concerned were on the whole elderly foresters who neglected the NKRZ's rules and requirements and ignored all recommended protective measures, and for example, continued to drink the milk from their cows.

On the basis of the data thus far presented and also of corresponding theoretical data, the following tables present prognostic assessments of the lifetime radiation doses in the population centres, which were carried out on the basis of the withdrawal of all restrictions from 1 January 1990. Table VII shows that for 216 000 members

<table>
<thead>
<tr>
<th>Region</th>
<th>Below 350 mSv</th>
<th>Projected dose 350-500 mSv</th>
<th>Above 500 mSv</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Population</td>
<td>No. of settlements</td>
<td>Population</td>
</tr>
<tr>
<td></td>
<td>(10^3)</td>
<td></td>
<td>(10^3)</td>
</tr>
<tr>
<td>Bryansk</td>
<td>90.22</td>
<td>170</td>
<td>14.32</td>
</tr>
<tr>
<td>Kiev</td>
<td>8.05</td>
<td>22</td>
<td>11.90</td>
</tr>
<tr>
<td>Zhitomir</td>
<td>27.37</td>
<td>30</td>
<td>1.46</td>
</tr>
<tr>
<td>Mogilev</td>
<td>14.82</td>
<td>132</td>
<td>5.09</td>
</tr>
<tr>
<td>Gomel</td>
<td>76.04</td>
<td>190</td>
<td>5.66</td>
</tr>
<tr>
<td>Total</td>
<td>216.50</td>
<td>544</td>
<td>38.43</td>
</tr>
</tbody>
</table>
of the public living in these areas, the radiation dose will not exceed a lifetime value of 350 mSv. However, the public living in certain population centres will, in the scenario in which all restrictions are removed, receive radiation doses in the range 350–500 mSv, while the public in the third group of population centres — some 18 000 individuals in all — could, according to the prognosis, receive radiation doses in excess of 500 mSv over 70 years.

This is how the NKRZ arrived at the concept of the lifetime dose limit, the level of which was recommended to be 350 mSv over 70 years of uninterrupted residence in these areas. The NKRZ recommendation refers to the accumulated internal and external dose during 70 years of life and includes doses to which the population has been exposed since 26 April 1986 and excludes exposure to natural background radiation and medical exposures. The observation of the LDL would be regulated by the mean individual dose equivalent to the critical group of each populated area.

In the Soviet Union the permissible radiation dose for the public living in the vicinity of nuclear power plants and other nuclear installations is 5 mSv/a. This dose limit could serve as a reference for putting the LDL in proper perspective. In practice, by introducing an LDL of 350 mSv, what the NKRZ actually ends up with after 70 years is the equivalent of the pre-accident standard of some 5 mSv/a for members of the public living in the vicinity of nuclear power installations, on the understanding that the annual dose during the initial stage (i.e. in the first years following the accident) has been up to around 200 times higher, and that it will subsequently level off. During the first years after introducing the LDL, the actual annual dose rate may be 3–4 times higher than the annual dose limits.

Another reference point is the internationally recommended limits in other de facto situations. According to ICRP Publication 39, the action levels recommended for existing buildings with radon contamination are 20 mSv/a, while for the purpose of planning new buildings this publication recommends a figure of up to 10 mSv/a. Such figures pertain in many countries and do not cause any distress to the public. Thus, an LDL of 350 mSv was suggested as the criterion for the maintenance or cancellation of countermeasures in certain populated areas and also for further official decisions on relocation from those sites where it could not be ensured that the standard would not be exceeded.

Since, as mentioned before, several areas will exceed the 350 mSv criterion, the NKRZ recommended the Soviet Government to take measures to ensure that if this concept was put into practice, very meticulously planned operations to resettle the population from some of the population centres in these areas would be undertaken, since any decontamination or prohibitive measures there would in practice fail to have the desired effect.

It must be emphasized that the NKRZ abides by the principle of keeping doses as low as reasonably achievable, taking into account social and economic factors.

Text continued on p. 335.
TABLE VIII. ESTIMATED POTENTIAL LATE EFFECTS FOR THE POPULATION OF AREAS UNDER CONTINUED CONTROL ASSUMING A LIFETIME DOSE LIMIT OF 350 mSv

<table>
<thead>
<tr>
<th>Soviet Republic</th>
<th>Region</th>
<th>Population</th>
<th>Collective dose</th>
<th>Additional mortality (expected cases)</th>
<th>Natural incidence mortality (expected cases)</th>
<th>Excess over natural incidence (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>(10^3)</td>
<td>(10^4 man·Sv)</td>
<td>In the population exposed 'in utero'</td>
<td>Leukaemia</td>
<td>Solid cancers</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Leukaemia</td>
<td>Solid cancers</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>31.2</td>
<td>0.77</td>
<td></td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td></td>
<td>20.8</td>
<td>0.42</td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>Soviet Republic</td>
<td></td>
<td>52.0</td>
<td>1.19</td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>85.7</td>
<td>1.76</td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>Gomel</td>
<td>23.3</td>
<td>0.48</td>
<td></td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td>Mogilev</td>
<td>109.0</td>
<td>2.24</td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>111.8</td>
<td>1.97</td>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>272.8</td>
<td>5.39</td>
<td></td>
<td>4</td>
<td>4</td>
</tr>
</tbody>
</table>
TABLE IX. ESTIMATED POTENTIAL LATE EFFECTS FOR THE POPULATION OF THE REGIONS OF THE UKRAINE, BYELORUSSIA AND RSFSR MOST HEAVILY CONTAMINATED FOLLOWING THE CHERNOBYL ACCIDENT

<table>
<thead>
<tr>
<th>Soviet Republic</th>
<th>Region</th>
<th>Population</th>
<th>Collective dose (10^3)</th>
<th>Collective dose (10^4 man·Sv)</th>
<th>In the population exposed ‘in utero’</th>
<th>In the exposed population</th>
<th>Total (except thyroid)</th>
<th>Natural incidence mortality (expected cases) (10^3)</th>
<th>Excess over natural incidence (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ukraine</td>
<td>Zhitomir</td>
<td>1547</td>
<td>2.6</td>
<td>2</td>
<td>2</td>
<td>7</td>
<td>86</td>
<td>96</td>
<td>6.3</td>
</tr>
<tr>
<td></td>
<td>Kiev</td>
<td>4446</td>
<td>4.7</td>
<td>3</td>
<td>3</td>
<td>13</td>
<td>157</td>
<td>177</td>
<td>14.3</td>
</tr>
<tr>
<td></td>
<td>Chernigov</td>
<td>1428</td>
<td>0.9</td>
<td>1</td>
<td>1</td>
<td>3</td>
<td>30</td>
<td>34</td>
<td>4.6</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>7421</td>
<td>8.2</td>
<td>6</td>
<td>6</td>
<td>23</td>
<td>273</td>
<td>307</td>
<td>25.2</td>
</tr>
<tr>
<td>Byelorussia</td>
<td>Gomel</td>
<td>1678</td>
<td>6.7</td>
<td>5</td>
<td>5</td>
<td>19</td>
<td>224</td>
<td>253</td>
<td>7.3</td>
</tr>
<tr>
<td></td>
<td>Mogilev</td>
<td>1282</td>
<td>1.8</td>
<td>1</td>
<td>1</td>
<td>5</td>
<td>60</td>
<td>68</td>
<td>5.6</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>2960</td>
<td>8.5</td>
<td>6</td>
<td>6</td>
<td>24</td>
<td>284</td>
<td>321</td>
<td>12.8</td>
</tr>
<tr>
<td>RSFSR</td>
<td>Bryansk</td>
<td>1472</td>
<td>5.0</td>
<td>4</td>
<td>4</td>
<td>14</td>
<td>167</td>
<td>188</td>
<td>7.9</td>
</tr>
<tr>
<td></td>
<td>Tula</td>
<td>1865</td>
<td>1.3</td>
<td>1</td>
<td>1</td>
<td>4</td>
<td>43</td>
<td>49</td>
<td>10.1</td>
</tr>
<tr>
<td></td>
<td>Orel</td>
<td>864</td>
<td>0.2</td>
<td>0</td>
<td>0</td>
<td>1</td>
<td>7</td>
<td>7</td>
<td>4.7</td>
</tr>
<tr>
<td></td>
<td>Kaluga</td>
<td>1035</td>
<td>0.6</td>
<td>0</td>
<td>0</td>
<td>2</td>
<td>20</td>
<td>23</td>
<td>5.6</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>5236</td>
<td>7.1</td>
<td>5</td>
<td>5</td>
<td>20</td>
<td>237</td>
<td>267</td>
<td>28.2</td>
</tr>
</tbody>
</table>

Total 15617 23.8 17 17 67 794 894 66.3 2218 0.13 0.04
TABLE X. POTENTIAL LATE EFFECTS OF TOTAL EXPOSURE FOR THE POPULATION OF AREAS UNDER CONTINUED CONTROL IN THE CASE OF REMOVING A LIFETIME DOSE LIMIT OF 350 mSv

<table>
<thead>
<tr>
<th>Soviet Republic Region</th>
<th>Population</th>
<th>Collective dose</th>
<th>Additional mortality (expected cases)</th>
<th>Natural incidence mortality (expected cases)</th>
<th>Excess over natural incidence (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>In the population exposed 'in utero'</td>
<td>In the exposed population</td>
<td>Total (except thyroid)</td>
</tr>
<tr>
<td></td>
<td>(10^3)</td>
<td>(10^4 man·Sv)</td>
<td>Leukaemia</td>
<td>Solid cancers</td>
<td>Leukaemia</td>
</tr>
<tr>
<td>Ukraine</td>
<td></td>
<td></td>
<td>31.2</td>
<td>0.93</td>
<td>1</td>
</tr>
<tr>
<td>Zhitomir</td>
<td></td>
<td></td>
<td>20.8</td>
<td>0.72</td>
<td>2</td>
</tr>
<tr>
<td>Kiev</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>52.0</td>
<td>1.65</td>
<td>1</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>Byelorussia</td>
<td></td>
<td></td>
<td>85.7</td>
<td>2.11</td>
<td>2</td>
</tr>
<tr>
<td>Gomel</td>
<td></td>
<td></td>
<td>23.3</td>
<td>0.79</td>
<td>1</td>
</tr>
<tr>
<td>Mogilev</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>109.0</td>
<td>2.90</td>
<td>2</td>
<td>2</td>
<td>8</td>
</tr>
<tr>
<td>RSFSR</td>
<td></td>
<td></td>
<td>111.8</td>
<td>2.71</td>
<td>2</td>
</tr>
<tr>
<td>Bryansk</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>272.8</td>
<td>7.26</td>
<td>5</td>
<td>5</td>
<td>21</td>
</tr>
</tbody>
</table>
TABLE XI. POTENTIAL LATE EFFECTS OF TOTAL EXPOSURE FOR THE POPULATION OF THE CENTRAL AREAS OF THE EUROPEAN PART OF THE USSR FOLLOWING THE CHERNOBYL ACCIDENT

<table>
<thead>
<tr>
<th>Soviet Republic</th>
<th>Region</th>
<th>Population (10^6)</th>
<th>Collective dose (10^4 man-Sv)</th>
<th>Additional mortality (expected cases)</th>
<th>Natural incidence mortality (expected cases) (10^3)</th>
<th>Excess over natural incidence (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>In the population exposed 'in utero'</td>
<td>In the exposed population</td>
<td>Total (except thyroid)</td>
<td>Leukaemia</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Leukaemia</td>
<td>Solid cancers</td>
<td>Leukaemia</td>
<td>Solid cancers</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ukraine</td>
<td>Central</td>
<td>13.60</td>
<td>9.2</td>
<td>6</td>
<td>6</td>
<td>26</td>
</tr>
<tr>
<td></td>
<td>Western</td>
<td>8.32</td>
<td>0.8</td>
<td>1</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td>Eastern</td>
<td>14.47</td>
<td>1.5</td>
<td>1</td>
<td>1</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>Southern</td>
<td>14.60</td>
<td>0.9</td>
<td>1</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>50.98</td>
<td>12.3</td>
<td>8</td>
<td>8</td>
<td>34</td>
</tr>
<tr>
<td>Byelorussia</td>
<td>South-Eastern</td>
<td>2.96</td>
<td>8.5</td>
<td>6</td>
<td>6</td>
<td>24</td>
</tr>
<tr>
<td></td>
<td>North-Western</td>
<td>7.05</td>
<td>1.9</td>
<td>1</td>
<td>1</td>
<td>5</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>10.01</td>
<td>10.4</td>
<td>8</td>
<td>8</td>
<td>29</td>
</tr>
<tr>
<td>RSFSR</td>
<td>Central</td>
<td>9.78</td>
<td>8.0</td>
<td>5</td>
<td>5</td>
<td>22</td>
</tr>
<tr>
<td>Moldavia</td>
<td>Central</td>
<td>4.14</td>
<td>0.4</td>
<td>0</td>
<td>0</td>
<td>1</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td>74.91</td>
<td>31.1</td>
<td>22</td>
<td>22</td>
<td>87</td>
</tr>
</tbody>
</table>
### TABLE XII. ESTIMATED POTENTIAL CONGENITAL ANOMALIES (CA) AND MENTAL RETARDATION (MR) FOLLOWING IN UTERO EXPOSURE

<table>
<thead>
<tr>
<th>Exposure conditions</th>
<th>Pathology</th>
<th>Children born in the year of accident (4500 children)</th>
<th>Children born over 3 years after the accident (13 500 children)</th>
<th>Children born over 30 years after the accident (135 000 children)</th>
<th>Children born over 70 years after the accident (315 000 children)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Induced anomalies</td>
<td>Natural incidence of anomalies</td>
<td>Induced anomalies as % of natural incidence</td>
<td>Induced anomalies</td>
</tr>
<tr>
<td>Lifetime dose unlimited</td>
<td>CA</td>
<td>5.22</td>
<td>270</td>
<td>1.9</td>
<td>12.3</td>
</tr>
<tr>
<td></td>
<td>MR</td>
<td>0.013</td>
<td>34</td>
<td>0.038</td>
<td>0.031</td>
</tr>
<tr>
<td>Lifetime dose limit of no more than 350 mSv</td>
<td>CA</td>
<td>5.22</td>
<td>270</td>
<td>1.9</td>
<td>12.3</td>
</tr>
<tr>
<td></td>
<td>MR</td>
<td>0.013</td>
<td>34</td>
<td>0.038</td>
<td>0.031</td>
</tr>
</tbody>
</table>
TABLE XIII. ESTIMATED HEREDITARY EFFECTS IN THE POPULATION OF SETTLEMENTS UNDER CONTINUED CONTROL

<table>
<thead>
<tr>
<th>Exposure conditions</th>
<th>No. of generations</th>
<th>Induced effects as % of naturally occurring effects</th>
<th>Severe hereditary effects</th>
<th>Lethal hereditary effects</th>
<th>Induced effects as % of naturally occurring effects</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Total hereditary effects</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Induced</td>
<td>Natural</td>
<td>Induced</td>
<td>Natural</td>
</tr>
<tr>
<td>Lifetime dose unlimited</td>
<td>First two</td>
<td>182</td>
<td>18 900</td>
<td>0.96</td>
<td>0.36</td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>728</td>
<td>81 000</td>
<td>0.90</td>
<td>1.44</td>
</tr>
<tr>
<td>Lifetime dose limit of no more than 350 mSv</td>
<td>First two</td>
<td>135</td>
<td>18 900</td>
<td>0.71</td>
<td>0.27</td>
</tr>
<tr>
<td></td>
<td>10</td>
<td>540</td>
<td>81 000</td>
<td>0.67</td>
<td>1.07</td>
</tr>
</tbody>
</table>
TABLE XIV. RATIO OF CONGENITAL ANOMALIES TO BIRTHS FOR 1985–1988 IN THE GOMEL, MOGILEV AND VITEBSK REGIONS WITH REFERENCE TO THE YEAR 1985

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Gomel</td>
<td>1.00 ± 0.6</td>
<td>0.87 ± 0.2</td>
<td>1.19 ± 0.2</td>
<td>1.19 ± 0.3</td>
</tr>
<tr>
<td>Mogilev</td>
<td>1.00 ± 0.2</td>
<td>0.93 ± 0.2</td>
<td>1.00 ± 0.3</td>
<td>1.15 ± 0.2</td>
</tr>
<tr>
<td>Mean (control)</td>
<td>1.00 ± 0.3</td>
<td>0.89 ± 0.2</td>
<td>1.09 ± 0.2</td>
<td>1.18 ± 0.2</td>
</tr>
<tr>
<td>Vitebsk</td>
<td>1.00 ± 0.3</td>
<td>1.21 ± 0.2</td>
<td>1.36 ± 0.2</td>
<td>1.29 ± 0.2</td>
</tr>
</tbody>
</table>

Half-width of the confidence interval is given, using a 95% confidence probability.

The principle will be particularly applied to restrain the forecast radiation exposure of the critical population group of children. It should also be noted that the levels of caesium incorporation in children are assessed according to a 90% quantile of the actual concentration of caesium in milk in these areas, which provides certain safety margins in the assessments. Moreover, the metabolism constants for caesium in a child’s organism are calculated on the basis of the parameters used for adults and the calculations are based on an uninterrupted period of residence of 70 years in these areas.

The recommended LDL is to be introduced as of 1 January 1990 and is expected to give rise to a whole series of problems. A main concern refers to the question of the fundamental validity of the criteria. The NKRZ came to the conclusion that the introduction of the LDL will not result in major consequences for the state of health of the population. In fact, the NKRZ prognoses indicate that the levels of presumed additional cancer, genetic effects and congenital development abnormalities will be from 0.5 to 2.0% above the corresponding natural incidence.

The following tables present specific information relating to predicted effects taking three nosological forms: malignant neoplasms, teratogenic effects and genetic effects.

Table VIII shows the levels of anticipated effects in the population of the constantly controlled areas in terms of leukaemia, other forms of cancer arising from in utero exposure, and leukaemia and other forms of malignant neoplasm, excluding tumours of the thyroid gland, if the LDL of 350 mSv were to be introduced. The
Table also shows the extent to which these figures exceed the natural incidence levels, i.e. 1.5% in the case of leukaemia and 0.5% for solid cancers.

Table IX has the same format as Table VIII, and relates to the delayed effects on the population — some 15 million in all — living in the provinces referred to before. The corresponding figures for leukaemia and solid cancers in excess of the natural incidence are 0.12% and 0.04%.

Table X relates to the possible delayed effects in the strict control areas (i.e. the same areas as in Table VIII, but with the removal of all restrictions, including those on the level of irradiation). Whereas in Table IV the rises above the spontaneous level of incidence are 1.5% and 0.5%, in Table X they are 2.1% and 0.7% respectively.

Table XI describes the delayed effects for the central areas of the European part of the Soviet Union, where there are some 75 million inhabitants. It shows the anticipated occurrences of malignant neoplasms and leukaemia over and above the corresponding spontaneous levels.

Table XII shows similar calculated data, this time relating to congenital development abnormalities for the population centres under strict control. It takes account of both the scenarios mentioned before, i.e. for removal and maintenance of the restrictions, and presents these data in relation to the number of children born in the first year, the first three years, the thirty years and the seventy years following the accident.

Table XIII contains data on the genetic effects in the first two generations of the population. The calculations take into account various types of hereditary effects, including lethal effects.

Table XIV shows specific estimates of the frequency of development abnormalities in the Gomel and Mogilev provinces using the Vitebsk province (which suffered no contamination) for the purposes of comparison, and relating everything to 1985. The data failed to indicate any difference over the years 1985 to 1988, either by comparison or in real terms.

In summary, therefore, since the NKRZ proposal is to introduce the LDL of 350 mSv from 1 January of next year in conjunction with the withdrawal of all restrictions for the population living in these areas, it was of interest for the NKRZ to hear the reaction of experts attending the meeting and, eventually, to receive their support in respect both of the NKRZ approach and of the value recommended for the LDL.

In this regard it should be noted that in the Soviet Union radiophobia and the commotion being made in the media in the aftermath of the Chernobyl accident are still important factors. The scientists dealing with the radiation protection aspects of the accident are constantly having to defend themselves against attacks on their integrity in the media, and find it very difficult to persuade the public of the scientific validity of the radiation protection standards they recommend. As an extreme example of this problem it was recounted at the meeting that when, against a background
of scientific information, a photograph of a calf with only one eye was shown by sensationalistic media, the public, as might be expected, reacted in undue panic and fear.

3. DISCUSSION
(Summary of opinions of experts attending the meeting)

3.1. On the terminology

The NKRZ has introduced what it has termed a ‘lifetime dose limit’ for dealing with the particular situation of the long term contamination caused by the Chernobyl accident. However, it is not a real lifetime dose limit, but rather a limit on the dose committed from decisions on intervention following an unanticipatable, de facto situation: the post-Chernobyl contamination. The decisions to be made include whether to evacuate a contaminated area or not, whether to let people enter an area or not, and whether to ban a contaminated food or not. These various decisions are usually made on the basis of intervention levels resulting from a balance between the negative consequences of the decision and the dose commitment avoided as a result of the decision, rather than on the basis of dose limits.

It is more coincidentally the relation between the recommended intervention level of 350 mSv and the normal dose limit of 5 mSv per year, because there is no expectation of a reduced margin of doses from other sources. In fact, the adoption of the recommended levels will not mean that people will be prevented from incurring other exposures to radiation, which they usually would be permitted to incur within the dose limit.

The term of dose limit is therefore preferred for anticipatable, normal conditions of exposure rather than for this de facto situation. The so-called limit is really either an action level or a non-action level, i.e. an intervention level. The key concept should be that people should not be put in a worse situation as a result of an intervention and, if normal limits are used for deciding on intervention after an accident, the intervention might put people in a worse situation.

3.2. On the approach of the ICRP

In particular, the ICRP would not term a limit what is in fact an intervention level, because the ICRP reserves the term ‘limit’ for situations in which it is planned to use a radiation source, or to set up an installation, or to perform an operation. Thus, the ICRP limits are really a part of the system that would constrain the doses in such planned situations. If the control of the source is lost and the only control possible is either to apply a remedial action or not, the ICRP does not use the term
limit for the levels set for deciding on action; rather the ICRP would term such levels 'intervention levels' or 'action levels'. But aside from the terminological differences, there are no conceptual differences between the NKRZ LDLs and the ICRP intervention and action levels.

In deciding on intervention in such cases of long term contamination, a common question is whether to let people stay in or enter the area, or whether to evacuate people and not allow them to enter the contaminated area. In the case of the Chernobyl accident, and following the NKRZ criteria, if the decision is that they could stay or they could enter, it would mean that the critical group should receive at most a dose of 350 mSv. The dose level would be the total dose that the worst irradiated person would receive as a result of the decision, which would be somebody living his or her whole life in that place; such persons would constitute the 'critical group'; everyone who was already older would receive a lower dose. It seems that the level of 350 mSv, by coincidence, was calculated by multiplying a dose limit of 5 mSv per year by 70 years.

But, following ICRP criteria, the annual dose limit for planned situations and the intervention levels for post-accident situations belong to different accounts, since for a normal situation, e.g. for a planned new Chernobyl reactor or even for the other units at Chernobyl operating today, the 5 mSv per year limit would still apply and radiation doses would be delivered (and allowed) no matter of whether the 350 mSv dose were incurred or not. So as these quantities relate to different accounts, the intervention level should not really be compared with an annual dose limit (either 5 mSv per annum or any other). Nevertheless, the comparison could be a way of justifying the 350 mSv value by conveying the feeling that the related risk is very small.

The ICRP procedure for deciding the intervention level would be to decide whether the intervention is justified and (if it is justified) to optimize radiation protection by optimizing the intervention level. For post-accident situations, it would imply assessing the advantages and disadvantages of deciding on a specific intervention (e.g. allowing re-entry or not allowing re-entry) and making the decision on the basis that the countermeasure should do more good than harm. Optimization can be performed by intuition, with ad hoc expert advice, or by any available quantitative technique such as cost–benefit analysis.

On the basis of cost–benefit analysis, some decisions have been planned to allow re-entry when the maximum annual dose is around 50 mSv per year and some optimization studies resulted in optimum levels for relocation of people as high as 10 mSv per month. For the case of contamination with caesium which is absorbed into the ground and has a mean half-life of about fifteen years, a level of 50 mSv per year would imply an overall dose commitment (from that decision of allowing re-entry) of 750 mSv in a lifetime. This value is higher than the value recommended by NKRZ but it is readily acceptable because it results from optimizing the level of protection.
On the same grounds it would be interesting to evaluate the cost of the evacuation per person and per year. Using this quote, it is almost sure that optimum intervention level would lead to a dose commitment over a whole lifetime that would be in the range indicated. Thus, if it were a saving of efforts caused by the intervention, it could be shown (e.g. on the basis of differential cost–benefit optimization) that it is quite likely that the optimum value of the action level would be much higher than 350 mSv. Although 350 mSv is a prudent action level, a higher level — if optimum — would be perfectly acceptable as well.

There was some concern, though, that NKRZ felt that its approach of taking non-objective factors into account in some way is in conflict with the ICRP recommendations on optimization. After all, even such ad hoc approaches represent some kind of optimization, although the factors are not easy to express in cost–benefit terms. The ICRP optimization surely should introduce all of the inconveniences of evacuation and relocation.

3.3. On the perspectives given by natural background exposure

In areas contaminated by Chernobyl there is a similar situation to that in areas with different background radiation. For instance, in China, in the Quantung province there are two populations near Taijuan: around one million people incur a normal level of natural radiation exposure of around 2 mSv per year, and a similar number of people incur three times this level. There is a significant number of people, with about the same differences between the normal background and with the NKRZ derived average level of 5 mSv per year, and this case could be used for providing an appropriate perspective to the NKRZ levels.

Moreover, rather than taking the normal range of natural background radiation for comparison, the range in the Soviet Union could be used. In such a large country there must be very great differences between the natural levels in different places, not only in the remote places where the density of population is very low, but also in the densely populated parts of the Soviet Union.

3.4. On the criteria used in other countries

Apart from the terminology in question, the principles used by the NKRZ are very similar to the principles used in other countries to control radon exposures. The first principle is to do anything reasonable to prevent non-stochastic effects. The second, overriding other principles, is to justify the intervention and to optimize protection as far as possible. By ‘justification’ and ‘optimization’ is meant that the situation should not be made worse for people by application of countermeasures and that the level of intervention should be the best under the prevailing circumstances.

In the case of radon, studies have shown that older people tend not to worry so much about radon and they tend not to take expensive countermeasures against
radon. That is perfectly natural and there should not be any intention at all to evict old people from their dwellings even if they incur doses of 100 mSv per year because the disruption would be very bad for them.

A somewhat similar example to radon control relates to the control of the Chernobyl consequences themselves. With respect to food restrictions it has been said that it is possible, without many consequences, to keep doses below 1 mSv per year. (This corresponds to a long term average, so it would mean 70 mSv in a 70 year lifetime). It has also been said that special consideration should be given to children and to pregnant women. So for those groups doses above 1 mSv per year should certainly be avoided. But there might be other groups for whom the consequences might be quite significant if individuals are forced to stay below this limit of 1 mSv per year. For instance, the whole culture of reindeer breeders in Nordic countries might be threatened if they were to change their lifestyle as dramatically as would be necessary to limit their doses to below 1 mSv per year. Fishermen and other people might also suffer quite severe consequences from changing their lifestyles to keep doses below 1 mSv per year. In these cases there were decisions not to take any further countermeasures as long as these groups stayed below 10 mSv per year, which means 700 mSv in a 70 year long life. The countermeasure which will be taken if it is noticed that critical groups will exceed 10 mSv per year will only be increased follow-up and increased information. But their life will not be disrupted to keep their doses below 10 mSv per year. They should, however, be properly informed as to what extent their risk is increased if they incur doses of 10 mSv per year over a long period.

3.5. On the universality of intervention levels

Although the NKRZ recommended intervention level seems to be appropriate, it would be difficult to recommend a universal number for deciding on intervention. It is going to vary from country to country and it must depend on the extent of the contamination. Its selection comes into part of the consideration of where to spend efforts and wealth. It seems clear that the whole of a country’s gross national product should not be expended on cleaning up an area of a major accident. The intervention level has to be a compromise. The sort of number that would be used could be in a range of doses probably in the first year or two (the dose per year and a lifetime dose) which spans the NKRZ range, or possibly extending it.

3.6. On people’s attitudes

Different people are going to have different attitudes towards intervention. Old people usually do not want to evacuate their homes. Conversely some younger people will say, “‘We’ll not go there, ever — and we’ll keep away’”. People’s attitudes vary and to face them is a difficult problem. The NKRZ figure is not very
different from the lifetime dose due to natural background radiation and, therefore, it might be expected that people would not be concerned about doubling their exposure to natural background radiation since this does not seem terribly alarming.

The selection of an average value of 5 mSv per year over 70 years for everyone may have advantages in that it encourages co-operation of the media, of the people and of the authorities. But it may be easier to follow the ICRP approach, which is more flexible and permits the level to be adapted in relation to the local situation (and it may be convenient to have differences between certain local situations). The disadvantages are the differences between the dose incurred by various people which are difficult to understand. If these differences are within the ranges of the natural background (and this is the case), it would be more understandable to the media or the authorities. But, in fact, with the system of simple optimization (not a very sophisticated optimization) it is possible to have more flexibility and to adapt the countermeasure to the situation in the local area. There is the possibility for different intervention levels depending on the de facto situation in the different areas.

It is assumed that an individual incurring a dose of 350 mSv over a lifetime will have a chance of around 1% of incurring cancer, which is a very small increase over the natural chance. Thus, the 350 mSv level may seem a sensible number to end up with, but on the basis of comparing it to doses due to natural background radiation and the possibility of incurring cancer rather than to annual dose limits.

3.7. On the media reaction

On a news programme by a television station in a western country, a film from the Soviet Union was presented which discussed the problem of intervention levels. The film was made in the region of Chernobyl. It showed two groups of Soviet citizens. One group complained about not being permitted to go back and the other group complained about having to go back. And a question was asked: "How would [that country's] radiation protection authorities react to a similar type of incident?". The answer given made a comparison with the natural background radiation: it explained that in that country there are regions where the background level is six times as high as in other regions. However, no one has ever suggested that such regions should be evacuated. Moreover, in that country, there are native inhabitants who live almost entirely on caribou meat which has been contaminated over the years with caesium. And it has been accepted that a 5 mSv per year extra dose was warranted in return for the better nutrition and the fact that their whole culture and way of life is dependent on hunting these caribou.

3.8. On risk perception

There is a tendency that the safer the country, the more worried people are about risks. People in generally safe countries are extremely worried about various
risks and commonly perform studies on risk perception. There is one observation which it is important to keep in mind about these studies: when people talk about risk perception they really mean that other people do not rank acceptance of the risks in the same way as they do. For example, if various sources of risk are ranked according to the actual scientifically estimated risk, it would be logical to conclude that the acceptance of risk would follow the same ranking. But this is not the case: people continue to smoke yet they may object very much to some other risk from a source that they do not like — even if that risk is very small in comparison with the risks of smoking. So, the acceptance of risk depends on many other things than the level of risk. And that is not unreasonable. People are very much influenced by whether they enjoy taking certain risks or not.

4. CONCLUSION

The accident at Chernobyl resulted in the release into the environment of an unprecedented large amount of radioactive radionuclides and in the contamination of areas and population on scale without precedent. This situation presented new problems that necessitated new approaches to radiation protection, particularly in the formulation of principles for setting up criteria for dealing with long term contamination in post-accident situations.

The informal meeting organized by the IAEA Secretariat to discuss the NKRZ criteria for intervention was very useful for sharing information in this field. The NKRZ Chairman informed the participants of the contamination situation and of the remedial actions and action levels, both decided and proposed. The meeting, which dealt with problems arising from long term contamination by radioactive substances after major radiation accidents in general, with particular consideration of the post-Chernobyl situation, provided members of UNSCEAR, radiation protection experts attending meetings at the IAEA and officers from the Agency Secretariat a forum and an opportunity for exchanging opinions on this very important matter.

The intervention criteria used by the NKRZ in the early and intermediate stages after the accident were very useful for the organization of radiation protection of the public in the aftermath of the accident in the area affected where countermeasures have to be taken under difficult conditions and within a short period of time. The initial remedial actions and action levels were consistent with international policy. There was little previous international experience, however, of how to deal with the long term effects of a nuclear accident causing widespread contamination.

There is no universal solution to a de facto situation like the post-Chernobyl long term contamination; it must differ from country to country and depend upon the seriousness of the situation. Therefore, the dose limitation criteria in such situations would have to be decided by national authorities; they would depend on the local situation and on the severity of the accident. In particular, the setting of the
action levels for major countermeasures, such as relocation of people, either in terms of an annual dose or for lifetime dose, should always be within the competence of national authorities. The case of Chernobyl was clearly under the competence of the USSR authorities which — in setting the levels for intervention — had to take into account the medical and biological consequence of the dose averted and economical and social factors influenced by the countermeasures.

On the basis of the above mentioned considerations, a ‘lifetime dose limit’ or action level of 350 mSv was set by the Ministry of Public Health of the USSR for the total lifetime dose received due to the accident by individuals in the critical groups of the areas contaminated as a result of the Chernobyl accident. Apart from a terminological question (radiation protection experts prefer to reserve the term ‘limit’ for dose limitation in anticipatable, planned situations, and to use the term ‘level’ for the doses which may trigger protective actions in existing situations), the 350 mSv level met with general support by the participants at the IAEA meeting in May 1989. From the discussion it was clear that for one reason or another, perhaps for many reasons, the figure is lower than, or in the range of, the levels that the experts might well have proposed had they been responsible in a similar situation.

**BIBLIOGRAPHY**


INTERNATIONAL ATOMIC ENERGY AGENCY, Principles for Establishing Intervention Levels for the Protection of the Public in the Event of a Nuclear Accident or Radiological Emergency, Safety Series No. 72, IAEA, Vienna (1985).


Intervention Levels for Nuclear Reactor Accidents, Gezondheidsraad, the Hague (Netherlands) (1986).


Zeihami, E.A., Morris, M.D., Thyroid Cancer Risk in the Population around the Nevada Test Site, Health Phys. 50 1 (1986) 19–32.
Abstract—Résumé

POST-ACCIDENT RADIOLOGICAL MEASUREMENTS: CONTAMINATION MAP.

In post-accident situations where a release has resulted in the deposition of radioactive materials on the soil, it is essential to take contamination measurements immediately after the accident and throughout the recovery operations. A description is given of measurement equipment which has been developed in France for performing efficient measurements over large areas: it includes a gamma mapping device and an in situ gamma spectrometry system. These are compared and their use is described.

1. INTRODUCTION

Après un rejet accidentel ou une dispersion de matières radioactives, la mise en place de contre-mesures et la conduite des opérations supposent la connaissance préalable de la contamination déposée sur le sol et donc des mesures pour:

— faire le diagnostic initial ;
— suivre l’évolution de la situation;
— caractériser l’efficacité de la restauration;
— effectuer un contrôle final avant déclassement de la zone.
Les moyens de mesure qui peuvent être mis en œuvre en France sont de quatre types:

1) les mesures par appareil portatif;
2) les prélèvements de sol;
3) la spectrométrie gamma in situ qui consiste à disposer au-dessus du sol un spectromètre étalonné pour donner des valeurs de contamination surfacique sur quelques centaines de mètres carrés;
4) la cartographie gamma par hélicoptère qui établit la carte des dépôts radioactifs au sol.

Après une description rapide de ces moyens de mesure, en particulier des deux derniers systèmes qui sont récents, on présente une intercomparaison des résultats obtenus et des propositions pour la stratégie d’emploi.

2. CARTOGRAPHIE GAMMA HELIOPERTEE

Le système de cartographie gamma que nous avons mis au point a été décrit en détail par Rosenberg. Il est constitué:

— d’un équipement embarqué à bord d’un hélicoptère léger pour l’acquisition de données;
— d’un équipement à terre qui, à partir des données acquises en vol, restitue la carte de la contamination.

2.1. Principe de l’acquisition de données en vol

Le schéma de la figure 1 montre les équipements essentiels destinés à l’acquisition de données en vol.

L’hélicoptère effectue au-dessus de la zone à mesurer un balayage selon des trajectoires prédéterminées. Chaque seconde, on enregistre:

— le spectre gamma provenant du scintillateur NaI, sur 256 canaux;
— la position de l’hélicoptère: altitude par rapport au sol, coordonnées dans un système rectangulaire (Lambert ou UTM).

Les données de positionnement peuvent être calculées par rapport à la position de balises préalablement disposées au sol. La précision de localisation est alors de l’ordre de cinq mètres, mais la surface mesurable en une seule opération est de quelques dizaines de kilomètres carrés, due à la portée des balises. Un nouveau système

de localisation pour de très grandes surfaces est en développement. Il utilise un ensemble de navigation par radar Doppler et permet l’examen de surfaces de plusieurs centaines à un millier de kilomètres carrés avec, toutefois, une résolution spatiale inférieure.

2.2. Traitement des données

Le traitement des données enregistrées en vol s’effectue après le retour de l’hélicoptère au sol. Il utilise un matériel informatique transportable qui restitue une carte en couleurs de la contamination au sol. Selon le traitement choisi, on peut obtenir la contamination caractéristique d’un radioélément donné, ou encore la carte des débits de dose.

2.3. Caractéristiques opérationnelles

La sensibilité du système est fonction de plusieurs paramètres:

— le rendement d’émission et l’énergie des rayonnements émis par le radio-élément recherché (seuil en énergie: 50 keV);
FIG. 2. **Efficacité du détecteur en fonction de la hauteur au-dessus du sol.**

— le bruit de fond dû à la radioactivité naturelle liée à la nature géologique des terrains.

Pratiquement, pour un hélicoptère volant à 40 m du sol et 70 km/h, on détecte des «taches» de $^{137}$Cs ayant une activité égale ou supérieure à 2 kBq/m$^2$ et une surface minimale de 2000 m$^2$, ou encore des sources ponctuelles de l'ordre de 4 MBq.

Le matériel est mis en œuvre par une équipe de quatre personnes. En France, il est mobilisable avec un délai de 12 heures et peut être déplacé en tout point du territoire dans un véhicule aérotransportable.

Il faut environ 2 heures pour équiper un hélicoptère, 1 heure 30 de vol pour examiner une surface de quelques dizaines de kilomètres carrés et 1 heure à 1 heure 30 pour obtenir les premières cartes de la contamination.

3. **SPECTROMETRIE GAMMA IN SITU**

Le moyen de spectrométrie gamma in situ permet de mesurer localement le dépôt de matières radioactives sur le sol, sur une surface de quelques dizaines à quelques centaines de mètres carrés, avec une excellente sensibilité.
Il est essentiellement constitué d'une diode Germanium HP installée sur un mât télescopique, lui-même monté sur un véhicule tout-terrain.

Le mât télescopique permet d'installer la diode à différentes hauteurs (3 à 9 m) et, pour chaque point, de modifier ainsi la surface vue par le détecteur. Le système a été étalonné en supposant une contamination surfacique homogène, ce qui permet, par analyse du spectre délivré par la diode, d'obtenir la valeur moyenne de la contamination de la surface mesurée.

La sensibilité du système est fonction:
— de la sensibilité du détecteur,
— de la radioactivité naturelle locale,
— du rendement d'émission et de l'énergie du rayonnement émis par le radioélément recherché,
— de la hauteur du détecteur au-dessus du sol,
— du temps de comptage (classiquement 1000 à 2000 secondes).

La courbe de la figure 2 donne la variation de l'efficacité de mesure pour le $^{137}$Cs sur un terrain calcaire, en fonction de la hauteur du détecteur.

On voit donc qu'il est possible de mesurer par ce moyen des activités très faibles, soit de l'ordre du kBq/m$^2$. Il serait toutefois inutile d'affecter cet ensemble à la mesure de contamination surfacique supérieure à 100 kBq/m$^2$, ce qui le contaminerait et pourrait le rendre non opérationnel.

4. PRELEVEMENTS DE SOL ET MESURE PAR APPAREIL PORTATIF

La technique des prélèvements de sol est bien connue et largement utilisée. Elle mérite cependant quelques commentaires quant à son utilisation pour les mesures post-accidentelles:

— la sensibilité que l'on peut attendre est excellente, mais la mise en œuvre des analyses peut être alors relativement longue, ce qui est un inconvénient lorsque le nombre de prélèvements est élevé;
— c'est la seule méthode qui permette de mesurer la distribution des radio-éléments en fonction de la profondeur sous la surface lorsque ceux-ci font l'objet d'une migration;
— la surface prélevée doit être suffisamment grande pour être représentative.

Concernant ce dernier point, les indications que nous avons relevées montrent que la surface prélevée par mesure doit être au minimum de 0,25 mètre carré pour obtenir une valeur statistiquement représentative.

Comme les prélèvements de sol, les appareils portatifs ont été largement utilisés jusqu'à présent pour effectuer des mesures de contamination surfacique. Cependant, les inconvénients évidents qui s'y attachent tels que la pénétration de per-
sonnels dans les zones contaminées lorsque des robots ne sont pas utilisables, la lenteur de l'opération et l'intégration des résultats limitent leur emploi à des zones de faible surface.

5. STRATEGIE D'EMPLOI EN SITUATION POST-ACCIDENTELLE

Pour établir une comparaison entre les moyens de mesure présentés ci-avant, on a effectué des relevés sur une zone faiblement contaminée par de l'uranium naturel. La figure 3 donne la valeur de la contamination surfacique mesurée le long d'un axe géographique de cette zone. On remarque le bon accord entre les différentes valeurs.

![Graphique comparaison des moyens de détection](image)

**FIG. 3.** Comparaison des moyens de détection: cartographie gamma par hélicoptère, spectrométrie gamma in situ, détecteur portatif IPAB 71 et prélèvements de sol.
TABLEAU I. COMPARAISON DE QUELQUES CARACTÉRISTIQUES DES MOYENS DE DÉTECTION UTILISABLES EN Situation POST-ACCIDENTELLE

<table>
<thead>
<tr>
<th>Caractéristiques</th>
<th>Cartographie $\gamma$ par hélicoptère</th>
<th>Spectrométrie $\gamma$ in situ</th>
<th>Prélèvement de sol</th>
<th>Appareil portatif</th>
</tr>
</thead>
<tbody>
<tr>
<td>Résolution en énergie (keV)</td>
<td>100</td>
<td>2</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>Résolution spatiale (m$^2$)</td>
<td>$\approx 5 \times 10^3$</td>
<td>10-100</td>
<td>0,2</td>
<td>1</td>
</tr>
<tr>
<td>Sensibilité (Bq·m$^{-2}$)</td>
<td>$10^3$-$10^4$</td>
<td>$10^2$-$10^3$</td>
<td>$10^2$</td>
<td>$10^3$-$10^4$</td>
</tr>
<tr>
<td>Rapidité des mesures (m$^2$·h$^{-1}$)</td>
<td>$10^7$-$10^9$</td>
<td>10-400</td>
<td>0,1</td>
<td>$10^2$-$10^3$</td>
</tr>
</tbody>
</table>

Le tableau I résume les caractéristiques opérationnelles principales des moyens de mesure utilisés. A partir de ces caractéristiques, on peut établir un guide pour l’utilisation de ces moyens en situation post-accidentelle.

La première opération à effectuer est l’établissement d’une carte à l’aide du dispositif de cartographie gamma héliportée.

Les résultats donneront rapidement une vue globale du dépôt contaminant et permettront, si ce n’est déjà fait, de prendre les décisions quant aux contre-mesures immédiates à mettre en place. À ce stade, il n’apparaît pas indispensable d’utiliser la spectrométrie gamma in situ ou d’effectuer des prélèvements de sol.

Les premières mesures qui pourront être reconfirmées aussi souvent que nécessaire permettront aussi d’établir et de suivre le plan de restauration.

Au cours du déroulement de ce plan, de nouvelles mesures de cartographie gamma héliportée seront réalisées pour vérifier la progression et l’efficacité des opérations. Mais à ce stade, la spectrométrie gamma in situ apparaît aussi comme un moyen de contrôle très intéressant à cause de sa sensibilité et de sa souplesse d’utilisation.

À la fin des opérations de restauration, les prélèvements de sol associés à la spectrométrie gamma in situ fourniront les résultats de contrôle finaux permettant de caractériser les opérations engagées.

Un tel schéma peut ne pas avoir une portée générale, mais il peut néanmoins guider les responsables techniques confrontés à une contamination de surface accidentelle.

6. CONCLUSION

Les matériels qui ont été décrits, notamment la cartographie gamma par hélicoptère et la spectrométrie gamma in situ, présentent, grâce à leur souplesse et
leur efficacité, un progrès très important pour les mesures à réaliser après un accident radiologique, surtout quand on les compare aux méthodes utilisant des appareils portatifs: ils permettent en très peu de temps d’avoir une connaissance globale des dépôts au sol. Au cours de la restauration, ils contrôlent la progression des opérations.

D’autres utilisations de ces moyens sont envisageables: établissement de la situation radiologique sur les sites où doivent être implantées les installations nucléaires, surveillance de l’environnement pendant le fonctionnement de ces installations, recherche de sources ponctuelles, etc.
DECONTAMINATION OF URBAN AREAS AFTER NUCLEAR ACCIDENTS

H. DE WITT, W. GOLDAMMER, H.D. BRENK
Brenk Systemplanung,
Aachen

R. HILLE, H. JACOBS, K. FRENKLER
Nuclear Research Centre Jülich,
Jülich

Federal Republic of Germany

Abstract

This study discusses the removal of radionuclides from urban construction materials by weathering and forced decontamination. With respect to intermediate and long term exposure, external irradiation due to deposited caesium causes the major contribution after nuclear accidents. Therefore, an experimental programme was designed and conducted to study in more detail the removal of ¹³⁷Cs from selected impervious urban surfaces. The objective of the experiments was to reveal important factors which influence the decontamination process. Particular effort was spent on the investigation of the applicability of ion exchange processes as a forced decontamination technique. The interpretation of the results is presented on the basis of a model which was set up in order to discriminate between the influence of the important physical and chemical processes. The purpose of this description is not only to aid understanding of the decontamination procedure but mainly to develop better decontamination techniques by a specific manipulation of the relevant processes.

1. INTRODUCTION

In the event of a serious accident in a nuclear power plant the release of radioactive isotopes to the atmosphere could lead to a contamination of surfaces in urban areas. The adverse health effects due to external irradiation from these surfaces are most important in regions with high population densities, where countermeasures such as evacuation are particularly problematic for social and economic reasons. Processes of natural decontamination that remove the activity from the surfaces have been studied for several years and with particular emphasis after the Chernobyl accident [1–6]. Unfortunately, it was found that these processes proceed very slowly, especially in the case of caesium, so that the protection of the inhabitants of urban areas after nuclear accidents requires the application of efficient and preferably non-destructive methods of forced decontamination.
The investigation of this issue was the first objective of the research programme. The other one was to extend the information on decision strategies after nuclear accidents. The work was funded by the Federal Minister of the Environment, Nature Conservation and Nuclear Safety and by the Commission of the European Communities.

The following discussion of results will be confined to the part of the project dealing with forced decontamination techniques. Our results on natural decontamination processes agree in general with those of other authors; details can be found in Ref. [7].

2. EXPERIMENTAL PROGRAMME

The experimental investigation of decontamination processes was conducted using typical urban construction materials, in particular concrete paving slabs, asphalt layers and clay roof tiles. The test specimens were contaminated to a level

![FIG. 1. Forced decontamination of paving slabs with (a) water; (b) tap water; (c) 0.1 mol/L NaOH; (d) rain; (e) 0.1 N HCl; (f) 0.5 N HCl; (g) 0.2 M NH₄NO₃.](image-url)
between 40 and 160 Bq/cm$^2$ by sprinkling with an aqueous solution of $^{137}$Cs and 50 $\mu$g/mL inactive $^{133}$Cs. The ratio of the initial activity to the residual activity after decontamination defines the decontamination factor. The activities were determined by gamma spectroscopy. Each experiment was repeated five times and the results were averaged.

Each decontamination experiment was carried out in 11 steps. One single step consisted in spraying the sample with a certain amount, in most cases 22.2 L/m$^2$, of decontaminant. Before the residual activity was measured the sample was allowed to dry.

In a first phase of the experiments the efficiency of different decontaminants was examined. Apart from water in different purities, sodium hydroxide and hydrochloric acid, the decontamination by an ion exchange process was investigated using ammonium nitrate ($\text{NH}_4\text{NO}_3$). The result, presented in Fig. 1, shows that in agreement with the findings of Sandalls [8, 9] the highest decontamination factor was achieved by the use of ammonium nitrate.

The main part of the experimental programme was aimed at the systematic investigation of the factors determining the decontamination properties of the chosen urban construction materials with ammonium nitrate. In particular, influences of the concentration and quantity of decontaminant, the contamination level and the surface condition prior to the contamination such as wet/dry or clean/dirty have been examined.

![FIG. 2. Forced decontamination of paving slabs with molar concentration of NH$_4$NO$_3$ of (a) 0.003; (b) 0.03; (c) 0.2; (d) 1.0.](image-url)
FIG. 3. Forced decontamination of clay roof tiles with 0.2 mol/L NH$_4$NO$_3$ on (a) dry and (b) wet surfaces.

FIG. 4. Forced decontamination with 0.2 mol/L NH$_4$NO$_3$ of (a) paving slabs; (b) asphalt layer and (c) clay roof tiles.
3. RESULTS

The main results of the decontamination with ammonium nitrate can be summarized as follows:

— In the range from 40 to 160 Bq/cm\(^2\) no dependency of the decontamination factor on the contamination level was found.
— The amount of decontaminant applied per decontamination step had no influence on the decontamination factor.
— The concentration of the decontaminant was varied between 0.003 and 1 molar. As can be seen in Fig. 2, a drastic effect on the decontamination factor resulted. For concentrations above 0.2 molar a certain saturation was observed.
— Samples that had been wetted prior to the contamination could be decontaminated much more easily afterwards (Fig. 3).
— The decontamination efficiency was found to depend strongly on the type of material (Fig. 4). The residual activity after 11 decontamination steps with 0.2 molar ammonium nitrate was about 90% for clay roof tiles, about 20% for asphalt and 10% for concrete paving slabs with reference to the original activity.

4. INTERPRETATION

In order to interpret the experimental results a phenomenological model has been developed. The main aim was to derive an expression for the decontamination factor as a function of a few parameters that are, on the one side, physically and chemically well defined, and, on the other side, experimentally accessible. Besides explaining the experimental results this model can support, notwithstanding its simplicity, the design of more powerful decontamination methods.

Basically, each non-destructive decontamination method has to employ two distinct processes. In the first step the atoms of the contaminant that are adsorbed at the outer surface of the material or at inner surfaces in pores have to be desorbed. In an ion exchange process this desorption is brought about by an exchange of the adsorbed contaminant ions with ions of the decontaminant such as NH\(_4\) ions, in the case of ammonium nitrate.

The chemical nature of this process will be complicated in general due to the variety of different adsorption sites that are present in materials such as clay, asphalt or concrete. In realistic situations this complexity will be further enhanced by the abundance of chemical substances present on these surfaces. A detailed description of the processes involved is impossible and, for practical purposes, not desirable.
Looking at these processes from a general point of view, however, reveals an important common aspect. The equilibrium of an ion exchange process is governed, for any type of adsorption site, by the chemical law of mass action. More precisely, this means that for any type of adsorption site the equilibrium concentration of contaminant ions mobilized by the ion exchange process is determined by the concentrations of contaminant and decontaminant ions which can be formulated in the expression shown in Eq. 1. In the simplest case one type of adsorption site will dominate, otherwise a few major types will have to be taken into account. A description of the mobilized portion of the activity in equilibrium has to consider only the equilibrium constant or constants $k$ and the number per unit volume $d_{ad}$ of the relevant type or types of adsorption sites as material parameters.

After the contaminant ions have been mobilized they have to be removed by the second step of the decontamination process. This will be particularly difficult for the portion of the contamination that is adsorbed in pores of the material. If no forces are applied to the fluid in the pores (through high pressure or vacuum) the removal of this portion is controlled by the diffusion of the contaminant ions through the pores to the surface of the material, where they can be washed off. The kinetics of this diffusion process is controlled by structural and chemical parameters of the material and by physical parameters such as temperature. A description is obviously complicated. Thus, it seems to be adequate to summarize the influence of these quantities within a single parameter $\alpha$, describing the portion of the mobilized contamination that is actually removed by a decontamination process from a certain material. A value $\alpha = 1$ represents the ideal case where all mobilized contaminant ions are removed. Within these assumptions an expression for the decontamination factor $F$ can be derived [7]

$$F = \frac{D + (1/k - 1) d_{ad}}{(1 - \alpha)D + (1/k - 1 + \alpha)d_{ad}}$$

(1)

where $D$ means the concentration of the decontaminant and $k$ is the equilibrium constant. In the ideal case $\alpha = 1$ this simplifies to

$$F = k \left\{ \frac{D}{d_{ad}} - 1 \right\} + 1$$

(2)

These equations can contribute on a qualitative level, as will be shown in the next section, to the understanding of the experimental results and to the identification of the main factors that influence the decontamination efficiency. Furthermore, this model could provide information for the improvement of decontamination results by designing more efficient methods that aim specifically at the relevant processes.
Finally, a classification of the material and process parameters seems to be possible that allows a calculation of the attainable decontamination factors in actual emergency situations as a basis of the planning of decontamination strategies.

The derivation of the above equations does not consider the time dependence of these processes which is, of course, also important for practical purposes. The decontamination factor defined in Eq. (1) is to be understood as the highest achievable factor after a sufficiently long decontamination time. In Ref. [7] some ideas for the description of the time dependence of the processes are given, which are not reported here for the sake of brevity. Moreover, the limited data basis from the experiments was not sufficient to elaborate and verify a well-founded description.

5. DISCUSSION OF THE RESULTS

The main results of the experiments summarized in Section 3 can be understood easily on the grounds of the described model. With the exception of the experiment where the surface was wetted prior to the contamination it seems to be reasonable to assume that the portion of the mobilized contamination that was removed remained constant in all experiments with the same material. Consequentially, only influences on the ion exchange process have to be discussed, as characterized by Eq. (2).

The observed differences in the decontamination efficiency for the materials studied can, in the first instance, be related to the material parameters $k$ and $d_{ad}$ in Eq. (2). A rough estimate of the number of adsorption sites per unit volume $d_{ad}$ shows that it will be some orders of magnitude smaller than the number of decontaminant ions within this volume. Thus, the main factor of influence is supposed to be the equilibrium constant $k$. This means that especially the material clay, which is very hard to decontaminate, contains very efficient adsorption sites characterized by a small equilibrium constant $k$ for the ion exchange with ammonium ions.

This assumption can be supported by chemical arguments. Clay basically consists of silicon–oxygen chains. Some of the silicon atoms are exchanged by metal atoms such as aluminium. Due to their different valence a net charge of the chains results. This charge is compensated for by positively charged ions (mainly the alkali ions $Na^+$ and $K^+$) which are located between the chains. This structural property of clay provides an efficient mechanism for the adsorption of an alkali ion like that of caesium at empty sites or through an exchange with a $Na^+$ or $K^+$ ion. This strong chemical binding brings about a retention mechanism for caesium ions within the structure of the clay material and may explain the observed small equilibrium constant $k$.

So far, we have not taken into account the effect of the factor $\alpha$. This negligence of diffusion processes is obviously not justified when experiments with different porous materials are compared. The differences in porosity between the materials will effect the diffusion of the mobilized containment ions to the surface of the
material. This provides an additional mechanism that retains the contamination for porous materials such as clay. A separation from the retention mechanism caused by the chemical properties is not possible on the basis of the experimental data. Since this would be very desirable for theoretical and for practical reasons, further experiments should be conducted that allow an assessment of these factors.

One direct piece of evidence for the importance of diffusion processes is revealed by the experimental result in Fig. 3, which shows that a clay surface that is wet prior to the contamination can be decontaminated much more easily than a dry surface. Since the wet surfaces were allowed to dry before the decontamination started this can only be explained by a different spatial distribution of the contaminant in the two cases. This is obviously related to the fact that a contaminant applied to a dry surface can soak into the pores of the clay material while in the case of a wet surface the contaminant ions can only get into the pores by a diffusion process since these are already filled with water. Thus, a greater portion of the contamination will be located immediately at the surface in this case, which facilitates the decontamination process drastically. Seen from this point of view, the benefit of the presence of water on the surface of porous materials prior to the contamination relies on the same processes that make the decontamination of these materials difficult if the contamination has taken place in a dry condition of the surface.

6. CONCLUSION

The experimental investigation has shown that the forced decontamination of surfaces contaminated with caesium should preferably rely on ion exchange processes, for example by the use of ammonium nitrate. This non-destructive method combines a comparably good efficiency with many practical advantages, such as low costs and environmental compatibility.

Although the principal applicability of this method has become evident, the experimental results cannot be regarded as satisfactory from a practical point of view. The overall dose reduction for the inhabitants of urban areas that can be achieved by the application of the investigated method is too small to allow the replacement of measures such as evacuation by forced decontamination to a significant extent. Consequentially, the efficiency of this method has to be enhanced in further experiments.

One possible way for the design of more efficient decontamination methods is based upon the model described in Section 4. The main aim in the development of this model was to reduce the important processes to a few phenomenological parameters that can be regarded as characteristic for the material and decontamination method. Though simple, this model has been shown to be able to explain the main experimental results.
Future development should be directed to a deeper understanding of the relevant processes and an identification of the main factors that have inhibited a better decontamination efficiency. This could lead to the design of more efficient methods that influence the relevant processes specifically. Once satisfactory decontamination methods have been developed, the model parameters could be used in a classification of urban materials and different decontamination methods. This could form the basis of the planning of decontamination strategies in actual emergency situations.

REFERENCES


DECONTAMINATION OF ROADS BY SPECIALIZED VEHICLE.

A vehicle for the decontamination of roads has been developed in France by the Directorate for Military Applications of the Commissariat à l’Énergie Atomique. This vehicle, equipped with special instruments (two soil damping systems, and a system for contamination recovery in a wet environment), operates without resuspending dust. A 2500 litre tank for supplying the damping systems and a 6000 litre tank for collecting the effluents enable this vehicle to operate independently for 30 minutes over an area of 600 m². An experiment to test the equipment was carried out at the Bourges Technical Establishment site with a significant activity (60 GBq) of lanthanum-140 spread over a concrete surface of 600 m². The lanthanum-140, a short-lived gamma-emitting radioisotope (40.2 h), was deposited in water-insoluble form on sand with a grain size ranging from a few micrometres to 500 μm. In these experimental conditions the vehicle gave a decontamination efficiency of more than 90% with a single pass over the affected area, and no resuspension of dust was detected. The vehicle is well suited to the decontamination of roads or runways in the event of an accident.

DECONTAMINATION DES VOIES ROUTIERES PAR UN VEHICULE SPECIALISE.

Un véhicule de décontamination des routes a été développé en France par la Direction des applications militaires du Commissariat à l’énergie atomique. Ce véhicule doté d’équipements spécifiques (deux rampes de mouillage du sol, un système de récupération de la contamination en milieu humide) est capable d’opérer sans remise en suspension des poussières. Un réservoir de 2500 litres pour alimenter les rampes de mouillage et un de 6000 litres pour recevoir les effluents assurent à ce véhicule une autonomie de fonctionnement de 30 minutes pour une surface de 600 m². Une expérimentation de qualification a été réalisée sur le site de l’Etablissement technique de Bourges avec une activité significative (60 GBq) de lanthane 140 répandu sur une surface bétonnée de 600 m². Le lanthane 140, radio-isotope émetteur γ de période courte (40,2 h) était déposé sous forme insoluble dans l’eau sur du sable de granulométrie comprise entre quelques micromètres et 500 μm. Dans les conditions de cette expérimentation, ce véhicule est capable d’une efficacité de décontamination supérieure à 90%; un seul passage du véhicule permet d’atteindre ce résultat et aucune remise en suspension des poussières n’a été détectée. Ce véhicule est bien adapté à la décontamination des voies routières ou de pistes d’aviation en situation accidentelle.
Dans une situation accidentelle, la décontamination de certaines voies d'accès est souvent importante et doit être entreprise dès que possible afin de limiter les transferts de contamination liés au passage de véhicules et à la remise en suspension sur des surfaces relativement lisses.

Parmi différentes méthodes comme le brossage et l'aspiration à sec dont l'efficacité est limitée pour les particules fines, la méthode retenue repose sur le lavage à l'eau sous pression et la récupération totale de l'effluent dans la cuve d'un véhicule.

Le système a été réalisé en adaptant un véhicule industriel de nettoyage de routes au cas particulier d'une contamination radioactive, d'un produit toxique ou gênant.

Afin de qualifier ce véhicule et de déterminer ses performances, une expérimentation a été réalisée dans des conditions représentatives d'une contamination en utilisant le lanthane 140 comme traceur.

1. GENERALITES

Le véhicule, nommé camion d'assainissement des routes (CAR), comprend les équipements suivants:

- à l'avant du véhicule, une rampe de distribution d'eau basse pression équipée de gicleurs pour créer une atmosphère humide pendant le passage du véhicule (débit 20 L/min);
- à l'arrière, une rampe d'arrosage moyenne pression (20 bars, débit 80 L/min) et un ensemble d'aspiration (débit 400 L/min) sous flux d'air créé par une turbine;
- un réservoir d'eau de 2500 L pour alimenter les deux rampes;
- un réservoir de 6000 L pour recevoir les effluents;
- un vérin permet de faire basculer le réservoir de recueil des effluents afin de faciliter sa vidange;
- un réservoir d'additif (détergent) est relié au dispositif de projection par l'intermédiaire d'un doseur.

Les conditions d'utilisation sont les suivantes en phase d'assainissement:

- vitesse: 1,4 km/h;
- largeur de route assainie par passage: 2,5 m;
- autonomie: environ 30 min.
FIG. 1. Schéma du véhicule CAR: vue de profil.

FIG. 2. Schéma du véhicule CAR: vue de dessus.
3. QUALIFICATION DU VEHICULE

Elle a été réalisée à l’Etablissement technique de Bourges (ETBS).

3.1. Objectifs de l’expérimentation

Les points analysés au cours de cette expérience sont les suivants:
— déterminer l’efficacité d’assainissement sur une route;
— évaluer la remise en suspension pendant le passage du véhicule;
— tester l’isolement de la cabine du conducteur;
— vérifier les possibilités de décontamination rapide et complète du véhicule.

3.2. Type de route utilisé

La portion de route retenue pour l’expérimentation avait un revêtement en béton brut et comprenait de nombreuses parties fissurées et dégradées, afin de qualifier le véhicule dans une situation réaliste. La surface traitée a été de 600 m².

3.3. Épandage du contaminant

Le lanthane 140 est choisi comme traceur à cause de sa période relativement courte (40,2 h) et des facilités de détection gamma.

On utilise 60 GBq de lanthane 140 qui sont mis sur une forme insoluble et déposés sur 4 kg de sable de granulométrie connue (tableau I). Après épandage, l’activité surfacique est de l’ordre de 0,1 GBq·m⁻².

TABLEAU I. CARACTERISTIQUES DU MATERIAU CONTAMINE

<table>
<thead>
<tr>
<th>Granulométrie (µm)</th>
<th>Masse de sable (kg)</th>
<th>Activité du ¹⁴₀La (GBq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 - 50</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>100 - 200</td>
<td>1</td>
<td>60</td>
</tr>
<tr>
<td>300 - 500</td>
<td>2</td>
<td></td>
</tr>
</tbody>
</table>
3.4. Mesure du lanthane 140

Les mesures de contamination surfacique sont réalisées avec des appareils portatifs équipés de sondes bêta-gamma collimatées ainsi que des babylines.

Pour évaluer la remise en suspension, on utilise des appareils de prélèvements d’aérosols à filtre fixe:

- débit de 1500 L/min pour l’environnement du véhicule;
- débit de 5 L/min au niveau du chauffeur.

4. RESULTATS

4.1. Efficacité d’assainissement

Elle est déterminée à partir des mesures effectuées tous les cinq mètres avant et après passage du véhicule.

Après un premier passage, la moyenne des mesures est de $92 \pm 1\%$. Ainsi, un seul passage est suffisant.

Après un deuxième passage, l’activité supplémentaire récupérée est faible, de l’ordre de 3 à 4% de l’activité déposée.

En définitive, un seul passage est suffisant et permet d’obtenir une efficacité d’assainissement supérieure à 90%.
4.2. Remise en suspension

Les prélèvements d'air effectués à poste fixe près de la chaussée avant et après le passage du véhicule n'ont pas mis en évidence de perturbation apportée par le véhicule.

Dans les conditions de l'expérimentation, l'air de la cabine n'est pas contaminé.

L'air prélevé en sortie de la turbine d'aspiration a une contamination 10 fois supérieure au bruit de fond.

4.3. Décontamination de véhicule

Elle est effectuée avec un dispositif classique d'eau sous pression associé à un produit détergent.

Les débits de dose relevés à 10 centimètres du véhicule sont indiqués au tableau II.

5. CONCLUSIONS

Le CAR développé à partir d'un véhicule industriel a été qualifié dans un environnement réaliste pouvant être rencontré dans une situation accidentelle.

Un seul passage du véhicule sur une route même dégradée permet de récupérer 90% de l'activité déposée.

La cabine du conducteur apporte une protection efficace contre la contamination et le passage du véhicule ne modifie pas le phénomène de remise en suspension.

Au niveau décontamination du véhicule, il faut noter quelques points particuliers au niveau du radiateur, des buses d'aspiration et des pneumatiques.

En définitive, ce véhicule est bien adapté pour assainir des voies d'accès ainsi que des pistes d'aviation.
THE ACCIDENT IN THE SOUTHERN URALS
IN 1957
(Special Session)

Chairman

E. KUNZ
Czechoslovakia
РАДИАЦИОННАЯ АВАРИЯ НА ЮЖНОМ УРАЛЕ В 1957 ГОДУ И ЛИКВИДАЦИЯ ЕЕ ПОСЛЕДСТВИЙ

Б.В.НИКИПЕЛОВ, Е.И.МИКЕРИН, Г.Н.РОМАНОВ,
Д.А.СПИРИН, Ю.Б.ХОЛИНА
Государственный комитет по использованию атомной энергии СССР

Л.А.БУЛДАКОВ
Институт биофизики Минздрава СССР,
Москва,
Союз Советских Социалистических Республик

Abstract–Аннотация

THE RADIATION ACCIDENT IN THE SOUTHERN URALS IN 1957 AND THE CLEANUP MEASURES IMPLEMENTED.

Data are given on the causes and consequences of the radiation accident which took place in 1957 near the town of Chelyabinsk. The cleanup measures employed are examined.

РАДИАЦИОННАЯ АВАРИЯ НА ЮЖНОМ УРАЛЕ В 1957 ГОДУ И ЛИКВИДАЦИЯ ЕЕ ПОСЛЕДСТВИЙ.

Представлены данные о причинах и последствиях радиационной аварии, имевшей место в 1957 году вблизи г.Челябинск. Рассмотрены мероприятия, направленные на ликвидацию последствий аварии.

1. ВВЕДЕНИЕ

Крупная радиационная авария на Южном Урале, имевшая своим результатом радиоактивное загрязнение обширной территории и приведшая к необходимости осуществления экстренных и долгосрочных мер радиационной защиты населения, имела место в 1957 г. почти одновременно с аварией в Уиндскейле (Великобритания). Авария произошла на первом советском ядерном предприятии, расположенном вблизи г.Кыштым в Челябинской области и предназначенном для производства плутония в военных целях; предприятие включало в свою технологию уран-графитовые реакторы для наработки плутония и радиохимическое производство для его выделения.

Как всегда и всюду, новая технология требовала решения ряда сложных задач. Если и сегодня обработка и хранение образующихся радиоактивных отходов представляет собой не до конца решенную проблему, то
на первом этапе производства плутония одним из практически приемлемых путей обращения с отходами радиохимической технологии было длительное хранение их в металлических охлаждаемых емкостях, помещенных, в свою очередь, в бетонные оболочки. Выделяющееся при радиактивном распаде входящих в состав отходов радионуклидов тепло отводилось за счет охлаждения емкостей проточной водой.

Нарушение системы охлаждения вследствие коррозии и выхода из строя средств контроля в одной из емкостей объемом 300 м$^3$ обусловило саморазогрев хранившихся там 70–80 т высокоактивных отходов преимущеcственно в форме нитратно-ацетатных соединений. Испарение воды, осушение осадка и разогрев его до температуры 330–350°C привели 29 сентября 1957 г. в 16.00 по местному времени к взрыву содержимого емкости; мощность взрыва, подобного взрыву порохового заряда, оценена близкой к 70–100 т трINITротолуола.

### 2. ХАРАКТЕРИСТИКИ РАДИАЦИОННОЙ ОБСТАНОВКИ

Рассеянное при взрыве радиоактивное вещество характеризовалось преимущественным содержанием короткоживущих радионуклидов $^{144}$Ce, $^{144}$Pr, $^{95}$Zr, $^{95}$Nb (табл.1), однако, основную радиационную опасность на протяжении длительного времени после аварии представляло наличие в смеси долгоживущего $^{90}$Sr (2,7% от суммарной активности) в равновесии с дочерним его продуктом $^{90}$Y. Состав смеси радионуклидов близок к составу смеси продуктов ядерного деления, образующихся в ядерном реакторе, после примерно одногодичной выдержки ее, когда распадаются все наиболее короткоживущие нуклиды, кроме одного отношения. Технологический процесс радиохимического завода по переработке отходов предусматривал концентрирование этих отходов методом "щелочного осаждения", т. е. осаждения с помощью NaOH. При этом в осадок, который после растворения отправлялся на хранение, переходили практически все радионуклиды, кроме цезия, как растворимого элемента первой группы, который оставался в щелочном растворе и концентрировался затем отдельно.

Поэтому в смеси радионуклидов практически не было цезия, что не учитывалось зарубежными исследователями и вызвало впоследствии неверные выводы анализа, в частности, в публикациях Ж. Медведева и, в первую очередь, в оценке масштабов последствий.

Выпавшая смесь характеризовалась наличием гамма-излучения с суммарной энергией на момент образования следа 7,63 МэВ на один распад $^{90}$Sr (принятого в качестве "реперного" радионуклида вследствие его значительного периода полуразпада) и бета-излучения с примерно в 3 раза
ТАБЛИЦА I. РАДИОНУКЛИДНЫЙ СОСТАВ АВАРИЙНОГО ВЫБРОСА.

<table>
<thead>
<tr>
<th>Радионуклид</th>
<th>Период полураспада</th>
<th>Вид излучения</th>
<th>Вклад в активность смеси, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{89}$Sr</td>
<td>51 сут</td>
<td>$\beta, \gamma$</td>
<td>следы</td>
</tr>
<tr>
<td>$^{90}$Sr + $^{90}$Y</td>
<td>28,6 года</td>
<td>$\beta$</td>
<td>5,4</td>
</tr>
<tr>
<td>$^{95}$Zr + $^{95}$Nb</td>
<td>65 сут</td>
<td>$\beta, \gamma$</td>
<td>24,9</td>
</tr>
<tr>
<td>$^{106}$Ru + $^{106}$Rh</td>
<td>1 год</td>
<td>$\beta, \gamma$</td>
<td>3,7</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>30 лет</td>
<td>$\beta, \gamma$</td>
<td>0,036</td>
</tr>
<tr>
<td>$^{144}$Ce + $^{144}$Pr</td>
<td>284 сут</td>
<td>$\beta, \gamma$</td>
<td>66</td>
</tr>
<tr>
<td>$^{147}$Pm</td>
<td>2,6 года</td>
<td>$\beta, \gamma$</td>
<td>следы</td>
</tr>
<tr>
<td>$^{155}$Eu</td>
<td>5 лет</td>
<td>$\beta, \gamma$</td>
<td>следы</td>
</tr>
<tr>
<td>Pu</td>
<td></td>
<td>$\alpha$</td>
<td>следы</td>
</tr>
</tbody>
</table>

Рис.1. Радиоактивный распад смеси нуклидов на протяжении первых пяти лет существования Восточно-Уральского радиоактивного следа.
большей суммарной начальной энергией. Вследствие последующего радиоактивного распада короткоживущих гамма-излучающих нуклидов (рис.1) гамма-излучение смеси существенно снизилось, и в настоящее время радиоактивное загрязнение, обусловленное практически только $^{90}$Sr + $^{90}$Y, радиационно значимо лишь по бета-излучению.

Из хранившихся в емкости 20 МКи радиоактивного вещества около 2 МКи радиоактивности было поднято в воздух на высоту около 1000 м, образовав радиоактивное облако. Осаждение радиоактивного вещества из облака, перемещавшегося под действием ветра в северо-восточном направлении от предприятия, обусловило радиоактивное загрязнение обширной территории по пути следования облака в Челябинской, Свердловской и Тюменской областях. Образовавшийся след позднее получил наименование Восточно-Уральского радиоактивного следа.

До 1957 г. подобных случаев радиоактивного загрязнения обширных территорий в Советском Союзе не было. Трагичность сложившегося положения, требовавшего скорейших мер по ликвидации последствий аварии и защиты населения, усугублялась не только отсутствием практических навыков по обращению с авариями подобного рода, но и отсутствием научных представлений о поведении радиоактивных нуклидов в окружающей среде, путях и условиях облучения человека и живой природы, о степени возникшей радиационной опасности. Наука о радиоактивном загрязнении окружающей среды, определяющем пути и уровни облучения населения, еще только зарождалась, а отдельные первые результаты отечественных и зарубежных исследователей были засекречены и недоступны для использования на практике.

Длительное время не публиковались данные о последствиях для жизни и здоровья людей, перенесших атомную бомбардировку в Хиросиме и Нагасаки. Не были опубликованы наиболее важные радиационные параметры аварии 1957 года в Уиндскейле, связанные с выбросом $^{210}$Ро.

Ни одной страной до сих пор не опубликованы в полной мере данные о влиянии облучения на профессионалов. Не публиковались ранее и данные об аварии на Южном Урале.

Как и другим ядерным странам, нашему государству пришлось накапливать свой опыт по организации дозиметрии, получать свои данные о влиянии экстремальных и систематических облучений персонала, т. е. учиться на собственном, часто печальном опыте.

Традиционная засекреченность сведений о влиянии радиоактивности на живые организмы, о снижении отрицательных последствий радиационных инцидентов, обусловленная ядерным противостоянием государств, принесла значительный вред.

Своевременная публикация этих данных позволила бы широкой общественности объективно оценить реальные количественные границы
Рис. 2. Схема размещения Восточно-Уральского радиоактивного следа (М.1 : 300 000).
влияния облучения на здоровье человека и экосистемы и предотвратила бы развитие радиофобии в тех масштабах, с которыми мы встречаемся в настоящее время.

Тем не менее, в тех сложных условиях были предприняты правильные пути оценки радиационной опасности и способы защиты населения, которые вскоре были тесно увязаны с мерами по восстановлению нормальной производственной и жизнедеятельности населения на значительной площади радиоактивного следа. Кроме этого, волею судеб советские исследователи в свое распоряжение получили уникальную по экспериментальным возможностям территорию, на которой, в основном, получила свое развитие отечественная радиоэкология.

Формирование Восточно-Уральского радиоактивного следа в основном было закончено в процессе осаждения радиоактивного вещества из проходящего облака. В момент взрыва при достаточном постоянстве направления ветра на северо-северо-восток наиболее вероятная скорость ветра на высоте перемещения облака составляла около 7,5 м/с, и время начала осаждения радиоактивного вещества в любой рассматриваемой точке определялось соотношением удаления ее от источника и средней скоростью движения облака. На расстоянии 100 км след сформировался в течение 4 ч после взрыва, в пределах минимально обнаруживаемых уровней загрязнения (максимальное удаление около 300 км) — за 11 ч. Продолжительность выпадений составляла от нескольких минут в начальной части следа до получаса-часа в наиболее удаленной его части.

Вследствие отсутствия атмосферных осадков в период образования следа, а также наличия отдельных периодов сухой погоды и сильных ветров до наступления постоянного осеннего ненастья и установления устойчивого снежного покрова, в течение первых 1–1,5 месяцев наблюдалось перераспределение радиоактивного вещества на местности под действием ветрового подъема, что привело к изменению плотности радиоактивного загрязнения на участках территории, прилегающих к головной части следа, где уровни радиоактивного загрязнения были максимальны. Поэтому поперечный размер следа в начальной его части больше, чем в остальной, и след здесь "размыт" в восточном направлении.

В границах плотности загрязнения 0,1 Ки/км² по 90Sr (минимально детектируемый уровень, равный удвоенному уровню глобального радиоактивного загрязнения территории 90Sr для данного региона в 1957 г.) максимальная длина образовавшегося следа достигала 300 км (вблизи г.Тюмень) (рис. 2) при ширине 30–50 км, в границах 2 Ки/км² по 90Sr — 105 км при ширине следа 8–9 км. Плотность загрязнения территории 2 Ки/км² по 90Sr была признана предельной для безопасного проживания населения и принята в качестве официальной границы Восточно-Уральского радиоактивного следа. Общая площадь территории, подвергшейся
ТАБЛИЦА II. РАСПРЕДЕЛЕНИЕ ПЛОЩАДИ ЗАГРЯЗНЕННОЙ ТЕРРИТОРИИ ПО ПЛОТНОСТИ ЗАГРЯЗНЕНИЯ

<table>
<thead>
<tr>
<th>Плотность загрязнения, Ки/км² по ⁹⁰Sr</th>
<th>Площадь территории, км²</th>
</tr>
</thead>
<tbody>
<tr>
<td>0,1–2</td>
<td>15 000</td>
</tr>
<tr>
<td>2–20</td>
<td>600</td>
</tr>
<tr>
<td>20–100</td>
<td>280</td>
</tr>
<tr>
<td>100–1000</td>
<td>100</td>
</tr>
<tr>
<td>1000–4000</td>
<td>17</td>
</tr>
</tbody>
</table>

Примечание. Границы 0,1 Ки/км² по ⁹⁰Sr определены с малой достоверностью.

радиоактивному загрязнению, составляет около 15 000 км², в т. ч. в границах 2 Ки/км² по ⁹⁰Sr — около 1000 км².

Восточно-Уральский радиоактивный след имеет достаточно закономерное распределение загрязнения на территории, а именно, выраженную ось, вдоль которой плотность загрязнения монотонно убывает (от 4000 Ки/км² по ⁹⁰Sr в головной части следа до 0,1 Ки/км² на наибольшем удалении). В поперечных направлениях плотность загрязнения резко убывает к периферии; распределение плотности загрязнения в поперечных направлениях характеризуется резко выраженным максимумом на оси следа, превосходящим периферийные плотности загрязнения на 1—4 порядка величины. Распределение площади загрязненной территории по плотности загрязнения ⁹⁰Sr приведено в табл. II.

Выпавшее радиоактивное вещество в начальный период не было закрепленным в окружающей среде, его присутствие обнаружилось во всех без исключения объектах окружающей среды, включая живые организмы и продовольственные продукты. Проникновение его в объекты окружающей среды усиливалось ветровой миграцией, механическими воздействиями и деятельностью человека. В зависимости от места расположения относительно источника загрязнения начальные уровни радиоактивного загрязнения по суммарной бета-активности по сравнению с предварительным периодом возросли в естественной траве в $10^2 - 2 \times 10^5$ раз, воде открытых водоемов — в $1,5 - 3 \times 10^4$ раз, зерне пшеницы — в 25–1000 раз, в молоке коров — в 10–2000 раз. Средние начальные концентрации суммы радионуклидов в расчете на единичную плотность загрязнения территории (на 1 Ки/км² по стронции-90) составляли: в траве — 14,
зере — 0,22, молоке коров — 0,0062 мкКи/кг. Основным путем поступления радиоактивности в растительную продукцию являлось непосредственное поверхностное загрязнение.

В начальные сроки существования следа мощность экспозиционной дозы гамма-излучения на открытом месте на высоте 1 м составляла 150 мкР/ч в расчете на 1 Ки/км² по стронцию-90, из этой величины около 90% приходилось на вклад 95Zr + 95Nb. При максимальной плотности загрязнения около 4000 Ки/км² по стронцию-90 начальная мощность экспозиционной дозы гамма-излучения достигала 0,6 Р/ч.

В последующее время существования Восточно-Уральского радиоактивного следа радиационная обстановка на его территории претерпела значительные изменения и стала более благоприятной с точки зрения радиационной опасности для человека и природных объектов. Основными факторами, которые влияли и влияют на радиационную обстановку на территории следа, являются следующие:

— радиоактивный распад короткоживущих гамма-излучающих нуклидов;
— перераспределение радиоактивного вещества в природных системах, в том числе заглубление в почве и донных отложениях;
— биогеохимическая миграция радионуклидов;
— хозяйственная деятельность человека, включая мероприятия по радиационной защите населения.

Общая динамика радиационной обстановки на территории Восточно-Уральского радиоактивного следа представлена в табл. III.

Под действием радиоактивного распада плотность загрязнения территории по смеси радионуклидов за 30 лет снизилась более чем в 30 раз, по стронцию-90 — в 2 раза. По этой же причине энергия гамма-излучения упала с 7,6 до 0,004 МэВ на 1 распад 90Sr, что привело к снижению мощности экспозиционной дозы гамма-излучения на высоте 1 м (с учетом заглубления в почве) в 2800 раз. Экспозиционная доза гамма-излучения, составляющая за 30 лет 0,5 Р в расчете на 1 Ки/км² по 90Sr и сформированная практически за первый год после аварии, за последующее время возросла всего лишь на 16%. Это означает, что гамма-облучение человека и живой природы на территории Восточно-Уральского радиоактивного следа было превалирующим только на протяжении первых одного-полутора лет. Концентрация суммы радионуклидов в различных объектах окружающей среды, в том числе в сельскохозяйственной продукции, снизилась за это же время в сотни-тысячи раз, причем максимальное снижение произошло в первые пять лет. В последующее время радиоактивное загрязнение всех без исключения объектов живой и неживой природы стало обусловлено только 90Sr, и последующее снижение уровней радио-
ТАБЛИЦА III. ДИНАМИКА РАДИАЦИОННОЙ ОБСТАНОВКИ НА ТЕРРИТОРИИ ВОСТОЧНО-УРАЛЬСКОГО РАДИОАКТИВНОГО СЛЕДА

<table>
<thead>
<tr>
<th>Показатель радиационной обстановки</th>
<th>Время после аварии, год</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0</td>
</tr>
<tr>
<td>Плотность радиоактивного загрязнения территории, относ.ед. по суммарной</td>
<td>1</td>
</tr>
<tr>
<td>β-активности по Sr⁹₀</td>
<td>1</td>
</tr>
<tr>
<td>Мощность экспозиционной дозы γ-излучения на высоте 1 м, мкР/ч</td>
<td>150</td>
</tr>
<tr>
<td>Экспозиционная доза γ-излучения на высоте 1 м, Ки Sr/км²</td>
<td>0</td>
</tr>
<tr>
<td>Концентрация суммы радионуклидов, относ. единицы:</td>
<td>1</td>
</tr>
<tr>
<td>трава</td>
<td>1</td>
</tr>
<tr>
<td>зерно</td>
<td>1</td>
</tr>
<tr>
<td>молоко</td>
<td>1</td>
</tr>
<tr>
<td>вода озер</td>
<td></td>
</tr>
</tbody>
</table>

активного загрязнения было обусловлено закономерностями поведения Sr⁹₀ в окружающей среде.
В данном обзоре мы не будем касаться ликвидации последствий аварии непосредственно в районе выброса, это проводилось профессионалами на неиспользуемойся в хозяйственной деятельности нежилой территории с применением специальных средств. Остановимся на территории, находящейся за пределами промышленной зоны.
Загрязненная территория представляет участок Зауральской лесостепи с типичным для этой зоны относительно ровным рельефом. Ландшафт территории представлен двумя типами растительных формаций — травянистой и лесной, а также водоемами, главным образом, мелководными озерами с атмосферным питанием.

Почвенный покров территории следа характеризуется неоднородностью почв. Наиболее распространены на следе серые лесные почвы, чернозем выщелоченный и дерново-подзолистые почвы с включением избыточно-влаженных и засоленных почв.

При рассмотрении вопросов о поведении и миграции радиоактивных нуклидов в окружающей среде и пищевых цепях применительно к данной аварийной ситуации необходимо выделять две временные фазы: фазу начального распределения радиоактивного вещества в ландшафтах или рассматриваемых системах и фазу последующего перераспределения радионуклидов под действием природных (в т. ч. биогенных) и антропогенных факторов. Процессы перераспределения осевшего радиоактивного вещества под действием природных сил начались непосредственно сразу после загрязнения, и их вертикальная составляющая, обусловленная массопереносом, обусловила последующее перемещение радиоактивного вещества с крон деревьев и травянистого покрова на поверхность почвы, в озерах — из массы воды в донные отложения. С течением времени почва и донные отложения стали основными аккумуляторами радиоактивности, и долговременное поведение радионуклидов в различных природных системах явилось отражением роли этих депо в общей подвижности радионуклидов в природе.

В лесонасаждениях, при первоначальном задерживании выпавшей радиоактивности в пределах 80–90% в кронах деревьев, процесссы нисходящей миграции начались тоже сразу непосредственно после загрязнения. Эти процессы были обусловлены действием ветра, атмосферных осадков, опадом листвьев (береза) и хвои (сосна). Смыв и выдувание задержанного радиоактивного вещества в течение первых нескольких месяцев привели к снижению содержания активности в кронах на 20–40%. В результате осеннего листопада на лесную подстилку переместилось около 80% содержащейся в кронах активности, в надземной биомассе деревьев березы содержалось 10–20% общего запаса радиоактивного вещества на единицу площади, деревьев сосны — 40–60%. В последующее время основными факторами перераспределения радионуклидов в лесных биогеоценозах стали биогенная миграция (усвоение радионуклидов корневой системой из подстилки и почвы, опад хвои, листвьев, мелких веток, травянистой расти-
Рис. 3. Динамика распределения радиоактивного вещества в вертикальном профиле почв разных типов.
Низходящая миграция радионуклидов в системе подстилка-минерализованная часть почвы, а также резко снижающееся в первые годы внекорневое загрязнение под действием ветрового подъема. Через 8 лет после выпадений в подстилке оставалось 10% от всего запаса $^{90}$Sr, а через 30 лет — уже 3-4%. В минерализованной части почвы запас $^{90}$Sr постепенно возрастал — до 75% через 5 лет и 95% через 30 лет.

Из-за вышеописанных особенностей поведения радионуклидов в течение первых 10 лет увеличивался запас $^{90}$Sr в древесине и коре деревьев. В последующие 20-25 лет поступление $^{90}$Sr в деревья практически полностью было обусловлено корневым путем, медленно снижаясь из года в год в листьях и ветках и нарастая в древесине и коре. Отмечено, что поступление $^{137}$Cs в древесные растения в целом на порядок величины меньше, чем $^{90}$Sr.

На территории Восточно-Уральского радиоактивного следа размещены ряд непроточных озер степного типа и три небольшие реки. Основными источниками радиоактивного загрязнения поверхностных водоемов явилось начальное осаждение радиоактивного вещества на водную поверхность и последующий многолетний поверхностный сток радиоактивности с водосборной площади. До достижения динамического равновесия в распределении радионуклидов в озерных гидроценозах происходило достаточно быстрое самоочищение воды. Период полуочищения воды составлял 120-190 сут для $^{90}$Sr, 18-110 сут для $^{106}$Ru, 1-24 сут для $^{144}$Ce. Вследствие малого запаса биомассы в озерах дальнейшее распределение содержания радионуклидов в озерах практически полностью зависело от характера взаимодействия воды с донными отложениями. Снижение концентрации радиоактивных веществ в воде озер протекает с периодом полууменьшения около 5-6 лет. $^{90}$Sr за этот период переместился в илах на глубину более 30 см, сосредоточившись в основном в слое 0-15 см.

Процессы водного стока и ветрового подъема (дефляции) являлись доминирующими среди всех природных процессов миграции радиоактивного вещества в плоскости земной коры (рис. 3).

В момент образования следа интенсивность ветрового подъема $\alpha$ была оценена равной $10^{-9}$ с$^{-1}$, в последующее время отмечено ее экспоненциальное снижение с периодом полууменьшения до 4 лет. Установленное максимальное значение $\alpha$ для весенне-летнего-осеннего периода составляет $10^{-11}$ с$^{-1}$.

Водная и ветровая миграция радиоактивного вещества не привела к дезактивации территории, изменению макроструктуры плотности загрязнения территории, смещению оси и границ следа. Суммарный эффект этих факторов в перераспределении активности на территории следа оце-
нен в 1–2% от запаса вещества на площади в начальный период и долями процента на протяжении всего последующего периода.

4. ДОЗЫ ОБЛУЧЕНИЯ НАСЕЛЕНИЯ

Облучение населения на территории образовавшегося Восточно-Уральского радиоактивного следа общей численностью около 270 тыс. человек было обусловлено несколькими путями.

Внешнее облучение (облучение всего тела и внутренних органов) было обусловлено преимущественно гамма-излучением и состояло из следующих компонентов:

(a) облучение от проходящего облака выброса;
(b) облучение от загрязненной почвы и среды обитания;
(c) облучение от загрязненной поверхности тела и одежды.

Внутреннее облучение было обусловлено поступлением радиоактивных веществ в организм с дыхаемым воздухом, пищевыми продуктами и питьевой водой и результатирующим кратковременным или долговременным (при отложении 90Sr в организме) пребыванием радионуклидов в соответствующих тканях и органах человека. В момент образования и начальный период существования следа действовали все указанные пути облучения при преобладании внешнего облучения, в последующее время превалирующим стало внутреннее облучение, обусловленное поступлением 90Sr с пищевым рационом и отложением его в скелете человека. В этой связи долговременное, на протяжении 30 лет, формирование доз облучения населения целесообразно соотнести с 2 периодами — “острым” или начальным протяженностью 1–1,5 года с преимущественно внешним облучением и поздним с преимущественно внутренним облучением. Внешнее облучение за время прохождения облака оценено равным 0,13 мбэр/(Ки 90Sr/км²), что соответствует максимальным дозам на население трех наиболее близко расположенных к предприятию деревень около 100–130 мбэр. Внутреннее облучение легких, обусловленное вдыханием радиоактивного вещества из облака, за все время последующего нахождения активности в легких оценено в пределах 5–300 мбэр/(Ки 90Sr/км²) в зависимости от степени растворимости радиоактивного вещества в легочной жидкости, что дает потенциальную максимальную дозу за этот период применительно к населению указанных пунктов в пределах 4–300 бэр. Учитывая наличие преимущественно растворимых форм солей в составе выброса, следует допустить возможными минимальные дозы.

На протяжении “острого” периода населением была получена основная доля дозы внешнего облучения (табл. IV). Из общей дозы за 30
ТАБЛИЦА IV. СРЕДНИЕ ДОЗЫ ОБЛУЧЕНИЯ НАСЕЛЕНИЯ НА ПРОТЯЖЕНИИ ВСЕГО ПЕРИОДА СУЩЕСТВОВАНИЯ ВОСТОЧНО-УРАЛЬСКОГО РАДИОАКТИВНОГО СЛЕДА, мБр/(Ки ⁹⁰Sr/км²)

<table>
<thead>
<tr>
<th>Путь облучения.</th>
<th>10 сут</th>
<th>30 сут</th>
<th>120 сут</th>
<th>1 год</th>
<th>2 года</th>
<th>5 лет</th>
<th>10 лет</th>
<th>15 лет</th>
<th>20 лет</th>
<th>30 лет</th>
</tr>
</thead>
<tbody>
<tr>
<td>Эквивалентная доза</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Внешнее облучение</td>
<td>33</td>
<td>67</td>
<td>160</td>
<td>220</td>
<td>230</td>
<td>250</td>
<td>260</td>
<td>260</td>
<td>260</td>
<td>260</td>
</tr>
<tr>
<td>Внутреннее облучение:</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>кость</td>
<td>3,1</td>
<td>9,1</td>
<td>46</td>
<td>720</td>
<td>1500</td>
<td>3300</td>
<td>5400</td>
<td>6600</td>
<td>7300</td>
<td>8000</td>
</tr>
<tr>
<td>красный костный мозг</td>
<td>0,97</td>
<td>2,9</td>
<td>14</td>
<td>220</td>
<td>490</td>
<td>1000</td>
<td>1700</td>
<td>2000</td>
<td>2300</td>
<td>2500</td>
</tr>
<tr>
<td>Эффективная эквивалентная доза</td>
<td>93</td>
<td>180</td>
<td>450</td>
<td>700</td>
<td>780</td>
<td>920</td>
<td>1100</td>
<td>1100</td>
<td>1200</td>
<td>1200</td>
</tr>
</tbody>
</table>
лет, равной 260 мбэр/(Ки \(^{90}Sr/\text{км}^2\)), более половины дозы (160 мбэр) было получено в первые 120 сут, около 90% — в первые 2 года. Наиболее критическими органами, получившими среди всех других максимальные дозы внутреннего облучения на протяжении "острого" периода, явились отделы желудочно-кишечного тракта. Из дозы облучения желудочно-кишечного тракта за 30 лет, равной 2 бэр/(Ки \(^{90}Sr/\text{км}^2\)), 12% было получено в течение первых 10 сут, а 80% — в течение первого года. В "острый" период также нарастала доза на kostную ткань и красный костный мозг за счет отложения \(^{90}Sr\) в кости: значения поглощенных в этих тканях доз возросли, соответственно, от 9 и 3 за первый месяц до 720 и 220 мбэр/(Ки \(^{90}Sr/\text{км}^2\)) по истечении первого года.

Ведущая роль \(^{90}Sr\) в формировании дозы внутреннего облучения населения и в значительной мере обусловленной этим облучением эффективной эквивалентной дозы проявилась, начиная со второго года после аварии. Вследствие преимущественного обеспечения сельского населения пищевыми продуктами местного происхождения, в первую очередь за счет ведения личных хозяйств, поступление \(^{90}Sr\) с пищевым рационом было обусловлено потреблением производимых на месте молока, мяса, картофеля и овощей. Вклад \(^{90}Sr\) в поступление с молоком составлял 50–70%, с мясом — 5–25%, с картофелем и овощами — 15–45%. На протяжении всего позднего периода это соотношение поступления не претерпело больших изменений, однако принятые меры радиационной защиты населения, а также природные процессы, влияющие на характер доступности \(^{90}Sr\) для растений, привели к систематическому снижению уровней загрязнения продукции и, в конечном счете, к постоянному уменьшению

### ТАБЛИЦА V. ГОДОВОЕ ПОСТУПЛЕНИЕ \(^{90}Sr\) В ОРГАНИЗМ ЧЕЛОВЕКА С ПИЩЕВЫМ РАЦИОНОМ

<table>
<thead>
<tr>
<th>Время после образования следа, год</th>
<th>Годовое поступление, (\text{мкКи/год}^{90}Sr/\text{км}^2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>11</td>
</tr>
<tr>
<td>1</td>
<td>1,4</td>
</tr>
<tr>
<td>2</td>
<td>0,9</td>
</tr>
<tr>
<td>5</td>
<td>0,2</td>
</tr>
<tr>
<td>10</td>
<td>0,06</td>
</tr>
<tr>
<td>15</td>
<td>0,03</td>
</tr>
<tr>
<td>20</td>
<td>0,015</td>
</tr>
<tr>
<td>30</td>
<td>0,010</td>
</tr>
</tbody>
</table>
годового поступления $^{90}\text{Sr}$ с пищевым рационом в организм населения (табл. V). Содержание $^{90}\text{Sr}$ в пищевом рационе населения убывает в два раза каждые 5,5 лет. Вследствие этих причин интенсивность поступления $^{90}\text{Sr}$ в организм и отложение его в скелете с течением времени снижались, что привело к снижению темпов нарастания дозы на скелет и красивый костный мозг в последние 15-20 лет. За 30 лет доза на костную ткань, в расчете на 1 Ки/км$^2$ по $^{90}\text{Sr}$, достигла 8 бэр, на красивый костный мозг — 2,5 бэр; половина каждой из этих доз была сформирована за первые 6-7 лет. За 30 лет эффективная эквивалентная доза достигла 1,2 бэр/(Ки $^{90}\text{Sr}$/км$^2$); на долю внешнего облучения приходится 22% этой дозы, внутреннего облучения костной ткани — 21%, красивого костного мозга — 28%.

5. МЕРЫ РАДИАЦИОННОЙ ЗАЩИТЫ НАСЕЛЕНИЯ

Несмотря на отсутствие опыта с радиационными авариями, связанными с интенсивным радиоактивным загрязнением больших территорий, стратегия и тактика экстренных, а затем и плановых мер радиационной защиты населения на территории Восточно-Уральского радиоактивного следа оказались правильными и с современной точки зрения.

Основные экстренные меры, которые были предприняты незамедлительно вскоре после образования Восточно-Уральского радиоактивного следа, включали в себя.

1. Эвакуацию населения из близлежащих населенных пунктов, где потенциальная доза внешнего облучения за первый месяц могла превысить 100 бэр.

2. Санитарную обработку эвакуированного населения с заменой личной одежды, введение запрета на вывоз личного имущества и имевшихся запасов продовольствия этим контингентом населения.

3. Введение радиационного и дозиметрического контроля на наиболее загрязненной части территории с одновременным ограничением контролируемого доступа на эту часть территории.

Экстренная эвакуация населения (которая впоследствии явилась практически отселением его) была проведена в четырех наиболее близких к предприятию деревнях с общей численностью населения около 1100 человек. Эвакуация была завершена в течение первых 10 суток. Население было вывезено и размещено в незагрязненных населенных пунктах, обеспечено жильем и работой.

Дальнейшие, теперь уже планомерные мероприятия по снижению уровней облучения населения на протяжении “острого” периода включали:
1. Контроль за уровнями радиоактивного загрязнения продовольствия и сельскохозяйственной продукции, выбраковку продукции с уровнями загрязнения свыше допустимых и обеспечение населения незагрязненным продовольствием взамен выбракованного.
2. Дополнительную эвакуацию населения.
3. Введение режима ограничения на доступ населения и хозяйственно-деятельность на части территории следа, признанной небезопасной для проживания.
4. Дезактивацию населенных пунктов и сельскохозяйственной территории.

Введение первой меры диктовалось необходимостью первоочередного снижения поступления радиоактивности в организм людей с пищевым рационом. Альтернативное решение этого вопроса — систематическое и долгосрочное обеспечение населения доставляемым из других районов незагрязненным продовольствием — было нереальным, так как при запрещении производства и потребления сельскохозяйственной продукции проживание населения в сельской местности теряет свой смысл. На основе разработанного временного норматива — допустимого годового поступления $^{90}\text{Sr}$ 1,4 мкКи/год — была признана необходимость проведения сплошного радиационного контроля с соответствующей выбраковкой продукции. Контролем была охвачена территория с минимальной плотностью загрязнения 0,5—1 Ки/км$^2$ по $^{90}\text{Sr}$ на площади около 1000 км$^2$ (50 населенных пунктов). За первые два года было выбраковано более 10 000 т различной продукции.

Вследствие невозможности полной замены загрязненного продовольствия на "чистое" в населенных пунктах с плотностью загрязнения свыше 10 Ки/км$^2$ по $^{90}\text{Sr}$, где наблюдалось превышение временного допустимого годового поступления $^{90}\text{Sr}$ с пищевым рационом, было принято решение о плановом дополнительном отселении населения с территории при плотности загрязнения свыше 4 Ки/км$^2$ по $^{90}\text{Sr}$. Очередность отселения устанавливали в зависимости от плотности загрязнения территории и степени хозяйственного использования прилегающей территории. Отселение было начато через 8 месяцев и закончено через 1,5 года после образования следа. Всего, вместе с экстренным отселением, было переселено более 10 тыс. человек из 23 населенных пунктов сельского типа (табл. VI). Экстренное отселение позволило снизить потенциальную дозу внешнего облучения за 30 лет в 77 раз, дозу облучения кости и красного костного мозга — практически в 500 раз, эффективную эквивалентную дозу — почти в 100 раз. Плановое отселение, в зависимости от увеличения сроков его осуществления, предотвратило потенциальную дозу за 30 лет по внешнему облучению всего лишь примерно на 20%, но по облучению кости и красного костного мозга — на 40—90%.
ТАБЛИЦА VI. ДИНАМИКА ОТСЕЛЕНИЯ И ЕГО РОЛЬ В СНИЖЕНИИ УРОВНЕЙ ОБЛУЧЕНИЯ НАСЕЛЕНИЯ*

<table>
<thead>
<tr>
<th>Показатель</th>
<th>I</th>
<th>II</th>
<th>III</th>
<th>IV</th>
<th>V</th>
<th>Всего</th>
</tr>
</thead>
<tbody>
<tr>
<td>Число населенных пунктов</td>
<td>4</td>
<td>1</td>
<td>5</td>
<td>7</td>
<td>6</td>
<td>23</td>
</tr>
<tr>
<td>Численность, тыс. чел.</td>
<td>1,1</td>
<td>0,3</td>
<td>2,0</td>
<td>4,2</td>
<td>3,1</td>
<td>10,7</td>
</tr>
<tr>
<td>Средняя плотность загрязнения, Ки 50Sr/км²</td>
<td>500</td>
<td>65</td>
<td>18</td>
<td>8,9</td>
<td>3,3</td>
<td></td>
</tr>
<tr>
<td>Продолжительность проживания до отселения, сут</td>
<td>10</td>
<td>250</td>
<td>250</td>
<td>330</td>
<td>670</td>
<td></td>
</tr>
<tr>
<td>Дозы облучения, полученные до отселения, бэр:</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Эквивалентная доза</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>внешнее облучение</td>
<td>17</td>
<td>14</td>
<td>3,9</td>
<td>1,9</td>
<td>0,68</td>
<td></td>
</tr>
<tr>
<td>внутреннее облучение</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>желудочно-кишечный тракт</td>
<td>150</td>
<td>98</td>
<td>27</td>
<td>13</td>
<td>5,4</td>
<td></td>
</tr>
<tr>
<td>кость</td>
<td>1,6</td>
<td>10</td>
<td>2,8</td>
<td>5,8</td>
<td>4,4</td>
<td></td>
</tr>
<tr>
<td>Эффективная эквивалентная доза</td>
<td>53</td>
<td>14</td>
<td>12</td>
<td>5,6</td>
<td>2,3</td>
<td></td>
</tr>
<tr>
<td>Доля потенциальной дозы за 30 лет, предотвращенная отселением, относ. единицы:</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>доза внешнего облучения</td>
<td>0,87</td>
<td>0,18</td>
<td>0,17</td>
<td>0,17</td>
<td>0,21</td>
<td></td>
</tr>
<tr>
<td>доза на кость</td>
<td>0,998</td>
<td>0,88</td>
<td>0,89</td>
<td>0,49</td>
<td>0,40</td>
<td></td>
</tr>
<tr>
<td>эффективная эквивалентная доза</td>
<td>0,91</td>
<td>0,38</td>
<td>0,40</td>
<td>0,28</td>
<td>0,20</td>
<td></td>
</tr>
</tbody>
</table>

* Максимальные дозы облучения примерно в два раза больше вследствие двухкратного превышения максимальных плотностей загрязнения жилых и сельскохозяйственных территорий над средними.
Режим ограничения на территории Восточно-Уральского радиоактивного следа был введен в границах плотности загрязнения 4 Ки/км² по ⁹⁰Sr в отдаленной части следа и 2 Ки/км² по ⁹⁰Sr в промежуточной и головной его частях путем создания санитарно-охранной зоны площадью около 700 км², охраняемой силами милиции. На территории зоны вплоть до 1961 года была исключена любая хозяйственная деятельность. Впоследствии площадь санитарно-охранной зоны была сокращена до 200 км².

Дезактивацию осуществляли силами специальных механизированных отрядов. Ее цель состояла в предотвращении повторного использования территории и строений эвакуированных населенных пунктов, а также в снижении уровней излучений и ветрового подъема радиоактивного вещества на сельскохозяйственных угодьях. Дезактивация сельскохозяйственных угодий была основана на перепашке их обычными плугами. В течение первых 1,5 лет перепашке было подвергнуто около 20 тыс. га в головной и промежуточной частях следа.

Меры радиационной защиты населения, осуществленные в ходе позднего периода, были направлены, главным образом, на снижение поступления ⁹⁰Sr в организм населения за счет уменьшения уровней радиоактивного загрязнения продукции, производимой в личных и общественных хозяйствах. Последний комплекс мероприятий включал также восстановление хозяйственной деятельности на территории следа при одновременной реорганизации сельскохозяйственного производства и других отраслей хозяйства.

Так как облучаемой частью незакуированного населения (критические группы населения) явилось население, проживающее на прилегающей к следу территории с плотностью загрязнения не свыше 2 Ки/км² по ⁹⁰Sr и получающее основные продукты питания преимущественно из своих личных хозяйств, то весь комплекс мероприятий по защите этих групп населения был направлен на снижение уровней радиоактивного загрязнения прежде всего молока, представляющего собой компонент пищевого рациона с наибольшим вкладом ⁹⁰Sr. Основным содержанием этих мер должно было быть по возможности полное исключение из использования естественных пастбищ и сенокосов, однако при учете необходимости создания нормальных условий для ведения личных хозяйств в населенных пунктах вблизи границы санитарно-охранной зоны и снижения стремления населения использовать территорию зоны в качестве кормовой базы.

Комплекс осуществленных организационно-технических мероприятий включал в себя:

— упорядочение распределения естественных сенокосных угодий и целевое выделение наименее загрязненных пастбищ для личного скота;
— создание окультуренных пастбищ и сенокосов на части территории;
— частичное обеспечение потребностей населения в кормах продукцией с пахотных угодий, производимой совхозами;
— создание специализированных совхозов.

Для реализации этих мер были разработаны и внедрены в практику соответствующие рекомендательно-нормативные документы, в соответствии с которыми упорядочивалась жизнедеятельность населения и реорганизовывалось сельскохозяйственное производство и лесное хозяйство.

Комплекс описанных мер позволил в определенной мере снизить уровни поступления ⁹⁰Sr в организм населения, особенно на протяжении первых 10 лет после аварии. При учете реальной максимальной плотности загрязнения территории проживания населения 1–2 Ки/км² по ⁹⁰Sr предел годового поступления ⁹⁰Sr в организм, устанавливаемый современными Нормами радиационной безопасности НРБ–76/87 и равный 0,32 мкКи/год, превышался по отношению к части незавуированного населения на протяжении первых 4 лет, в настоящее время годовое поступление ⁹⁰Sr в организм этой группы населения составляет в среднем около 3% от предела годового поступления с максимальным значением 12%.

6. ПОТЕНЦИАЛЬНЫЕ РАДИОЛОГИЧЕСКИЕ ПОСЛЕДСТВИЯ

Облучение населения в результате образования Восточно-Уральского радиоактивного следа могло иметь своими последствиями определенные эффекты в состоянии здоровья населения применительно и к эвакуированной его части, и к части, оставшейся проживать на территории с малой плотностью радиоактивного загрязнения. Предваряя рассмотрение фактических данных о состоянии здоровья населения, целесообразно провести оценку предполагаемых радиационных эффектов на основе оценки доз облучения населения и имеющихся данных о риске проявления ранних (от облучения в “острый” период) и поздних (от хронического облучения на протяжении 30 лет) радиационных эффектов.

Оценка потенциальных последствий на основе данных табл. IV и VI и публикаций МКРЗ №№ 26, 27 привела к следующим выводам.

В качестве предела эффективной эквивалентной дозы, который полностью исключает появление нестохастических соматических эффектов, МКРЗ предлагает 50 бэр/год; пороговые дозы, при которых возникают нестохастические проявления в органах и тканях человека, составляют сотни–тысячи рад. Исключением являются кровь и семенники, реагирующие на облучение в 10–100 рад (сдвиг в количестве лимфоцитов в пери-
ферической крови и красном костном мозге и временная стерильность семенников). 

Это означает, что для условий радиационного воздействия на население в результате образования радиоактивного следа вряд ли можно ожидать проявления клинических соматических эффектов, кроме изменений в крови, так как максимальная эффективная эквивалентная доза для более критической группы эвакуированного населения не превышала 100 бэр.

Риск проявления стохастических канцерогенных эффектов после разового облучения определяется в виде максимальной вероятности проявления злокачественных опухолей для наиболее радиочувствительных тканей и лейкемии у 1 млн. облученных на 1 бэр в течение всей оставшейся продолжительности жизни после облучения; этот риск в среднем равен $105 \times 10^{-6}$ бэр$^{-1}$ для рака и $20 \times 10^{-6}$ бэр$^{-1}$ для лейкемии.

Вследствие наличия латентного периода проявления рака (20–25 лет) и лейкемии (10 лет) уровень их риска уменьшается при достижении определенного возраста. На этой основе средний относительный риск потенциальных канцерогенных эффектов, вызванных облучением и существованием следа, будет меньше максимального и будет определяться длительностью облучения и изменением половозрастной структуры облучаемого контингента. Поэтому средневзвешенные по возрастным группам коэффициенты риска всех проявлений рака и лейкемии меньше максимального риска на 21%. С учетом хронического облучения и смены популяционных групп во времени вследствие естественной смертности откорректированные оценки риска проявления рака различной локализации и лейкемии, приведенные в табл. VII, показывают, что максимальный риск рака и лейкемии у эвакуированной части населения (коллективная эффективная эквивалентная доза 144 000 чел.-бэр) составляет, соответственно, $6,8 \times 10^{-4}$ и $1,8 \times 10^{-4}$ случаев/чел.-год, у незавакуированного населения (коллективная эффективная эквивалентная доза 112 000 чел.-бэр) — $3,8 \times 10^{-5}$ и $1,1 \times 10^{-5}$ случаев/чел.-год. В расчете на всю облучаемую популяцию численностью 270 тыс. человек это означает потенциальную возможность возникновения 18 случаев рака и 5 случаев лейкемии на протяжении 70 лет существования следа. Если предположить наличие летальных исходов всех канцерогенных проявлений и их равномерное распределение на протяжении оставшихся 30 лет жизни при среднем возрасте населения 40 лет, то максимальная смертность от рака и лейкемии для всей облучаемой популяции не должна превысить, соответственно, $1,5 \times 10^{-6}$ и $4,0 \times 10^{-7}$ случаев/(чел.-год). Эти оценки предполагаемой смертности от радиационно индуцированных рака и лейкемии значительно ниже частоты реальной смертности от спонтанного рака: в течение 1970–1987 гг. средняя частота смерти рака в Челябинской области
ТАБЛИЦА VII. ОЦЕНКА РИСКА РАКА И ЛЕЙКЕМИИ У НАСЕЛЕНИЯ, ОБЛУЧЕННОГО В РЕЗУЛЬТАТЕ ОБРАЗОВАНИЯ ВОСТОЧНО-УРАЛЬСКОГО РАДИОАКТИВНОГО СЛЕДА

<table>
<thead>
<tr>
<th>Показатель</th>
<th>Эвакуированное население</th>
<th>Незвакуированное население</th>
<th>Всего</th>
</tr>
</thead>
<tbody>
<tr>
<td>Общая численность, тыс.чел.</td>
<td>10,7</td>
<td>262</td>
<td>270</td>
</tr>
<tr>
<td>Суммарная по всем группам коллективная эффективная эквивалентная доза за 30 лет, чел.-бэр</td>
<td>143 700</td>
<td>111 600</td>
<td>255 000</td>
</tr>
<tr>
<td>Максимальный риск на всю оставшуюся жизнь (на протяжении 70 лет существования следа), случай/(чел.-год):</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>рак для популяции</td>
<td>7,3</td>
<td>10,2</td>
<td></td>
</tr>
<tr>
<td>— для индивидуума</td>
<td>$6,8 \times 10^{-4}$</td>
<td>$3,8 \times 10^{-5}$</td>
<td></td>
</tr>
<tr>
<td>лейкемии для популяции</td>
<td>1,9</td>
<td>3,0</td>
<td></td>
</tr>
<tr>
<td>— для индивидуума</td>
<td>$1,8 \times 10^{-4}$</td>
<td>$1,1 \times 10^{-5}$</td>
<td></td>
</tr>
<tr>
<td>Сроки максимальной частоты проявления эффектов, лет после аварии:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>рак</td>
<td>22–24</td>
<td>24–30</td>
<td></td>
</tr>
<tr>
<td>лейкемия</td>
<td>18–20</td>
<td>18–20</td>
<td></td>
</tr>
<tr>
<td>Максимальный риск смерти от радиационноиндуцированных эффектов, случай/(чел.-год):</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>рак</td>
<td>$2,3 \times 10^{-5}$</td>
<td>$6,3 \times 10^{-7}$</td>
<td>$1,5 \times 10^{-6}$</td>
</tr>
<tr>
<td>лейкемия</td>
<td>$6,0 \times 10^{-6}$</td>
<td>$1,8 \times 10^{-7}$</td>
<td>$4,0 \times 10^{-7}$</td>
</tr>
</tbody>
</table>

$(1,4 - 1,7) \times 10^{-3}$ случаев/(чел.-год). Можно констатировать, что ожидаемый дополнительный риск рака, обусловленный наличием Восточно-Уральского радиоактивного следа, недостаточно велик, чтобы быть обнаруженным на фоне большой спонтанной частоты рака и недостаточной численности облучаемых групп населения.
7. СОСТОЯНИЕ ЗДОРОВЬЯ НАСЕЛЕНИЯ

Непосредственно сразу после образования Восточно-Уральского радиоактивного следа был наложен систематический контроль за состоянием здоровья населения, подвергшегося отселению и продолжающего проживать на загрязненной территории. В соответствии с уровнями дозового воздействия весь контингент населения был классифицирован 3 группами: эвакуированное население; проживающее вблизи границ следа с плотностью загрязнения 2 Ки/км$^2$ по $^{90}$Sr и проживающее вдали от этих границ. Исследовали такие показатели здоровья, как физическое состояние, состояние кроветворения, неврологический статус, развитие детей, состояние новорожденных, их физическое развитие, аллергизация, состояние желудочно-кишечного тракта, инфекционная заболеваемость, ранняя детская смертность. В первые три года после аварии обследование проводили ежегодно, в последующие поздние периоды 1 раз в 10 лет. Исследования продолжаются и в настоящее время с целью выявления злокачественных опухолей и других заболеваний и установления причин смерти лиц, короткое время проживавших на загрязненной территории и длительное время в контрольных районах.

В первые три года у эвакуированного населения и групп лиц, проживавших на территории с плотностью загрязнения по $^{90}$Sr свыше 2 Ки/км$^2$ (табл. VIII), не было зарегистрировано (дополнительно к спонтанному

<table>
<thead>
<tr>
<th>Синдром</th>
<th>Частота проявления, % к числу обследованных</th>
</tr>
</thead>
<tbody>
<tr>
<td>Лучевая болезнь, все формы</td>
<td>Нет</td>
</tr>
<tr>
<td>Угнетение костного мозга</td>
<td>Нет</td>
</tr>
<tr>
<td>Уменьшение количества лейкоцитов в крови</td>
<td>21</td>
</tr>
<tr>
<td>Уменьшение количества тромбоцитов</td>
<td>Единичные случаи</td>
</tr>
<tr>
<td>Функциональные неврологические</td>
<td>Единичные случаи</td>
</tr>
<tr>
<td>расстройства</td>
<td></td>
</tr>
<tr>
<td>Органические неврологические изменения</td>
<td>Нет</td>
</tr>
<tr>
<td>Аллергизация</td>
<td>Нет</td>
</tr>
</tbody>
</table>
НИКИПЕЛОВ и др.

уровню) таких специфических проявлений, как лучевая болезнь в любых ее формах, не было случаев угнетения костного мозга, не выявлены органические неврологические изменения, случаи аллергизации. Не обнаружены проявления напряжения в виде учащения вегетососудистых расстройств, инфарктов миокарда, гипертонии и др. Вместе с тем у 21% общего числа обследованных в разное время более 5 тыс. человек хотя бы однократно было зафиксировано снижение количества лейкоцитов в периферической крови, изредка встречали уменьшение числа тромбоцитов и функциональные неврологические расстройства.

Особое внимание было обращено на самый показательный и чувствительный критерий общего состояния здоровья населения, быстро реагирующий на радиационную обстановку — ранняя детская смертность, то есть смертность детей в возрасте до года. Исследования были проведены среди детей, проживающих на территории радиоактивного следа (около 2 Ки/км² по 90Sr), проживающих на территории с плотностью загрязнения почвы 0,1-1 Ки/км² по 90Sr (контроль № 1) и проживающих в той же области, но удаленных от границ следа (0,1 Ки/км², контроль № 2).

Как видно из табл. IX, даже на фоне очень высокой в те годы ранней детской смертности не удалось выявить отягощающего влияния повышения облучения на этот показатель. Некоторое возрастание ранней детской смертности во втором контрольном районе обусловлено повышенной частотой пневмонии и болезни новорожденных.

Известно, что особые опасения вызывает теоретическая предпосылка о появлении аномалий у потомства облученных родителей. Исследование этого показателя проведено в 1980-1987 гг., то есть когда полностью сформировалась и реализовалась доза не только у первого, но и второго поколения лиц, подвергшихся лучевому воздействию. Данные табл. X, полученные на большом материале, позволяют говорить об отсутствии влияния радиоактивного следа на появление и реализацию в виде смерти от врожденных пороков развития у лиц, облучавшихся в первом и втором поколениях.

Анализ заболеваемости причин и смертности от злокачественных новообразований, выполненный по десятилетиям, показал отсутствие связи заболеваемости, структуры и смертности среди облучившегося и необлучившегося контингента (табл. XI).

Ни каких различий в смертности в зависимости от места проживания населения не выявлено, хотя на территории радиоактивного следа со временем увеличивается частота смертности от злокачественных опухолей. Последнее обусловлено общими процессами ухудшения экологической обстановки, наблюдаемыми как во всем мире, так и в нашей стране. Эффективная эквивалентная доза 50-100 бэр была получена ограничен-
ТАБЛИЦА IX. СМЕРТНОСТЬ ДЕТЕЙ В ВОЗРАСТЕ ДО ГОДА НА 1000 РОДИВШИХСЯ

<table>
<thead>
<tr>
<th>Причина смерти</th>
<th>Территория радиоактивного следа, км²</th>
<th>Контроль № 1</th>
<th>Контроль № 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Все причины</td>
<td>27,7</td>
<td>31,4</td>
<td>38,6</td>
</tr>
<tr>
<td>Расстройство питания</td>
<td>15,2</td>
<td>12,2</td>
<td>5,1</td>
</tr>
<tr>
<td>Пневмония</td>
<td>1,7</td>
<td>3,1</td>
<td>16,1</td>
</tr>
<tr>
<td>Инфекционные болезни</td>
<td>1,6</td>
<td>2,3</td>
<td>3,0</td>
</tr>
<tr>
<td>Болезнь новорожденных</td>
<td>8,7</td>
<td>13,8</td>
<td>14,5</td>
</tr>
</tbody>
</table>

ТАБЛИЦА X. СМЕРТНОСТЬ НОВОРОЖДЕННЫХ С ВРОЖДЕННЫМИ ПОРОКАМИ РАЗВИТИЯ НА 1000 ЖИВОРОЖДЕННЫХ

<table>
<thead>
<tr>
<th>В целом по зоне влияния, включая территорию радиоактивного следа</th>
<th>Челябинская область</th>
<th>Свердловская область</th>
</tr>
</thead>
<tbody>
<tr>
<td>0,95 ± 0,08</td>
<td>1,0 ± 0,08</td>
<td>1,1 ± 0,07</td>
</tr>
</tbody>
</table>

ТАБЛИЦА XI. СМЕРТНОСТЬ ОТ ЗЛОКАЧЕСТВЕННЫХ ЗАБОЛЕВАНИЙ, ЧИСЛО СЛУЧАЕВ НА 100 ТЫС. ЧЕЛ. НАСЕЛЕНИЯ В ГОД

<table>
<thead>
<tr>
<th>Время исследований, год</th>
<th>В целом по зоне влияния, включая территорию радиоактивного следа</th>
<th>Челябинская область</th>
<th>Свердловская область</th>
</tr>
</thead>
<tbody>
<tr>
<td>1970 — 1980</td>
<td>145,8</td>
<td>146,6</td>
<td>—</td>
</tr>
<tr>
<td>1980 — 1987</td>
<td>160,7</td>
<td>167,6</td>
<td>159,4</td>
</tr>
</tbody>
</table>
ним числом людей (табл. IV), поэтому до настоящего времени отличий в структуре заболеваемости этой группы населения не выявлено.

Приведенные данные о состоянии здоровья населения в зоне влияния Восточно-Уральского радиоактивного следа в целом соответствуют вышеприведенным теоретическим оценкам радиационного риска.

8. ВОССТАНОВЛЕНИЕ ХОЗЯЙСТВЕННОЙ ДЕЯТЕЛЬНОСТИ НА ЗАГРЯЗНЕННОЙ ТЕРРИТОРИИ

Задача восстановления хозяйственной деятельности на территории Восточно-Уральского радиоактивного следа, в первую очередь задача восстановления сельскохозяйственного производства, возникла одновременно с введением ограничений на нее и считалась первоочередной наряду с осуществлением комплекса мер радиационной защиты человека.

Для обоснования практических мер по восстановлению сельскохозяйственного производства на загрязненной территории были проведены обширные исследования поведения $^{90}$Sr в сельскохозяйственной цепи почва—сельскохозяйственные растения—продукция растениеводства (корнеплоды и зерно)—сельскохозяйственные животные—продукция животноводства. Они позволили выявить звенья цепи, характеризующиеся наибольшими и наименьшими уровнями накопления $^{90}$Sr. Средние значения концентрации $^{90}$Sr в продукции в расчете на 1 Ки $^{90}$Sr/км$^2$ составляли, в нКи/кг: трава естественная — 3,5, силос — 0,40, корнеплоды и зерно — 0,10, картофель — 0,07, молоко — 0,09, мясо говяжье — 0,02. Также были определены условия, влияющие на интенсивность поступления $^{90}$Sr в каждое звено цепи.

Дополнительно к теоретическим исследованиям закономерностей поступления $^{90}$Sr в сельскохозяйственную продукцию были проведены многочисленные разработки практических приемов по снижению поступления $^{90}$Sr в конкретных условиях общепринятых технологий растениеводства, в том числе кормопроизводства, и животноводства, а также разработки специальных приемов, не типичных для общепринятой практики сельского хозяйства.

В зависимости от предназначения и способов их осуществления изученные и в большинстве своем внедренные приемы могут быть подразделены на агромелиоративные, агротехнические, зоотехнические и хозяйственные.

Одним из наиболее эффективных агромелиоративных приемов являлась первичная дезактивация почвы с захоронением верхнего загрязненного ее слоя в подпахотные горизонты (глубокая вспашка). Для ее осуществления на основе плантажных плугов типов ПП-50П и ППУ-50А были разработаны конструкции специальных почвообрабатывающих
орудий (плуг плантажный переоборудованный, переместитель почвенных горизонтов), которые характеризовались высокой эффективностью в снижении концентрации $^{90}$Sr в созданном заново слое корнеобитания растений. Плуг плантажный переоборудованный обеспечивает захоронение загрязненного слоя на глубину 30–40 см, снижая концентрацию $^{90}$Sr в пахотном слое на 80%, переместитель горизонтов — на глубину 30–70 см при снижении в образующемся пахотном слое до 10–50 раз. Иллюстрация распределения слоев почвы при такой обработке приведена на рис. 4. В экспериментах подобная мелиорация пахотных угодий снижала накопление $^{90}$Sr в зерне пшеницы до 75%, в клубнях картофеля — до 99%. В дополнение к первичной дезактивации пахотных угодий путем обычной отвальной вспашки, проведенной в 1958–1959 гг. на площади более 20 тыс. га и существенно улучшившей радиационную обстановку, в течение 1960–1961 гг. глубокой вспашке было подвергнуто 6400 га, в том числе на приусадебных участках отдельных населенных пунктов. Практическая эффективность глубокой вспашки, оцененная по снижению накопления $^{90}$Sr в урожае, составляла 2–5 раз.

На естественных угодьях, а также на посевах многолетних трав в качестве одноразовой первоначальной мелиорации возможно окультуривание угодий, включающее их перепашку или фрезерование с последующим заложением многолетними травами. Концентрация $^{90}$Sr в травах при проведении этого приема снижается вследствие минерализации дернины, содержащей более доступный $^{90}$Sr, и перевода его в менее доступное состояние; по завершении минерализации через 2–3 года уровень загрязнения трав уменьшается в 2–4 раза.

Зоотехнические приемы снижения накопления $^{90}$Sr в продукции животноводства были направлены на определение состава кормовых рационов с минимальным содержанием $^{90}$Sr, а также на изыскание способов содержания животных, которые, с учетом физиологических их особенностей, могли бы дополнительно снизить уровни загрязнения получаемой продукции.

Разработка оптимальных структур кормовых рационов сельскохозяйственных животных и птицы основывалась прежде всего на различиях в концентрации $^{90}$Sr в разных его компонентах, из которых наиболее загрязненными являются корма с естественных угодий. Поэтому содержание $^{90}$Sr в кормовых рационах может быть значительно сокращено за счет увеличения вклада картофеля, корнеплодов и зерна. Замена пастбищно-сенного рациона крупного рогатого скота на смешанный в условиях производства кормов на угодьях с одинаковой плотностью радиоактивного загрязнения позволяет получить молоко и мясо с концентрацией в 2,8 раза меньшей, а замена на силосно-концентратный рацион — в 5,6 раза. Вследствие этого было выработано принципиальное положе-
РИС. 4. Распределение слоев почвы и радиоактивности при различной обработке ее техническими орудиями, где:
1 — необработанная почва (вся активность сосредоточена в верхнем пятисантиметровом гумусовом слое (крестообразная штриховка)); 2 — почва, подвергнутая обычной вспашке (верхний пятисантиметровый слой равномерно распределен в пахотном горизонте на глубине 25–30 см); 3 — почва, подвергнутая глубокой вспашке (верхний пятисантиметровый слой захоронен в подпахотный горизонт на глубину более 50 см. Его место занял вывернутый слой почвы из подпахотных горизонтов (косая штриховка)).
ние о максимальном сокращении или полном исключении из состава рациона кормов с естественных угодий.

При разработке хозяйственных мер особое внимание было обращено на изыскание систем сельскохозяйственного производства, которые могли бы давать возможность получения валовой продукции с минимальными уровнями ее радиоактивного загрязнения.

Принципы возможностей восстановления сельскохозяйственного производства были следующими:

1. На наименее загрязненных площадях (2–5 Ки 90Sr/kм²) должно быть размещено производство продовольственных культур, фуражные культуры могут быть размещены на угодьях с плотностью загрязнения примерно в 10 раз большей.

2. Животноводство должно быть интенсивным с максимальным исключением из рациона сельскохозяйственных животных кормов с естественных угодий, а также грубых кормов. Следует обеспечить высокий вклад в рацион концентратов и корнеклубнеплодов.

3. Производство кормов для молочного скота должно быть организовано на площадях с уровнями загрязнения в 3–4 раза меньшими, чем для мясного скота.

4. Среди всех отраслей мясного животноводства наиболее предпочтительными являются свиноводство и разведение птицы как поставщиков наименее загрязненной мясной продукции.

5. Зерновая продукция и картофель с уровнями загрязнения свыше допустимых могут быть использованы на семенные цели и для глубокой технической переработки, например, для производства этилового спирта.

Правильная организация землепользования и создание специальных севооборотов в сочетании с разработанными агротехническими и зоотехническими приемами должны были стать основой восстанавливаемого сельскохозяйственного производства. При этом предлагаемая специализация хозяйств стала зависеть не только от средних уровней загрязнения и площади угодий, входящих в состав землепользования этих хозяйств, но и от необходимости радиационной защиты населения — персонала хозяйств, которое имело потенциальные возможности потреблять в пищу любую продукцию, производимую на вовлекаемых в хозяйственное использование угодьях. Это заставило отказаться от многоотраслевого производства, в первую очередь от получения продовольственных молока, зерна и овощей на вовлекаемых землях, и прийти к выводу о необходимости специализации сельскохозяйственного производства на производстве наиболее чистой продукции, к которой относится прежде всего мясо (допустимые плотности загрязнения территории при производстве мяса приведены в табл. XII).
ТАБЛИЦА П. ДОПУСТИМЫЕ ПЛОТНОСТИ ЗАГРЯЗНЕНИЯ СЕЛЬСКОХОЗЯЙСТВЕННЫХ УГОДИЙ $^{90}\text{Sr}$, УСТАНОВЛЕННЫЕ В КАЧЕСТВЕ НОРМАТИВОВ ДЛЯ ВОССТАНОВЛЕНИЯ СЕЛЬСКОХОЗЯЙСТВЕННОГО ПРОИЗВОДСТВА

<table>
<thead>
<tr>
<th>Отрасли животноводства</th>
<th>Допустимые плотности загрязнения угодий $^{90}\text{Sr}$, Ки/км$^2$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Без специальных систем</td>
</tr>
<tr>
<td></td>
<td>хозяйства с использованием естественных угодий</td>
</tr>
<tr>
<td>Мясное скотоводство</td>
<td>10</td>
</tr>
<tr>
<td>Молочное скотоводство</td>
<td>2,5</td>
</tr>
<tr>
<td>Свиноводство</td>
<td>100</td>
</tr>
</tbody>
</table>

Подобная специализация в большинстве случаев не соответствовала сложившейся экономической и хозяйственной структуре, поэтому такая специализация была рекомендована в качестве обязательной только в применении к использованию угодий, размещенных на территории следа в границах 2 Ки/км$^2$ по $^{90}\text{Sr}$, а вне этих границ допускалась многоотраслевость с преимущественным, наряду с производством мяса, производством фуражного зерна.

После установления границ Восточно-Уральского радиоактивного следа в 1958 г. оказались выведенными из хозяйственного использования и лишенными населения около 106 тыс. га земель, из которых сельхозугодья составляли примерно 55%, в том числе пашня до 30%. Задача вовлечения этих земель в хозяйственное использование осложнялась недостаточно большой долей (около 55 тыс. га) земель с потенциально допустимой плотностью радиоактивного загрязнения около 10 Ки/км$^2$ по $^{90}\text{Sr}$. Вследствие этого первые и все последующие мероприятия исходили из принципа дифференцированного использования земель в зависимости от уровней их радиоактивного загрязнения.

Решениями Челябинского и Свердловского облисполкомов по постановлению Совета Министров РСФСР в 1959–1960 гг. вместо мелких хозяйств были созданы специализированные совхозы: в Челябинской области — 4 (впоследствии 6), в Свердловской области — 3. Совхозы территориально размещены вне пределов Восточно-Уральского радиоактивного следа, определяемых плотностью загрязнения 2 Ки/км$^2$ по $^{90}\text{Sr}$, и
включают вовлекаемые в использование угодья как часть своего землепользования. Вовлечение выведенных земель в хозяйственное использование началось в 1961 г., когда были возвращены все земли в Свердловской области (с плотностью загрязнения не свыше 8 Ки/км² по 90Sr) и первые 2 тыс. га сельхозугодий в Челябинской области. Практически полная передача загрязненных земель специализированным совхозам на территории Челябинской области была завершена в 1968–1982 гг., когда из 32 тыс. сельхозугодий с максимальной плотностью загрязнения около 100 Ки/км² по 90Sr совхозам было передано 16 тыс. га. Остальная часть территории, размещенная в головной части следа и характеризующаяся наличием уровней загрязнения свыше 100 Ки/км² по 90Sr, не подлежит вовлечению в сельскохозяйственное использование вследствие потенциально высоких уровней загрязнения продукции и является базой для проведения натурных, в том числе сельскохозяйственных, экспериментальных исследований. Здесь создан Восточно-Уральский государственный заповедник общей площадью 16 700 га, на территории которого осуществляются разнообразные комплексные исследования по общим и прикладным вопросам радиоэкологии.

В границах следа 2 Ки/км² по 90Sr находится в сумме 16 тыс. га или 8,7–17,6% от площади общего землепользования каждого отдельного совхоза. Плотность загрязнения землепользования совхозов варьирует от 0,1–2 до 0,4–37 Ки/км² по 90Sr.

Каждое из специализированных хозяйств размещает севообороты культур достаточно оптимальным образом, что позволяет достигать необходимого "разбавления" уровней загрязнения продукции (кормов) с наиболее неблагополучных угодий за счет добавки продукции с менее загрязненных угодий.

Специализация совхозов на производстве мяса (5 совхозов с преимущественным производством говядины и 1 совхоз — свинины) обусловливает около 68% общей суммы стоимости реализованной продукции от реализации мяса, 20% — реализации молока и 12% — реализации продукции растениеводства.

Организация специальных хозяйств и внедрение рекомендаций позволили существенно снизить уровни содержания 90Sr в производимой продукции по сравнению с "неупорядоченным" сельскохозяйственным производством и личными хозяйствами. Производимая специализированными совхозами продукция содержит 90Sr меньше по сравнению с другими совхозами в 2,5–5 раз, а по сравнению с личными хозяйствами до 10 раз.
FORMATION OF DOSE BURDENS TO MAN AND THE ENVIRONMENT
Long term management of an area subjected to radioactive contamination
as a result of a radiation accident beyond the design basis

G.N. ROMANOV
USSR State Committee on the Utilization
of Atomic Energy,
Moscow,
Union of Soviet Socialist Republics

Abstract
FORMATION OF DOSE BURDENS TO MAN AND THE ENVIRONMENT: LONG TERM MANAGEMENT OF AN AREA SUBJECTED TO RADIOACTIVE CONTAMINATION AS A RESULT OF A RADIATION ACCIDENT BEYOND THE DESIGN BASIS.

The experience gained when cleaning up the accident in the Southern Urals in 1957 has served as a basis for the development of plans for long term safe habitation of areas contaminated by long lived nuclides, and the safe use of such areas for production purposes. Whether land is fit for long term habitation depends upon the permissible density of radioactive contamination in the area, and this can be evaluated in terms of the ratio of the emergency dose limit over the whole lifetime (70 years) to the potential combined exposure dose over the same period. Whether contaminated land can be used for production purposes is determined on the basis of an evaluation of the permissible radioactive contamination density, calculated as the ratio of the established permissible concentrations of a radionuclide in a specific type of produce to the real concentrations expressed in terms of the unit contamination density. When permissible concentrations rise, the type of production must be changed; for instance agricultural land must be turned over to other types of produce for which the permissible concentrations are higher. This means a change in the economic structure of the contaminated region.

1. INTRODUCTION

The conceptual proposals presented in this paper are based on the following problems:

(1) The need to ensure the radiation safety of the population living in the contaminated area for the first 70 years after the accident; and

(2) The need to make arrangements for the use of the contaminated land in such a way as to ensure the radiation safety of the population and to ensure that market produce complies with radiation and hygiene standards.
In order to solve these problems, the second of which should be organically linked to the first, it is necessary to divide the area up into zones according to the density of radioactive contamination, taking into account the possibilities for the long term residence of the population and the use of the contaminated land. It is also necessary to evaluate the prospects for using the land and ways of doing so in the various zones which have been subdivided according to the density of radioactive contamination.

2. DETERMINATION OF THE POSSIBILITIES FOR RESIDENCE OF THE POPULATION

The possibilities for the long term residence of the population in the contaminated area, for one generation after the accident (70 years), are determined by comparing the potential radiation burden to the population for that period of time with the limits established by the USSR Ministry of Health for exposure resulting from an accident. On the basis of the equivalence of the radioactive contamination density of the area and the consequent combined external and internal exposure doses of the population, the evaluation of the permissible radioactive contamination density (PCD) for the long term residence of the population can be derived from the following relationships:

\[ PCD = \frac{PD_a \cdot C_{eff} - D_1}{C_s \cdot NAD_1 \cdot \int_1^{70} \exp(-\lambda_{eff} t) dt} \]  

If \( D_1 = ND_1 \cdot PCD \), then

\[ PCD = \frac{PD_a \cdot C_{eff}}{ND_1 + \frac{C_s}{\lambda_{eff}} \cdot NAD_1 \cdot \exp[-\lambda_{eff}(70 - 1)]} \]  

where

PCD is the permissible radioactive contamination density for the long term residence of the population (taking a long lived radionuclide as a reference, for example, \(^{137}\text{Cs}\), Ci/km\(^2\);
PDₐ is the permissible accident dose for 70 years after the accident, in rem; the USSR Ministry of Health has established a whole body dose equivalent for $^{137}$Cs (the dose of combined external and internal exposure), equal to 35 rem for 70 years after the accident; if necessary, the PDₐ can be established from the effective dose equivalent, which means that the contributions of the long lived radionuclides $^{90}$Sr and plutonium to the contamination density and resulting exposure dose have to be calculated;

$D_1$ is the dose equivalent of the combined external and internal exposure to the population at the time of the proposed evaluation of the PCD;

$ND_1$ is the dose $D_1$ calculated per unit of radioactive contamination density (normalized $D_1$), rem·Ci⁻¹·m⁻²;

$NAD_1$ is the annual dose for combined external and internal exposure at the time of evaluating the PCD calculated per unit of radioactive contamination density (normalized annual dose) $(\text{rem/year})/\text{(Ci/km}^2\text{)}$;

$C_{eff}$ is the coefficient for the effectiveness of the set of measures, implemented or planned, for the radiation protection of the population, determined from the relationship between the doses with and without the application of the radiation protection measures. The radiation doses should be reduced as a result of the decontamination measures and the change in the way of life, and conditions of production, and also as a result of the reduction in the content of radionuclides in the population’s diet through the change in the proportion of locally produced products and the introduction of special measures in the agro-industrial sector, $C_{eff} \geq 1$;

$C_S$ is the ‘safety coefficient’ introduced to take account of the uncertainties and variable statistical distribution of the exposure levels of the population group examined; $C_S$ may have preference values within the limits 1.2–2;

$\lambda_{eff}$ is the constant for the rate of effective reduction in the normalized annual dose with approximation of this reduction by the exponential function, year⁻¹; the effective reduction is due not only to the radioactive decay of the nuclide, but also to the reduction in the external exposure dose rate as a result of migration and penetration of the radioactive substance into the soil, and also due to the natural reduction in the intake of radionuclides into the organism via food as a result of the migration of radionuclides in the environment and the reduction of their biological availability.

The evaluation of the PCD from Eq. (3) may be complicated by the fact that it is necessary to have correct evaluations of $C_{eff}$ and $\lambda_{eff}$ for the combined exposure.

---

1 rem = 0.01 Sv, 1 Ci = 3.70 × 10¹⁰ Bq.

2 Hereinafter in the text the word ‘dose’ is understood to mean dose equivalent.
In order to overcome these difficulties, we can introduce a simplification which involves determining the PCD separately for external and internal exposure. In doing this, we have to make the assumption, which may very well be conservative and may possibly lead to stricter evaluations of the PCD, that during the subsequent period the ratio of the external and internal exposure doses which has been established will be maintained (approximately 1:1).

Given this, the contamination density evaluated for the first criterion (external exposure) may be calculated from the equations

\[
PCD_1 = \frac{C_{\text{eff}} \cdot PD_a - D_{1,\text{ext}}}{C_S \cdot NDR_1 \cdot Y}
\]

or, if \(D_{1,\text{ext}} = ND_{1,\text{ext}} \cdot PCD_1\),

\[
PCD_1 = \frac{C_{\text{eff}} \cdot F_1 \cdot PD_a}{ND_{1,\text{ext}} + C_{\text{eff}} \cdot NDR_1 \cdot Y}
\]

where, in addition to the designations contained in Eqs (1-3),

- \(PCD_1\) is the permissible contamination density for the area, evaluated in terms of external exposure, \(\text{Ci/km}^2\);
- \(F_1\) is the contribution of external exposure to the combined internal and external exposure dose established at time 1;
- \(D_{1,\text{ext}}\) and \(ND_{1,\text{ext}}\) are respectively the external exposure dose, rem, and the external exposure dose, normalized in terms of the contamination density, rem per \(\text{Ci per km}^2\), which was received by the population at time 1 of the evaluation of the PCD;
- \(NDR_1\) is the external exposure dose rate (annual dose) calculated per unit of contamination density or the normalized dose rate at time 1, \((\text{rem/year})/\text{Ci/km}^2\);
- \(Y\) is the factor taking into account the change in \(NDR_1\) over the course of time resulting from the effect of the environment; primarily, the effect of penetration of the radioactive substance into the soil.

The permissible contamination density estimated for the second criterion (internal exposure resulting from the intake of radionuclides into the organism via (food) can be calculated from the equation

\[
PCD_2 = \frac{C_{\text{eff}} \cdot PD_a \cdot (1 - F_1) - D_{1,\text{int}}}{C_S \cdot NAD_{1,\text{int}} \cdot \int_1^{70} \exp(-\lambda_{\text{eff}}) dt}
\]

If \(D_{1,\text{int}} = ND_{1,\text{int}} \cdot PCD_2\)
then
\[
P_{CD2} = \frac{C_{eff}(1 - F_1) \cdot PD_a}{ND_{1,int} + \frac{C_S}{\lambda_{eff}} \cdot NAD_{1,int} \cdot \exp[-\lambda_{eff}(70 - 1)]}
\]  
where:

- $P_{CD2}$ is the permissible contamination density for the area evaluated in terms of internal exposure, Ci/km²;
- $D_{1,int}$, $ND_{1,int}$ are respectively the internal exposure dose, rem, and the internal exposure dose, normalized in terms of the contamination density in rem per Ci per km², which was received by the population at time 1 of the establishment of the PCD;
- $NAD_{1,int}$ is the annual internal exposure dose, normalized in terms of the contamination density at time 1 of the establishment of the PCD (rem/year)/(Ci/km²);
- $\lambda_{eff}$ is the constant of effective rate of reduction in the normalized annual internal exposure dose or the radionuclide content in the annual intake of locally produced food products, year⁻¹.

The other designations are as above.

In order to evaluate $P_{CD2}$, the first task is to establish numerical values for $\lambda_{eff}$. This can be done by careful analysis of the information which has been accumulated and acquired so far within the Soviet Union and abroad on the rate of reduction in the uptake of radionuclides over the course of time in the basic foodstuffs, produced in the region of radioactive contamination, as a result of the various radioecological processes, including reduction in the biological availability of radionuclides for plants through uptake from the soil by the roots. In the case of $^{137}Cs$, the most probable limits for a reduction by half in the time taken for uptake in plant products and fodder, and consequently in livestock products, may be 5–15 years, depending on the soil conditions and the agricultural production practices.

It should not be difficult to evaluate $NAD_{1,int}$ using conventional methods of evaluating internal exposure doses resulting from the intake of radionuclides via food, provided data are available regarding the structure of the diet of the population group examined, the proportions and annual intake of the various categories of locally produced products and the concentration or radionuclides in the products produced at time 1, calculated per unit of radioactive contamination density.

Similarly, in order to evaluate the PCD for the first criterion, it is also necessary to have an evaluation which is as accurate as possible for the coefficient $C_{eff}$, the numerical value of which should reflect the effect of the change in the level of contamination in the population's diet as a result of the reduction in the proportion of locally produced products, as well as the reduction in the concentration of radionuclides in them as a result of the special measures taken in the agricultural and processing sectors.
The evaluation of $PCD_1$ and $PCD_2$, based on the actual ratio of the external and internal exposure doses to the population at time 1, may nevertheless lead to differences in the numerical values of $PCD_1$ and $PCD_2$, which may be caused by the difference in the rate of reduction of the external and internal exposure dose during the period of time examined. In this case, there are three ways of deciding whether residence of the population in the region examined is possible:

1. Of the two values obtained for the PCD, the lower one is taken as a standard and the value is such that the population can live indefinitely in the area and no restrictions are placed on the production and consumption of locally produced products.

2. The value which is obtained from the criterion for external exposure and which exceeds the PCD for internal exposure is established as the PCD. This may mean that it is impossible to produce foodstuffs making up part or all of the diet. This situation may be the preferred one for populated centres such as towns.

3. The value which is obtained from the criterion for internal exposure and which exceeds the PCD for external exposure is established as the PCD. This should be regarded only as a way of organizing food production in this area without permanent residence of the population, for example, by the adoption of the duty tour method (migratory livestock farming) or by carrying out the activities which do not require frequent presence (laying-in of fodder).

3. USE OF THE CONTAMINATED AREA

As a necessary preface to the evaluation of possible uses of the contaminated area, it should be pointed out that the concepts for the long term use of this area should cover the whole area, regardless of whether it is populated or not.

The preliminary stage in evaluating possible uses of the area is the optimization of the use of the land in terms of agricultural, forestry and other types of activities. The result of this optimization may be represented as a chart showing the differentiations between the contaminated areas in terms of all the possibilities for organizing work which represents no danger to personnel and for obtaining products which comply with the radiation and hygiene requirements. The differentiation of the area should be based on the evaluation of the permissible radioactive contamination densities in terms of the criteria for the permissible doses of external exposure to personnel and the permissible levels of radioactive contamination of the products produced. In order to evaluate the permissible contamination densities, the following relationships can be used:

$$PCD_1 = \frac{PAD \cdot C_{eff \_1}}{C_s \cdot NAD}$$

(8)
where

PCD is the permissible radioactive contamination density in the area at the time of differentiation, Ci/km$^2$;

1 and 2 are the indices designating respectively the criteria for the permissible external exposure of personnel involved in the restoration of economic activity in the contaminated area and the permissible levels of radioactive contamination of the products produced;

PAD is the permissible annual dose of external exposure of personnel, rem/year; the numerical values for the PAD are established by the USSR Ministry of Health in the form of a quota of the maximum permissible dose (MPD) for personnel belonging to category A or a quota of the maximum dose (MD) for personnel belonging to category B; the quota of the MPD is established if the potential annual radiation burden exceeds the quota of the MD, which means that it is necessary to restrict the involvement of category B personnel in work in the contaminated area;

NAD is the annual external exposure dose, normalized in terms of the contamination density, (rem/year)/(Ci/km$^2$);

PLCP$_i$ is the permissible level of radioactive contamination established by the USSR Ministry of Health (for foodstuff $i$ and products other than foodstuffs) Ci/kg; particles·s$^{-1}$ cm$^2$;

NLCP$_i$ is the radioactive contamination of product $i$ normalized in terms of the contamination density, (Ci/kg)/(Ci/km$^2$); (particles·s$^{-1}$ cm$^2$)/(Ci/km$^2$);

C$_{eff}$ is the coefficient of effectiveness for the set of planned protection measures and measures for the reorganization of economic activities, C$_{eff}$ $\geq$ 1;

C$_S$ is the ‘safety coefficient’.

The proposed relationships will produce a series of values for the PCD which should be used as a basis for differentiation of the area in terms of the possibilities for the organization of safe work and the production of different types of foodstuffs and products other than foodstuffs. (See Table I.)

Differentiation of the land use on the basis of the criteria for the permissible levels of radioactive contamination of the products produced, including the evaluations for the permissible radioactive contamination density for the area, PCD$_{2i}$, should make it possible to solve the following problems:

1. Development of a structure for the distribution of agriculture, livestock farming, forestry and other types of economic activity using soil, plant and water resources, within the confines of the administrative region or an agro-industrial complex which would make it possible to produce foodstuffs and products other than foodstuffs with a level of radioactive contamination which does not exceed the established PLCP.
<table>
<thead>
<tr>
<th>Category of radioactive contamination density (zone)</th>
<th>Location of the category of contamination density in the scale of criteria for permissible external exposure</th>
<th>Conditions for the organization of work in the contaminated area</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>(&gt;\text{PCD}_1) (category A personnel) or (&gt;a \times 5\ \text{rem/year}), where (a) is the quota established from the MPD</td>
<td>Employment of category A personnel for urgent tasks</td>
</tr>
<tr>
<td>II</td>
<td>From (\text{PCD}_1) (category B personnel) to (\text{PCD}_1) (category A personnel) or (b \times 0.5 - a \times 5\ \text{rem/year}), where (b) is the quota established from the MD</td>
<td>Continuous employment of category A personnel Employment of category B personnel for urgent tasks</td>
</tr>
<tr>
<td>III</td>
<td>(&lt;\text{PCD}_1) (category B personnel) or (&lt;b \times 0.5\ \text{rem/year})</td>
<td>Continuous employment of category B personnel</td>
</tr>
</tbody>
</table>

(2) Development of a structure for the distribution of agriculture and livestock farming within the confines of one farm, provided that the radioactive contamination density gradients of the area of that farm are sufficient.

(3) Determination of the possibilities for the management of private farms and identification of the activities which have the highest level of radiation safety.

(4) Determination of ways of using products, including those other than foodstuffs, if it is not possible to comply with the PLCP levels.

(5) Determination of parts of the area (regions), where the immediate introduction of special measures is necessary to reduce the level of contamination of products.

The quantitative criteria which should be used to optimize the use of land, on the basis of the preliminary differentiation of the area according to the radioactive contamination levels, should take into account the permissible concentration density \(\text{PCD}_{2i}\), calculated using the temporary permissible levels established by the USSR Ministry of Health for the content of \(^{137}\text{Cs}\) in various products — TPL — which are, in essence, the same as the PLCP in Eq. (9).
TABLE II. DIVISION INTO ZONES OF THE LAND SUBJECT TO RADIOACTIVE CONTAMINATION AS A RESULT OF A RADIATION ACCIDENT BEYOND THE DESIGN BASIS IN TERMS OF THE POSSIBILITIES FOR ECONOMIC ACTIVITY
(without taking into account the effectiveness of the special measures introduced)

<table>
<thead>
<tr>
<th>Category of radioactive contamination density (zone)</th>
<th>Possible economic activity (in decreasing order of magnitude of the PPC\textsubscript{21})</th>
</tr>
</thead>
<tbody>
<tr>
<td>I (Zone of extremely high radioactive contamination)</td>
<td>Economic activity is not possible at present. Rehabilitation of certain sectors of economic activity (see Zone II) after implementation of special measures, for example, decontamination of the area</td>
</tr>
<tr>
<td>II (Zone of high radioactive contamination)</td>
<td>Production of products other than foodstuffs:</td>
</tr>
<tr>
<td></td>
<td>(1) Industrial production for non-domestic use</td>
</tr>
<tr>
<td></td>
<td>(2) Raw materials for local industry and the construction industry</td>
</tr>
<tr>
<td></td>
<td>(3) Fuel</td>
</tr>
<tr>
<td></td>
<td>(4) Products for domestic use (processing of forestry resources, clay, peat, etc.)</td>
</tr>
<tr>
<td>III (Zone of permissible radioactive contamination)</td>
<td>In addition to production recommended in Zone II, production of foodstuffs and products for domestic purposes:</td>
</tr>
<tr>
<td></td>
<td>(1) General industrial production</td>
</tr>
<tr>
<td></td>
<td>(2) Products for domestic purposes</td>
</tr>
<tr>
<td></td>
<td>(3-4) Agricultural and other production subjected to thorough industrial processing (production of alcohol, starch, sugar, vegetable oil, and other products for domestic purposes)</td>
</tr>
<tr>
<td></td>
<td>(3-4) Sowing of agricultural and forestry crops</td>
</tr>
<tr>
<td></td>
<td>(5) Fodder for animals not intended for human consumption (fur, farming, animal breeding)</td>
</tr>
<tr>
<td></td>
<td>(6) Crop products forming part of the population’s diet</td>
</tr>
<tr>
<td></td>
<td>(7) Fodder for farm animals destined for human consumption</td>
</tr>
<tr>
<td></td>
<td>(8) Farm animal products for human consumption</td>
</tr>
<tr>
<td>IV (Zone of controlled radioactive contamination)</td>
<td>No restrictions on economic activity</td>
</tr>
</tbody>
</table>

\textsuperscript{a} In this scheme, any earlier stage of production can be located in an area of higher radioactive contamination density than the following stages. Alteration can be made after the implementation of special protection measures in accordance with their level of effectiveness (C\textsubscript{eff}).
The numerical values for the radioactive contamination of products produced, normalized in terms of the contamination density (NLCP), should be evaluated before or during the differentiation of the land on the basis of experimental studies of the actual conditions. The specific radioecological conditions for each region studied must be taken into account. This involves primarily the change in the values for the NLCP caused by the soil properties, but also the role of the agricultural (or other type of) production technology, which already exists or which has been introduced, taking into account the effectiveness of the radiation protection measures introduced.

Optimization of land use should be conducted on different scales: within the boundaries of all the zones in the area contaminated as a result of the accident, within the administrative regions, the agro-industrial and other complexes, the individual farms, organizations and enterprises. The result of the optimization, as shown above, should be the division of the area into zones in accordance with the possibilities for the production of different categories of foodstuffs and products other than foodstuffs on the basis of the calculated PCD₂. The principles of such a division into zones can be seen in Table II. As a rule, the production of products for general industrial and non-domestic purposes should be located in the areas with the highest radiation contamination levels, whereas the production of foodstuffs should be located in the least contaminated areas, with the production of milk and meat in areas with minimum contamination.

Optimization of the land use should be repeated as special measures, such as decontamination, are implemented and after a sufficient period of time has elapsed to permit a significant reduction in the concentration of ¹³⁷Cs in the products produced, normalized in terms of the contamination density.

If it is not possible to comply with temporary permissible levels for specific types of products produced using conventional or specialized technology, the possibility of restructuring the economic activity in a given region should be considered. This may include the production of other products or new ways of using them, resulting in the production of products with different temporary permissible levels after industrial processing or in their use as intermediate products.

Restructuring and switching of production to the production of market products, including products other than foodstuffs, for which the PCD₂ makes it possible to keep within the actual radiation contamination density limits for the area, may be one of the most effective ways of rehabilitating economic activities and reducing the levels of internal exposure to the population. These levels may be determined approximately from the share of the production of each type of product in the area under consideration which is excluded from direct consumption by the population; there may be a potential reduction by a factor of several tens or hundreds in the intake of activity into the organism as a result of restructuring of agricultural production.
Restructuring of agricultural production in the case of one farm, and even more so in the case of a complex, region or province, entails considerable expense associated with the necessary restructuring of complicated production structures and technical infrastructures; another consequence of restructuring may be the need to compensate in other regions for the share of production of a given type of product which has been halted. The main criteria for the decision, based on economic and social considerations, regarding the introduction of production restructuring should be a comparison of the cost of restructuring and the profit which is expected from the restructured activities, taking into account the costs of the earlier production of products which have exceeded the temporary permissible levels in terms of the contamination levels and have become impracticable.

The basic structure for the use of the area after restructuring could be as follows.

If it is not possible to carry out agricultural production, particularly of foodstuffs, the economic activity in the region should be directed towards the production of products other than foodstuffs; in this connection, the possibility of developing the forestry industry should be considered (with simultaneous reforestation of land withdrawn from agricultural use and the long term development of production of virtually uncontaminated industrial timber as the conifer plantations mature after 50–70 years), local industry, exploitation of peat, sand, ballast and other mineral resources.

If it is possible to restructure agricultural production, changes may include:

— The production of industrial crops such as raw materials for goods other than foodstuffs (for example, the production of flax and hemp for the textile industry and the varnish and paint industry);
— Production of industrial crops such as raw materials for the production of foodstuffs subject to thorough industrial processing such as the production of potatoes for alcohol and starch, grain for alcohol, sugar beet for sugar and fodder yeast, sunflower, flax and hemp for oil, etc.;
— Seed breeding of cereals, potato, vegetable crops, seed grasses;
— Shift from dairy to beef cattle where it is possible to raise the head of beef cattle on ‘clean’ fodder prior to slaughter;
— Shift from dairy and beef cattle to pig farming and poultry farming, provided a given proportion of concentrated industrially produced fodder can be guaranteed from centralized stocks of uncontaminated or less contaminated fodder;
— Industrial flower growing (with flower breeding from seeds) and production of medical plant raw materials;
— Fur production, primarily with carnivorous animals.

The possibility should be considered of reducing the levels of radioactive contamination in products for human consumption by means of preliminary industrial preparation or processing.
Conventional technology for processing grain is characterized by a reduction in the concentration of $^{137}\text{Cs}$, in comparison with the initial grain, in wheat flour by a factor of 1.5–3.5, in rye flour by a factor of 1.5–2 and in cereals by a factor of 1.5–3. The purification factor in the case of potato starch is 50–20, for ethyl alcohol and sugar beet 1000, for sunflower oil 100 and for fodder yeast from beet 5–25. In the case of products resulting from the processing of cows' milk, a significant reduction in the initial concentration of $^{137}\text{Cs}$ occurs only in butter (a factor of 5–9) and rendered butter (a factor of 100–150).

The methods of restructuring production which have been indicated do not take into account the possible positive effect of the whole radiation protection plan or individual measures such as decontamination of the area by removal of the top layer of soil or burying it in the sub-ploughing level by deep ploughing, the introduction of special agrotechnical (agricultural melioration) and animal husbandry methods, as well as other special measures. It can be expected that, with the exception of decontamination, the effectiveness of these measures, evaluated in terms of the reduction in the level of radioactive contamination of products, will not generally exceed a factor of 2–3.

The general methodology for evaluating the possible use of the contaminated area should include the following four stages:

1. Differentiation of the land in terms of the permissible radioactive contamination levels $\text{PCD}_2$, as applied to the production of each type of foodstuff and products other than foodstuffs based on the existing and additional temporary permissible levels established, $\text{TPL}_4$;
2. As in (1) but taking into account the assumed effectiveness of the planned protection measures;
3. Determination of the territorial boundaries of possible production of individual products or all types of products not exceeding (in terms of the values for the $\text{PCD}_2$ established in accordance with (1) and (2)) the actual densities of radioactive contamination of the area and the setting aside of that part of the area where the values are higher than the $\text{PCD}_2$ for traditional products; selection of new types of economic activity; evaluation of the numerical values for $\text{PCD}_2$ for non-traditional products; comparison of these values for $\text{PCD}_2$ with the actual contamination density; the socioeconomic and technical-economic justification for the selected types of restructuring of farms or the region;
4. Selection of methods for organizing the work and identifying firms (organizations) capable of taking on responsibility for long term management, taking into account the different levels of radiation effect on personnel.

This sequence of activities may be carried out in conjunction with maps which can be compiled from the results of differentiation of the area in terms of the permis-
sible contamination density $PCD_2$ without restructuring and $PCD_2$ with restructuring, and also in terms of the permissible density $PCD_1$, taking into account the external exposure of personnel.

The $PCD_2$ has to be revised as the numerical values for the temporary permissible levels are corrected.
MEDICAL CONSEQUENCES OF THE RADIATION ACCIDENT IN THE SOUTHERN URALS IN 1957

Biophysics Institute of the USSR, Ministry of Health, Moscow
I.A. TERNOVSKIJ
USSR State Committee on the Utilization of Atomic Energy, Moscow
Union of Soviet Socialist Republics

Abstract

MEDICAL CONSEQUENCES OF THE RADIATION ACCIDENT IN THE SOUTHERN URALS IN 1957.

As a result of the radiation protection measures implemented after the accident causing radioactive contamination of the area, the maximum radiation dose received by 1150 people was 52 cSv; 9500 people received doses of 2.3–12 cSv. This portion of the population was evacuated. Another 270 000 people living in areas where the contamination density was less than 2 Ci/km² received doses of less than 1.4 cSv over 30 years. Monitoring of the health of evacuees and non-evacuees revealed no irregularities in the frequency with which somatic illnesses occurred, either during the earlier or later (30 years) monitoring period. No increase in the occurrence of malignant tumours was registered; no increase was found in infant mortality, stillbirths, congenital abnormalities or disturbances in the reproductive function, over the whole monitoring period; there was also no change in the birth rate. This was due to the low doses of radiation received.

On 29 September 1957, owing to a fault in the cooling system used for concrete tanks containing highly active nitrate-acetate wastes, a chemical explosion occurred in these materials and radioactive fission products were released into the atmosphere and subsequently scattered and deposited in parts of the Chelyabinsk, Sverdlovsk and Tyumensk provinces. The aggregate amount of activity released amounted to about $2 \times 10^6$ Ci ($7.4 \times 10^{16}$ Bq). The composition of the material released is indicated in Table I.
TABLE I. CHARACTERISTICS OF THE RADIONUCLIDE MIXTURE RELEASED IN THE ACCIDENT

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Contribution to total activity of the mixture (%)</th>
<th>Half-life</th>
<th>Type of radiation emitted</th>
<th>Nature of radiological hazard</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{89}$Sr</td>
<td>traces</td>
<td>51 d</td>
<td>$\beta, \gamma$</td>
<td>Internal irradiation (skeleton)</td>
</tr>
<tr>
<td>$^{90}$Sr + $^{90}$Y</td>
<td>5.4</td>
<td>28.6 a</td>
<td>$\beta$</td>
<td></td>
</tr>
<tr>
<td>$^{95}$Zr + $^{95}$Nb</td>
<td>24.9</td>
<td>65 d</td>
<td>$\beta, \gamma$</td>
<td>External irradiation</td>
</tr>
<tr>
<td>$^{106}$Ru + $^{106}$Rh</td>
<td>3.7</td>
<td>1 a</td>
<td>$\beta, \gamma$</td>
<td>External</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>0.036</td>
<td>30 a</td>
<td>$\beta, \gamma$</td>
<td>External and internal</td>
</tr>
<tr>
<td>$^{144}$Ce + $^{144}$Pr</td>
<td>66</td>
<td>284 d</td>
<td>$\beta, \gamma$</td>
<td>External</td>
</tr>
<tr>
<td>$^{147}$Pm</td>
<td>traces</td>
<td>2.6 a</td>
<td>$\beta, \gamma$</td>
<td></td>
</tr>
<tr>
<td>$^{155}$Eu</td>
<td>traces</td>
<td>5 a</td>
<td>$\beta, \gamma$</td>
<td></td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>traces</td>
<td>-</td>
<td>$\alpha$</td>
<td></td>
</tr>
</tbody>
</table>

For the area with a $^{90}$Sr contamination density of 0.1 Ci/km$^2$ (double the level of global fallout)$^1$, the maximum length of the deposition track under the radioactive plume formed reached 300 km; for $^{90}$Sr contamination density of 2 Ci/km$^2$ it reached 105 km, with a width of 8–9 km. The area density distribution is shown in Table II.

The presence of gamma emitters among the contaminating nuclides was responsible for the external irradiation of the population and the environment. During the initial period the dose rate was about 150 $\mu$R/h in the area$^2$ with a $^{90}$Sr contamination density of 1 Ci/km$^2$, with maximum values of 0.6 R/h at the head end of the track, where the contamination density ($^{90}$Sr) reached 4000 Ci/km$^2$.

Owing to radioactive decay of the short lived nuclides, contamination levels and gamma dose rates in the area of the accident fell off fairly rapidly during the first few years after formation of the deposition track (see Table III), and subsequently the radiation situation was governed entirely by the presence of $^{90}$Sr and its

$^1$ 1 Ci = $3.7 \times 10^{10}$ Bq.
$^2$ 1 R = $2.58 \times 10^{-4}$ C/kg.
TABLE II. AREA AND POPULATION OF THE CONTAMINATED REGION

<table>
<thead>
<tr>
<th>Density of radioactive contamination, $^{90}\text{Sr}$ (Ci/km$^2$)</th>
<th>Area (km$^2$)</th>
<th>Population ($\times 10^3$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.1</td>
<td>15 000</td>
<td>270</td>
</tr>
<tr>
<td>including:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>1 000</td>
<td>10</td>
</tr>
<tr>
<td>100</td>
<td>120</td>
<td>1.5</td>
</tr>
<tr>
<td>1000</td>
<td>20</td>
<td>1.154</td>
</tr>
</tbody>
</table>

rate of radioactive decay. The exposure of the population in the contaminated territory was due in the first instance to external irradiation from the soil and from objects in their dwellings — including their own clothing — and also to internal irradiation due to the consumption of contaminated food and drinking water and inhalation of activity at the time when the cloud was being formed. Subsequently (after six months to a year) internal exposure from contaminated food was predominant.

The radiation protection measures adopted for the population were as follows:

— Evacuation of the population;
— Decontamination of some portions of the agricultural land;
— Monitoring of contamination levels in agricultural produce and rejection of produce with activity levels exceeding the accepted norms;
— Limitations imposed on the utilization of contaminated land;
— Reorganization of agriculture and forestry, with the creation of specialized state farms and forestry enterprises operating in accordance with the special recommendations worked out in the light of the accident.

The dynamics of the evacuation exercise for persons living in regions with a $^{90}\text{Sr}$ contamination density above 2 Ci/km$^2$ are shown in Table IV.

In the immediate aftermath of the accident — that is, within 7 to 10 days — six hundred persons were evacuated from the settlements in the most severely affected area; and about ten thousand persons were evacuated in the 18 months following the accident. Altogether 10 180 persons were evacuated. Maximum average exposure doses preceding evacuation reached 17 rem$^3$ in external exposure and 52 rem in effective dose equivalent (150 rem to the gastro-intestinal tract). These doses can be doubled in view of the non-uniformity of contamination density and the conditions in which the exposure occurred.

$^3$ 1 rem = 0.01 Sv.
Mass medical surveillance of the inhabitants of the affected region was arranged a year after the accident. It included examinations by paediatricians and therapists, by neuropathologists and gynaecologists, peripheral blood analysis, and determinations of body weight and height. Factors entailing a risk of cancer were assessed, as was cardiovascular pathology, and the presence of harmful habits; and urine tests for albumin and sugar were carried out. Serum cholesterol levels were determined and oto-laryngeal examinations were organized; all the individuals studied were given ECGs.
TABLE V. FREQUENCY OF ABNORMALITIES OF ARTERIAL BLOOD PRESSURE AND SYSTOLE FREQUENCY (PULSE RATE) AMONG EXPOSED INDIVIDUALS BELONGING TO POPULATION GROUP A

<table>
<thead>
<tr>
<th>Index</th>
<th>Fraction of group affected (%)</th>
<th>(range)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tachycardia (pulse &gt; 90)</td>
<td>5.5</td>
<td>(4-7.4)</td>
</tr>
<tr>
<td>Brachycardia (pulse &lt; 60)</td>
<td>8.5</td>
<td>(0-14.1)</td>
</tr>
<tr>
<td>Hypertension (Blood pressure &gt; 160/95 mm Hg)</td>
<td>3.3</td>
<td>(1.7-4.0)</td>
</tr>
<tr>
<td>Borderline hypertension (blood pressure 140/90-159/94 mm Hg)</td>
<td>10</td>
<td>(7.5-14.5)</td>
</tr>
<tr>
<td>Hypotension (blood pressure &lt; 110/60 mm Hg)</td>
<td>16.4</td>
<td>(10.8-24.0)</td>
</tr>
</tbody>
</table>

The portion of the population most seriously affected by radiation (Group A) was comparatively young: persons 0-17 years of age accounted for 45%, 18-45 years of age 39% and above 50 only 16% of this group.

In the clinical studies carried out on this population no cases of radiation sickness were noted. During the early period of the investigations, 21% of the cases examined showed no decrease in the peripheral blood leucocyte count. However, the peripheral blood indices showed, in adults, average values for thrombocytes (236-280 × 10⁹/L), leucocytes (7.2-7.5 × 10⁹/L) and neutrophils (4.1-4.7 × 10⁹/L) which were no different from those found in normal unexposed adults. The distribution function for these indices during the first examination period was the same, in terms of median values, for the irradiated group as for the references, but among the exposed persons a larger percentage showed two-sigma deviations from the average. Thus, the fraction of individuals with leucocyte counts above 9 × 10⁹/L was 17-19%; with thrombocyte counts above 350 × 10⁹/L it was 7-8%.

The reaction of the cardiovascular system was studied in all the individuals investigated on the basis of arterial blood pressure and systole frequency (pulse rate). The results among the essentially healthy members of Group A (Table V) indicated no regular increase in the frequency with which these indices departed from a normal distribution, nor any deviation as a function of radiation dose.
### TABLE VI. NATURE AND FREQUENCY OF DISEASE IN THE INDIVIDUALS STUDIED

<table>
<thead>
<tr>
<th>Class of illness, nosological units</th>
<th>Fraction of individuals affected (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parasitic infection, helminthiasis</td>
<td>0.6</td>
</tr>
<tr>
<td>Nodular goitre and thyrotoxicosis</td>
<td>0.5</td>
</tr>
<tr>
<td>Psychic disorders, neurasthenia</td>
<td>1.9</td>
</tr>
<tr>
<td>Circulatory disorders:</td>
<td></td>
</tr>
<tr>
<td>Rheumatic heart disease</td>
<td>1.8</td>
</tr>
<tr>
<td>Hypertension</td>
<td>2.5</td>
</tr>
<tr>
<td>Ischaemia</td>
<td>3.3</td>
</tr>
<tr>
<td>Coronary and cerebral atherosclerosis</td>
<td>5.1</td>
</tr>
<tr>
<td>Varicose veins</td>
<td>1.0</td>
</tr>
<tr>
<td>Respiratory disorders:</td>
<td></td>
</tr>
<tr>
<td>Acute nasopharyngitis</td>
<td>5.3</td>
</tr>
<tr>
<td>Bronchitis</td>
<td>2.3</td>
</tr>
<tr>
<td>Emphysema</td>
<td>0.7</td>
</tr>
</tbody>
</table>

The population studies indicated that up to 75% were for all practical purposes healthy individuals. Twenty-five per cent of those investigated revealed general somatic problems of one kind or another. Among these, as can be seen from Table VI, more than half were suffering from cardiovascular disorders, and almost 30% had respiratory illnesses.

Thus the examinations conducted on individuals living in settlements at the head end of the deposition track showed no clinical evidence of radiation pathology. It is fair to assume that certain distortions in the distribution of blood indices are associated with the haematological reaction to irradiation observed in the early period: leucopenia, relative lymphopenia and a leftward shift in the neutrophil formula.

Long after the accident medical studies were performed on individuals who could be said to belong to the critical group: these were people whose exposure to radiation had occurred during the period of body formation and development and in whom the exposure levels had been highest (Groups A and B). Of these individuals, one third were, for all practical purposes, healthy. In the rest careful investigation
showed chronic infection of one kind or another (in 18% chronic otitis, in 13% chronic tonsillitis, and in another 16% chronic gastritis and cervicitis). The frequency of osteochondrosis increased with the age of the individual. Three persons had epilepsy associated with alcoholism and cranial trauma. The morbidity of the exposed individuals revealed no special characteristics in comparison with the control contingent. Peripheral blood indices were in the normal range. With increasing age the frequency of dystrophic changes in the ECG increased (classes 4, 5 and 9 in the Minnesota code). The frequency of ECGs in the zero class (with no changes) was no less in the irradiated persons than in the controls.

Serum cholesterol concentrations (mm/L) were the same among the irradiated contingent as among the controls: under 29 years of age 4.78 ± 0.1; up to 39 years of age 5.25 ± 0.06; up to 49 years 5.41 ± 0.06; and above 50 years 5.69 ± 0.06.

Certain disorders thought to be cancer risk factors were recorded no more frequently among irradiated persons than among the controls. Thus, in the 28-year-olds, chronic gastritis, endocervicitis and cervical erosion were encountered in 2.3, 11.1 and 11.1% of cases, respectively, whereas in the 50-year-olds they were found in 9, 20 and 0% of cases. One of the sensitive criteria for damage caused by ionizing radiation is infant mortality and intra-uterine developmental anomalies. Over the 35 year period, 35 cases of death due to congenital anomalies have been found among the offspring of the population living on land covered by the radioactive deposition track. In the first group, consisting of 10 270 individuals living in areas with a \(^{90}\text{Sr}\) concentration of 1–2 Ci/km\(^2\), there were 10 cases, and in the second group, consisting of 23 230 individuals living in an area with a \(^{90}\text{Sr}\) density of 0.1–1 Ci/km\(^2\) there were 25 cases. In the control group, consisting of 21 537 individuals living in an area with less than 0.1 Ci/km\(^2\) (\(^{90}\text{Sr}\)), there were 39 cases of death due to congenital anomalies. In the overall mortality structure we found that mortality due to developmental defects accounts for 0.36–0.67% of all cases (see Table VII).

As can be seen from Table VII, the differences between the groups are statistically unreliable; nor did any significant differences emerge during the first two years following the accident.

The figures in Table VIII demonstrate the absence of any statistically significant difference in infant mortality although they are higher for the settlement closest to the radiation source. The reason for these higher figures still needs further clarification.

Highly instructive data were obtained from analyses of infant mortality during the years after the accident (Table IX). As can be seen from this table, there is no appreciable difference in infant mortality between the three groups compared, even against the background of the rather high infant mortality prevailing during those years. Here, too, the cases of infant mortality are apparently not associated so much with the levels of radiation exposure as with inequalities in the medical treatment accorded to newly born infants.
TABLE VII. EXTENSIVE (%) AND INTENSIVE INDICES OF MORTALITY DUE TO CONGENITAL DEVELOPMENTAL ANOMALIES

<table>
<thead>
<tr>
<th>Population group</th>
<th>Extensive indices</th>
<th>Intensive indices after 35 years</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 - 10 270 individuals</td>
<td>0.36</td>
<td>4.2</td>
</tr>
<tr>
<td>2 - 23 230 individuals</td>
<td>0.38</td>
<td>4.2</td>
</tr>
<tr>
<td>K - 21 537 individuals</td>
<td>0.67</td>
<td>7.4</td>
</tr>
<tr>
<td>Chelyabinsk province</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- 1965</td>
<td>0.53</td>
<td>3.6</td>
</tr>
<tr>
<td>- 1986</td>
<td>0.23</td>
<td>2.2</td>
</tr>
</tbody>
</table>

TABLE VIII. MORTALITY OF NEWBORN INFANTS WITH CONGENITAL DEVELOPMENTAL DEFECTS BETWEEN 1980 AND 1987 (PER 1000 LIVE BIRTHS)

<table>
<thead>
<tr>
<th>Location</th>
<th>Rate of Mortality (per 1000 live births)</th>
</tr>
</thead>
<tbody>
<tr>
<td>In the entire affected zone</td>
<td>0.95 ± 0.08</td>
</tr>
<tr>
<td>At the nearest settlement</td>
<td>1.7 ± 0.4</td>
</tr>
<tr>
<td>In Chelyabinsk province</td>
<td>1.0 ± 0.08</td>
</tr>
<tr>
<td>In Sverdlovsk province</td>
<td>1.1 ± 0.07</td>
</tr>
</tbody>
</table>

The remote (long term) effects of radiation exposure were studied in parallel among the irradiated population and the control contingent, and also in a zone where the effects of a nuclear facility might be expected to make themselves felt. In this way, more than 100,000 persons were surveyed.

Table X shows the effects of exposure to radiation on the most severely irradiated contingent. Among these persons, intensive mortality indices in Groups 1, 2, 3 and K were 272, 2760, 6578 and 5873 cases, respectively, and the corresponding mortality coefficients were 9.5, 11.5, 11.0 and 10.9 × 10⁻³. It will be seen that there are no differences as compared with the control contingent.

At the same time, age-related mortality indices show substantial deviations from the control contingent in individuals under age 4 and older than age 60. Nevertheless, it has proved impossible to find any link with the radiation dose. Thus, in Groups 1, 2, 3 and K the mortality coefficients for children aged up to one year
TABLE IX. MORTALITY OF INFANTS AGED UP TO 1 YEAR (PER 1000 LIVE BIRTHS)

<table>
<thead>
<tr>
<th>Causes of death</th>
<th>Territory covered by deposition track</th>
<th>Control No. 1 on track boundary</th>
<th>Control No. 2 far from track boundary</th>
</tr>
</thead>
<tbody>
<tr>
<td>All causes</td>
<td>27.7</td>
<td>31.4</td>
<td>38.6</td>
</tr>
<tr>
<td>Nutritional disorders</td>
<td>15.2 ± 2.8</td>
<td>12.3 ± 3</td>
<td>5 ± 1</td>
</tr>
<tr>
<td>Pneumonia</td>
<td>1.7 ± 1.0</td>
<td>3.1 ± 1.5</td>
<td>16.1 ± 1.8</td>
</tr>
<tr>
<td>Infectious diseases</td>
<td>1.6 ± 0.9</td>
<td>2.3 ± 1.3</td>
<td>3.0 ± 0.8</td>
</tr>
<tr>
<td>Disease of the newborn</td>
<td>8.7 ± 2.2</td>
<td>13.8 ± 3.2</td>
<td>14.5 ± 1.7</td>
</tr>
</tbody>
</table>

TABLE X. SIZE OF THE EXPOSED POPULATION AND AVERAGE DOSES RECEIVED

<table>
<thead>
<tr>
<th>Group</th>
<th>Number of inhabitants</th>
<th>Duration exposure</th>
<th>External gamma irradiation</th>
<th>Internal irradiation</th>
<th>Effective dose eq.</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>G.I. tract</td>
<td>Lungs</td>
<td>Red bone marrow</td>
</tr>
<tr>
<td>1</td>
<td>1 054</td>
<td>10 d</td>
<td>17</td>
<td>150</td>
<td>2.7</td>
</tr>
<tr>
<td>2</td>
<td>10 720</td>
<td>30 a</td>
<td>0.4</td>
<td>2</td>
<td>0.2</td>
</tr>
<tr>
<td>3</td>
<td>23 230</td>
<td>30 a</td>
<td>0.1</td>
<td>0.7</td>
<td>0.1</td>
</tr>
<tr>
<td>K</td>
<td>21 537</td>
<td>30 a</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

were 91, 32, 63 and 52, for children between one and four years of age 13.7, 1.7, 5.0 and 3.3, and for individuals over 60 years of age, 39.2, 50.4, 43.1 and 46.9, respectively. In all other age groups the mortality indices and coefficients reflected no differences between the groups and the control contingent.

A fact to be noted is that among the 272 individuals from Group 1 who died, cancer is not in second place but in third place as a cause of death after heart disease, injuries and accidents. Another peculiarity is the predominance of death from infectious diseases over that from respiratory disorders.
TABLE XI. EXTENSIVE (%) AND INTENSIVE ($\times 10^{-5}$) INDICES OF MORTALITY DUE TO MALIGNANT TUMOURS

<table>
<thead>
<tr>
<th>Population group</th>
<th>Number of cases</th>
<th>%</th>
<th>$10^{-5}$</th>
<th>Confidence intervals 95%</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>25</td>
<td>11.7</td>
<td>115.9</td>
<td>75-165</td>
</tr>
<tr>
<td>2</td>
<td>376</td>
<td>13.6</td>
<td>157.4</td>
<td>142-174</td>
</tr>
<tr>
<td>3</td>
<td>775</td>
<td>11.8</td>
<td>129.2</td>
<td>120-142</td>
</tr>
<tr>
<td>K</td>
<td>707</td>
<td>12.0</td>
<td>131.9</td>
<td>122-142</td>
</tr>
</tbody>
</table>

TABLE XII. STRUCTURE OF MORTALITY DUE TO MALIGNANT NEOPLASMS (PER 100 000 POPULATION)

<table>
<thead>
<tr>
<th>Principal tumour sites</th>
<th>International classification number</th>
<th>Irradiated individuals</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>1</td>
</tr>
<tr>
<td>Oesophagus</td>
<td>150</td>
<td>26.5</td>
</tr>
<tr>
<td>Stomach</td>
<td>151</td>
<td>35.3</td>
</tr>
<tr>
<td>Other organs of the digestive system</td>
<td>152-159</td>
<td>8.8</td>
</tr>
<tr>
<td>Respiratory organs</td>
<td>160-163</td>
<td>17.7</td>
</tr>
<tr>
<td>Bones</td>
<td>170</td>
<td>0</td>
</tr>
<tr>
<td>Skin, oral cavity</td>
<td>140-147</td>
<td>0</td>
</tr>
<tr>
<td>Mammary gland</td>
<td>172-173</td>
<td>0</td>
</tr>
<tr>
<td>Corpus et cervix uteri</td>
<td>174</td>
<td>4.4</td>
</tr>
<tr>
<td>Other urogenital organs</td>
<td>180-182</td>
<td>0</td>
</tr>
<tr>
<td>Lymphatic and haematopoietic tissue</td>
<td>183-189</td>
<td>4.6</td>
</tr>
<tr>
<td></td>
<td>200-209</td>
<td>13.2</td>
</tr>
</tbody>
</table>
Particularly interesting is the analysis of mortality due to malignant tumours since these are the principal later manifestation of the effects of irradiation. The highest mortality indices — on the boundary of reliable statistical significance — were noted in individuals belonging to Group 2 (see Table XI). However, the groups selected for the purposes of this comparison were not large enough and hence it is impossible to conclude definitely that there is any appreciable difference between the magnitudes observed.

In the overall picture of neoplastic disorders observed, the most important place is occupied — over the entire period of interest — by cancer of the digestive tract, and in particular by cancer of the oesophagus (see Table XII). We see a tendency towards increased cancer of the oesophagus in Group 1, which was the group subjected to the highest doses, but even so these higher figures are not statistically significant.

Attention is also drawn to fatal cases of lymphatic neoplasms and neoplasms in haematopoietic tissue. The mortality coefficient in Group 1 was $13.2 \times 10^{-5}$ as compared with $4.7 \times 10^{-5}$ in the other groups. Although these differences are not statistically reliable because they are based on only three fatalities, we should note that the effective dose equivalent in this group amounted to 52 cSv, which is close to the critical dose for leukaemia induction.

The level of mortality due to neoplastic disorders, broken down by decades, is shown in Table XIII for the zone directly affected by the deposition track and compared with intensive indices for the neighbouring regions Chelyabinsk (1) and Sverdlovsk (2). The figures show that lethality regularly increases from the first decade to the next. Overall, in the zone affected by the track and by the operations of nuclear facilities, it rose from 145.8 to 160.7 ± 25 per 100 000 of population, and in regions one and two, from 167.6 ± 3.2 and 159.4 ± 6.6 per 100 000. In the nearest settlement the figure was only 105 ± 12.7, although the radiation dose here was higher. This lower figure is due entirely to the younger average age of the population in this settlement.

Analysis of the cause of morbidity due to malignant neoplasms following the irradiation accident suggests a frequency grouping for the initially diagnosed
TABLE XIV. PERCENTAGE OF INDIVIDUALS WHO MARRIED AND HAD CHILDREN

<table>
<thead>
<tr>
<th>Group</th>
<th>Age at time of accident</th>
<th>Number of individuals</th>
<th>% married</th>
<th>% with children</th>
</tr>
</thead>
<tbody>
<tr>
<td>Infants</td>
<td>Under 1 year</td>
<td>56</td>
<td>91 (82-97)</td>
<td>84 (73-92)*</td>
</tr>
<tr>
<td>Children</td>
<td>1-9 years</td>
<td>295</td>
<td>93 (89-96)*</td>
<td>90 (86-93)*</td>
</tr>
<tr>
<td>Juveniles</td>
<td>10-19 years</td>
<td>203</td>
<td>93 (89-96)*</td>
<td>93 (89-96)</td>
</tr>
<tr>
<td>Adults</td>
<td>20-29 years</td>
<td>201</td>
<td>95 (92-98)*</td>
<td>91 (87-94)</td>
</tr>
<tr>
<td>Adults</td>
<td>30-59 years</td>
<td>308</td>
<td>98 (96-99)*</td>
<td>98 (96-99)*</td>
</tr>
<tr>
<td>Controls</td>
<td></td>
<td>81.9-82.6</td>
<td>94.6</td>
<td></td>
</tr>
</tbody>
</table>

(USSR as a whole)

* Significant differences from control.

tumours as a function of certain external factors. It was noted, for example, from the morbidity data in Chelyabinsk, that there was no connection between enhanced morbidity and dose rate. On the other hand, a clear and complete correlation was found between morbidity and releases of SO$_2$ to the atmosphere. Although SO$_2$ is not itself a carcinogen, it is extremely useful as a gauge of general chemical contamination. Actual data show that when there are no SO$_2$ releases morbidity amounts to 225 cases per 100 000 individuals per year, whereas in situations where SO$_2$ is released in amounts of 50 000, 100 000 and 150 000 t per year the morbidity figures rise to 250, 275 and 300 cases per 100 000, respectively. Accordingly, on the Chelyabinsk map the cancer mortality figures correlate not with the radioactive contamination track but with the location of metallurgical and chemical plants.

A great deal of attention has been given to the reproductive state of individuals irradiated at different ages. The figures in Table XIV indicate no systematic deviations from the norm — as far as this extremely important demographic indicator is concerned — among the individuals who received the largest doses. It will be seen from Table XIV that those who were newborn infants at the time of the accident and who married by the time they were 27 years of age have, as yet, comparatively few children. Among those who were older at the time, on the other hand, marriage frequency has been higher than among the controls, and the number of children has been either no different or slightly less than among the controls (this applies to individuals who were up to nine years old at the time of the accident). At the same time, the birth rate coefficients per 1000 inhabitants, as can be seen from Table XV, are higher among the population in the affected region than in the district as a whole.
TABLE XV. DYNAMICS OF BIRTH RATE COEFFICIENTS AMONG THE EVACUATED POPULATION (PER 1000)

<table>
<thead>
<tr>
<th>Time after accident (a)</th>
<th>1</th>
<th>5</th>
<th>10</th>
<th>15</th>
<th>20</th>
<th>25</th>
<th>30</th>
<th>1–30</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of children</td>
<td>51</td>
<td>271</td>
<td>491</td>
<td>717</td>
<td>960</td>
<td>1242</td>
<td>1586</td>
<td>1616</td>
</tr>
<tr>
<td>Birth rate coefficient</td>
<td>37.4</td>
<td>42.2</td>
<td>30.2</td>
<td>27.5</td>
<td>26.4</td>
<td>27.8</td>
<td>30.0</td>
<td>31.8</td>
</tr>
<tr>
<td>Standardized coefficients</td>
<td>40.4</td>
<td>48.7</td>
<td>31.8</td>
<td>26.9</td>
<td>24.8</td>
<td>26.2</td>
<td>26.6</td>
<td>31.8</td>
</tr>
<tr>
<td>Birth rate coefficients for Chelyabinsk province</td>
<td>24.1</td>
<td>20.8</td>
<td>14.8</td>
<td>16.0</td>
<td>16.7</td>
<td>19.8</td>
<td>16.7</td>
<td>18.4</td>
</tr>
</tbody>
</table>

One gets the impression that living conditions and social factors among the population evacuated at the time of the accident are somewhat more favourable than among the rest of the agricultural population in the region. There may also be some other factors, such as special national characteristics, involved.

In conclusion, we may note that observations on health, morbidity and mortality among the population subjected to the accidental release of radiation — with whole body exposure doses from 1–52 cSv and irradiation of individual organs up to 150 cSv — have revealed no significant variations from the comparable values found among healthy unexposed individuals.
RADIOACTIVE CLOUD TRACE FORMATION DYNAMICS AFTER THE RADIATION ACCIDENT IN THE SOUTHERN URALS IN 1957

Migration processes

I.A. TERNOVSKIJ, G.N. ROMANOV, E.A. FEDOROV,
E.N. TEVEROVSKIJ, Yu.B. KHOLINA
USSR State Committee on the Utilization
of Atomic Energy,
Moscow,
Union of Soviet Socialist Republics

Abstract

RADIOACTIVE CLOUD TRACE FORMATION DYNAMICS AFTER THE RADIATION ACCIDENT IN THE SOUTHERN URALS IN 1957: MIGRATION PROCESSES.

The formation dynamics for the radioactive trace produced by the accident are examined, and an analysis given of migration processes.

On 29 September 1957 at 16:40 local time, as a result of a chemical explosion in a concrete tank containing radiochemical waste, 670-740 PBq of contaminants, the radionuclide composition of which is given in Table I of paper IAEA-SM-316/55-2, were dispersed.

About 10% of the radioactive substances involved in the explosion (74 PBq) was raised by air currents to a height of one kilometre and moved to the north and northeast in the form of a radioactive cloud towards the towns of Kamenets-Ural’skij and Tyumen’. The total mass of the fission products comprised approximately 6.3 t or 12–15 t of their compounds.

The isotopic composition of the contaminants deposited on the Earth’s surface as a result of the accident was characterized by the presence of $^{144}$Ce + $^{144}$Pr (66%), $^{95}$Zr + $^{95}$Nb (24%), $^{90}$Sr + $^{90}$Y (5.4%) and $^{106}$Ru + $^{106}$Rh (3.7%). The total quantity of the isotopes $^{137}$Cs, $^{89}$Sr, $^{155}$Eu, $^{147}$Pm and Pu was less than 1% (see Table I).

The radioactive trace on the surface of the Earth (it was called the East Ural radioactive trace) formed in two phases:

— The initial radioactive fallout phase;
— The wind migration phase.

The extent of the trace, and the time it took to form during the first phase was determined by the speed of movement of the cloud, and the duration of fallout.
The wind speed in the surface air layer when the discharge occurred was 5–6 m/s over the 100 km long displacement track, and the wind direction was constant. At the displacement height of the radioactive cloud the mean wind speed was approximately 7.5 m/s.

A 74 kBq/m$^2$ (for $^{90}$Sr) radioactive trace formed virtually within four hours in the direction of movement, extending for approximately 100 km; for minimum detectable contamination density values (less than 3.7 kBq/m$^2$ for $^{90}$Sr) the formation time was over 11 hours. Radioactive fallout lasted from a few minutes in the head of the trace, to up to half an hour to an hour in the most distant portion (300 km).

Redistribution of the contaminants owing to wind resuspension was most noticeable during the few days immediately after the accident.

The main sources of secondary fallout were the crowns of trees and soil surfaces. In October–November 1957, before a permanent covering of snow had fallen, redistribution of the contaminants owing to wind resuspension caused a rise in the intensity of local fallout by a factor of 2 to 20–50, which moved the boundaries of the trace 5–10 km to the southeast (the extent of the line of displacement was 20–50 km).

The formation of the radioactive trace was practically complete by 1958, when wind migration was redistributing less than 1% of the original radioactive fallout. Over the next thirty years, wind transfer did not affect redistribution of the contaminants. The stability of the trace boundaries and its geographical distribution remained constant during subsequent years as further measurements showed (aerial gamma photography, on-the-ground beta- and gamma-photography, mass radiochemical analysis of samples).

The radionuclides were fixed in the upper soil layer, and the stability of that fixation was confirmed by their resistance to wind and water migration factors.

The East Ural radioactive trace takes the shape of a long strip 4.5–6 km wide and 105 km long with a total area of approximately 1000 km$^2$ for a fallout density of 74 kBq/m$^2$ for $^{90}$Sr; for a contamination density of 3.7 kBq/m$^2$ for $^{90}$Sr it covers approximately 15 000 km$^2$, of which approximately 40 km$^2$ was subjected to secondary contamination by wind transfer.

Though there were micro- and meso-landscape variations in the distribution of the contamination, the trace had a characteristic surface macrostructure with a clearly defined axis and monotonic decrease in the contamination density along the axis and perpendicular to it. This is due to the fact that wind conditions were relatively stable during the main formation phase, there was no precipitation, and the land surface was relatively level.

The structure of the gamma and beta fields was dependent not only on surface distribution but also on the nature of vertical migration in the surface plant layer, the soil covering, and in watercourses. Thus, after heavy rain the gamma radiation dose strength on the surface of ploughed soils went down by 1.5–2 times (beta radia-
tion decreased by up to 10 times). Penetration of the contaminants downwards increases in the following order according to the nature of the landscape: virgin plots of land with heavy soils, plots of land with light soils (arable land, sands), boggy plots of land, marshes, watercourses.

Ninety per cent of the radionuclides deposited on forest areas were retained by the crowns.

The geographical distribution of the contaminants is shown on the map by isolines which mark the mean contamination density values for a density gradient variation of ±50%.

Owing to the radioactive decay of the isotopes cerium, ruthenium, and zirconium–niobium, by 1965 total activity levels were mainly due to $^{90}\text{Sr}$ and $^{90}\text{Y}$. At present, $^{90}\text{Sr}$ and $^{90}\text{Y}$ account for 99% of the activity and $^{137}\text{Cs}$ for 0.7%.

After the fallout, biogeochemical processes began which altered the distribution of the radionuclides in relation to the meso- and micro-landscape.

As a result of vertical migration owing to absorption by the turf and leaching by rain, there was swift redistribution of the radionuclides through the soil profile. In the spring of 1958, 90–95% of the activity was concentrated in the turf, 0.5–1.5% in living plants, and 5–10% in the mineralized part of the soil. In turf marshes 90–95% of the activity was concentrated in the upper 2 cm layer at this time.

In surface watercourses the initial redistribution of the activity was dependent on fallout processes, mixing, absorption by bottom deposits, inflow from drainage areas, and outflow. The half-periods for elimination from water in lakes varied from 1 to 24 days for $^{144}\text{Ce}$ and from 780 to 1100 days for $^{90}\text{Sr}$. Immediately after the accident the contamination densities in the more highly contaminated rivers were from $4 \times 10^3$ to $28 \times 10^3$ times greater than the pre-accident level.

By the summer of 1958, owing to radioactive decay, absorption by bottom deposits, and natural migration, contamination of rivers by the East Ural trace had gone down by a factor of 150. This in turn caused a drop in the contamination level in lakes by a factor of 20–30. In 1958, in the majority of lakes the total beta activity level in the water was 0.5, in water and shore plants it was 600–3000, and in the muscles and skeleton of fish it was 100 and 600 respectively (Bq/kg per 1 Bq $^{90}\text{Sr}/m^2$).

The downwards migration speed in the profile of unbroken soils was 0.3–0.5 cm/a for $^{90}\text{Sr}$ and 0.15–0.25 cm/a for $^{137}\text{Cs}$. Vertical migration was responsible for the redistribution of radionuclides in the soil profile: by comparison with the initial post-accident period when the $^{90}\text{Sr}$ was contained mainly in the upper 0–2 cm of the soil layer, 25 years later more than 90% of it was in the 2–10 cm layer.

Geochemical circulation of radionuclides in the mineralized portion of the soil (leached chernozem soil, grey forest soil, turfy podzol soil) was characteristically mobile and rose in the following order:

$$^{137}\text{Cs} < ^{90}\text{Sr} < ^{144}\text{Ce} < ^{106}\text{Ru}$$
with maximum movement in turfy podzol soils and minimum movement in leached chernozem soils.

Owing to vertical migration of radionuclides from the turf into the mineralized portion of soils, accumulation of $^{90}$Sr in natural herbaceous plants showed a tendency to rise 6–12 years after contamination took place, and then accumulation subsequently fell. During this time the role played by root uptake increased. On average, the contribution of root uptake of $^{90}$Sr from soils rose from 10–20% in 1957–1958 to 95% in the last decade.

The contribution of root accumulation of radionuclides in ligneous plants became predominant ten years after the accident. During the initial period there was radioactive fallout on the forest canopy and 80–90% of the contamination was retained in the crowns. Eight months later only 10–20% of the initial activity remained in the crowns of leaf-bearing species owing to leaf falls and wind; 40–50% remained in the crowns of coniferous species.

Wind resuspension during the initial period (1957–1958) amounted to $10^{-9}$ s$^{-1}$ (3%/a) of the total amount of contaminants per unit of area, reducing subsequently to $10^{-11}$ s$^{-1}$ (10$^{-2}$%/a). Wind transfer seemed to be limited to a maximum of a few hundred metres and did not result in significant changes in the trace boundaries. Contamination of natural plant life owing to extra-root contamination from wind resuspension and transfer fell from 20–70% in 1957–1958 to 0.2% over 25 years.

Radionuclide biogeochemical migration processes caused the $^{90}$Sr contamination levels in the above-ground biomass of trees to fall from 80–90% in 1957 to 0.02% (pine) and 7% (birch) per unit of area over 20–25 years. During the period where root uptake of $^{90}$Sr was occurring, the mean value of the proportionality coefficients, Bq $^{90}$Sr/kg air–dry matter and kBq $^{90}$Sr/m$^2$, was 2.7 and 0.3 for pine needles and wood, and 12 and 1.8 for birch leaves and wood.

Accumulation of $^{90}$Sr in forest produce (berries, mushrooms) was dependent on the stable calcium requirements of the plants and mushrooms. The $^{90}$Sr proportionality coefficients during the root uptake period were in the range of 0.05–40 Bq/kg air–dry matter for the fruits of bushes and berry plantations, and 0.5–10 Bq/kg air–dry matter for mushrooms, per kBq $^{90}$Sr/m$^2$. These values tally with the proportionality for herbaceous and ligneous plants.

The radionuclide accumulation coefficients in trophic chains, and in dry-route, water and near-water chains, expressed in terms of the ratio of the concentrations in the consuming organisms and the producing organisms, are tied to the complex migration processes whereby mineral substances are transferred to animals higher in the trophic chain, or to species which close the trophic chain. The accumulation coefficient for dry-route herbivorous birds and mammals was in the range of 10–45, in insectivorous birds and mammals it was 20–30, and in herbivorous fishes and water birds it was 1.5–7.5.
The highest $^{90}$Sr accumulation was found in reptiles, herbivorous birds and birds of prey, and rodents — 100 Bq $^{90}$Sr/kg of skeleton per kBq $^{90}$Sr/m$^2$; for $^{137}$Cs the levels in invertebrates, amphibians and reptiles were 10 Bq $^{137}$Cs/kg of body per kBq $^{137}$Cs/m$^2$. The characteristic difference between the behaviour of $^{90}$Sr and $^{137}$Cs in animal trophic chains is that accumulation of $^{90}$Sr in organisms rises as one goes up the trophic chain, and accumulation of $^{137}$Cs decreases.

The concentration of $^{90}$Sr in the water of lakes in the radioactively contaminated territory decreased by a factor of 30 over 25 years. It decreased by 20–30% in the upper layer of bottom deposits (0–5 cm), and by a factor of 35 in fish.

Prognoses of the radiation situation in the territory contaminated by the East Ural trace are based on $^{90}$Sr levels, since this is the main component of the contamination at present (99%). Over 25 years the area of the 74 kBq $^{90}$Sr/m$^2$ radioactive trace has decreased by over 35%. It is expected that it will decrease by 80% over 100 years.
LESSONS LEARNED FROM RADI OLOGICAL ACCIDENTS

(Session IV)

Chairman (Part 1)

Y. MOURÈS
France

Chairman (Part 2)

M.A. FERADAY
Canada

Chairman (Part 3)

S. CHAKRABORTY
Switzerland
Invited Paper

INTERNATIONAL GUIDANCE ON OFF-SITE POST-ACCIDENT ASSESSMENT AND RECOVERY OPERATIONS: DEVELOPMENT AND STATE OF THE ART

S. CHAKRABORTY, Y.G. GONEN*
Swiss Federal Nuclear Safety Inspectorate,
Würenlingen,
Switzerland

Abstract

INTERNATIONAL GUIDANCE ON OFF-SITE POST-ACCIDENT ASSESSMENT AND RECOVERY OPERATIONS: DEVELOPMENT AND STATE OF THE ART.

The development of international guidance on emergency planning is reviewed in the light of past approaches to nuclear safety, specific accidents and other relevant factors. The nature and limitations of international work and their effect on the resulting guidance are discussed. The involvement of various international organizations is discussed. The most important recent publications of the IAEA are reviewed with attention to some specific issues. Recommendations for the further development of the existing guidance and its implementation are given.

1. INTRODUCTION

The development of international guidance on emergency planning and preparedness in general and on its offsite components in particular — including the assessment of the off-site consequences, their mitigation and the required recovery operations — can be properly appreciated only in the light of past approaches to nuclear safety and some specific events — accidents — which on reflection can be seen as turning points in its evolution.

In retrospect it seems that perception of the need for, and the desirable extent of, emergency planning and preparedness have been influenced primarily by the following factors:

— the size of NPP units
— the number of units, i.e. the spreading of nuclear power
— perceptions regarding the probability and the possible effects of a major nuclear accident on the public in the NPP’s surroundings (nearest population centres)

* Permanent address: NRCN, 84190 Beer Sheva, Israel.
— the limits for acceptable consequences established by the regulatory bodies (in the case of a major accident)
— accident experience, and
— public sentiments and reactions to the above factors.

2. DEVELOPMENT OF EMERGENCY PLANNING

Table I illustrates the quantitative aspects of the development of commercial nuclear power. It shows, as a function of time, the number of operating nuclear units worldwide and gives an indication of their power range. The number of operating research and experimental power reactors is also noted.

2.1. Early years

Until the early nineteen-sixties, research, test and experimental power reactors dominated the scene. The only country with a sizeable power reactor programme was at that time the United Kingdom. The average power of NPPs just reached the 70 MW(e) mark and the largest units were approaching three times this size. The

<table>
<thead>
<tr>
<th>Year</th>
<th>Number of units</th>
<th>Total net capacity (MW(e))</th>
<th>Power of largest unit (MW(e))</th>
<th>Average power of all units (MW(e))</th>
<th>Average power of new units (MW(e))</th>
<th>Number of research and experimental reactors</th>
</tr>
</thead>
<tbody>
<tr>
<td>1955</td>
<td>1</td>
<td>5</td>
<td>5</td>
<td>5</td>
<td>5</td>
<td>27</td>
</tr>
<tr>
<td>1956</td>
<td>3</td>
<td>105</td>
<td>50</td>
<td>35</td>
<td>50</td>
<td>37</td>
</tr>
<tr>
<td>1960</td>
<td>16</td>
<td>1 106</td>
<td>200</td>
<td>70</td>
<td>77</td>
<td>165</td>
</tr>
<tr>
<td>1965</td>
<td>48</td>
<td>5 243</td>
<td>265</td>
<td>109</td>
<td>130</td>
<td>324</td>
</tr>
<tr>
<td>1970</td>
<td>89</td>
<td>16 648</td>
<td>620</td>
<td>187</td>
<td>278</td>
<td>363</td>
</tr>
<tr>
<td>1975</td>
<td>253</td>
<td>13 6809</td>
<td>1 067</td>
<td>541</td>
<td>733</td>
<td>~370</td>
</tr>
<tr>
<td>1980</td>
<td>283</td>
<td>16 0732</td>
<td>1 149</td>
<td>568</td>
<td>797</td>
<td>~370</td>
</tr>
<tr>
<td>1985</td>
<td>380</td>
<td>25 5000</td>
<td>1 450</td>
<td>671</td>
<td>970</td>
<td>~370</td>
</tr>
<tr>
<td>1988</td>
<td>432</td>
<td>31 3500</td>
<td>1 450</td>
<td>726</td>
<td>1 125</td>
<td>~370</td>
</tr>
</tbody>
</table>

a The average power of research and experimental reactors, excluding US production reactors and the Superphénix is approximately 10 MW(th), with less than 10% above 40 MW(th).
US built experimental and naval reactors were smaller and so were most of the units built in other countries.

Licensing was based on the maximum credible accident (MCA) approach (developed in the USA). According to this concept the operator had to show that even in the case of the most severe imaginable ('credible') accident the exposure of the most exposed members of the public from the associated release of fission products would not exceed the limits set by the regulatory body. Proper siting of reactors, provision of large exclusion and low population zones, sufficient distances and optimal direction from population centres (down the dominant wind) and containment seemed to result in satisfactory solutions and to provide adequate safety.

The hazard assessors did not mind exaggerating the size of the release assumed for the MCA even if no technically supportable scenario could be identified to justify the release considered, as long as they could show that the consequences were within the limits set.

Public opinion ranged from strong support of nuclear power generation to rejection. However, the reasons for opposition were mainly on the grounds of economics, conservation or political opposition to centralized 'big industry' [1-3]. Governments or utilities (in the US) were generally able to accommodate most of the opposition.

There were several reactor accidents during this period (e.g. at the NRX and NRU in Canada, the Westinghouse Test Reactor and SL-1 in the USA, the Windscale reactor in the United Kingdom), but these — apart from the Windscale accident in 1957 — had no significant off-site consequences. The off-site consequences of the Windscale accident could be considered relatively mild, due to two factors — the low population density in the vicinity of the plant and the composition of the release, which consisted mostly of noble gases and relatively short lived iodine isotopes. Although these accidents were not considered relevant to the NPPs existing or being constructed at that time because of the large technical differences between the facilities damaged by them and the power plants, they did strengthen the understanding that unexpected scenarios might arise. In spite of these, the perception that even a relatively major reactor accident should have only limited, local effects prevailed.

The WASH-740 [4] report, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Plants, issued in 1957 in connection with the enacting of the Price-Anderson Act, concluded that although a very large accident with extremely severe off-site consequences could happen in theory, the probability that it would occur was so remote that it should be considered 'incredible'. This can be appraised from the fact that the analysis of class 9 accident consequences was not required in US regulatory process until the mid nineteen-seventies at least [5].

Technical improvements in NPPs were made; their protection against accidents was enhanced by the elaboration of the defence in depth concept and the addition of engineered safety features, but emergency planning and preparedness were not at that time considered necessary.
2.2. Years of growth

In view of the fast growth of unit size and the accelerated construction rate of NPPs in the following years, these positions could not be sustained much longer. The MCA approach became unsatisfactory for proving compliance with the limits in force. Consequently the MCA-based licensing procedure was replaced by the design base accident and single failure criteria approach, considered more realistic. The assumption was made that multiple coinciding failures are too remote possibilities to guard against by extra engineered safety features. In parallel there was an erosion of the 'protection by remote siting' concept due to the economic penalties associated with it, the increasing distances it would have required with increasing unit size and the belief that the absence of this protection feature could be sufficiently compensated for by enhanced engineered safety features [6]. The excellent safety record of nuclear power plants up to that date seemed to validate these assumptions.

On the other hand, radiation risks in the stochastic range were gradually considered to be more severe than had been thought earlier and the public confidence in the licensing authorities lessened. The radiation risks which had been considered acceptable by the authorities — based on comparison with other risks of life — were seen by increasing numbers of the public as unacceptable, especially following the start of the radiation risk debates.

In the mid nineteen-sixties radiological emergency plans started to appear. The first plans addressed only on-site consequences as general accident experience implied no need for more. The central issues (concerns) in these plans were the control of radiation fields and facility contamination, the prevention of exposure to on-site personnel, medical care of overexposed or seriously contaminated workers, decontamination, etc. As can be seen from the papers presented in the IAEA symposium [7] on Handling of Radiation Accidents, held in 1969, such emergency plans had been established mostly following initiatives from within the facilities and less often following regulatory requirements. Most of them were essentially on-site incident response plans, purposefully co-ordinating the intervention in industrial (fire) and radiation related incidents, should they happen, using the facilities' existing rescue type resources.

More extensive planning which included also low population zones was performed in the United Kingdom and in some US facilities, mainly due to the lessons derived from the Windscale accident. However, it seems that with regard to the off-site areas these plans remained on the conceptual level [8].

In parallel, studies indicated that at the size of NPPs reached by this time (~ 600 MW(e)) and those under construction (~ 750 MW(e)) the off-site consequences of a major accident might be of such magnitude as to necessitate the implementation of off-site countermeasures to protect the public at large. It was also realized that to ensure the full effectiveness of countermeasures the response would have to be fast. Such a target could not be achieved without pre-accident planning, preparations (logistics, training, etc.) and co-ordination.
The estimates of stochastic (including genetic) effects of low level exposures by scientific committees (UNSCEAR, BEIR) aroused public concern, and fuelled the opposition to nuclear facilities. The opposition also used safety and environmentally related argumentation, rarely upheld in the courts, to delay and to disturb the construction of NPPs [9–10]. The effect this achieved was the spread of fear of radiation and the strengthening of public concern even with regard to exposures well within the stochastic range.

In view of such developments, several national authorities started to investigate the possible (realistic) off-site consequences of major reactor accidents (called later class 9 or severe accidents) and to build up assessment and limited intervention capabilities in addition to the facility based resources (for example — ARAC in the USA, radiological assistance teams in France and the Federal Republic of Germany and public alarm system in Switzerland). In spite of this, the WASH-1250 report [11] issued in 1973 by the US Atomic Energy Commission, although it notes the availability of countermeasures to mitigate the consequences of conceivable (i.e. ‘credible’) accidents in the vicinity of NPPs, makes no explicit reference to emergency planning and preparedness and still refers to severe accidents as events of such low probability that there is no practical need to consider them.

The generally accepted positions in these years can be summarized as follows: It was understood that a major reactor accident having off-site consequences was conceivable; however, the operators as well as the regulators believed that the probability of occurrence of such an accident was extremely low and that the public residing outside the low population zone would be sufficiently safe even in such a case. They realized the availability of countermeasures, such as evacuation, sheltering and food controls and their value in mitigating the accident’s consequences but thought that their implementation should be limited mainly to the low population zone so that it was considered manageable.

Due to the very low estimated probability associated with such events, they felt that it was not justifiable to translate these ideas into actions such as emergency planning (physical preparations, organizational arrangements, assurance of logistics, training, etc.). They thought that if a severe accident ever happened they would be able to handle it with improvised solutions based on general information and knowledge, including experience with evacuations in other fields, especially as actions would be needed only in a small area and would concern a limited number of people.

2.3. Start of consolidation

It was clear that using a deterministic calculation combined with crude estimation of the probabilities would not be sufficient to prove the case and that the more sophisticated probabilistic approach is needed to do so.
The Reactor Safety Study (the Rasmussen Report) [12] was commissioned in 1972 for this purpose. This landmark study was the first which applied thoroughly the probabilistic methodology to estimate the risks from accidents in NPPs. It considered their whole spectrum, including severe (class 9 or beyond design basis) accidents and in estimating the consequences it considered also the effects of available mitigating actions and a variety of countermeasures for the protection of the public. The draft completed in 1974 and the final version published as WASH-1400 in 1975 [12] showed that although the probability of a major reactor accident was very low compared to the frequency or probability of other similar sized catastrophes from natural or man-made sources, nevertheless, if all operating reactors are considered, such an event is not an extremely remote one which should be dismissed. The criticism of the Report by the Lewis Committee as regards its treatment of the uncertainties only emphasized the point. The other conclusion of the Report, that timely implementation of countermeasures is essential to mitigate the possible consequences, went unchallenged.

The Browns Ferry accident (1975) — a near miss — provided a further impetus towards emergency planning and preparedness. This was in addition to the technical conclusions derived from the event regarding fire prevention, fire fighting and the need for the physical separation of redundancies (equipment and cabling) in NPPs.

2.4. Evolution of international guidance

Possibly there was some reluctance to handle the problem on the national level. The situation became ripe for treatment on the international one.

We are all aware of the processes by which most international organizations, including the IAEA, develop guidance on matters of interest. However it is less often realized that these processes have a profound effect on the resulting guidance. After the identification and definition of the subject, usually a small group of experts — sometimes even individuals — are entrusted with the preparation of a working paper, which serves later as a basis for discussions in larger groups. As there is no 'international work' or international knowledge as such, obviously the experts preparing the working paper are drawing mainly from the work done on the issue in their home countries and their own organizations, and therefore the document produced reflects mostly the practices already accepted there. As the experts invited are usually from industrially developed countries, leading in the field of nuclear development, the nature of the propositions and the value judgements behind them are primarily applicable to countries and situations similar to theirs.

In the composition of the working groups themselves there is no basic change. The participating experts usually represent already established practices and try to ensure that the resulting international guidance shall reflect these or at least that it should not lead to contradiction with them.
Therefore, the resulting international guidance does not necessarily consider the viewpoint of developing countries, but represents an agreement or a compromise between the approaches of the developed ones. Due to these circumstances the international guidance is often not concise and does not present appropriate methodologies which allow the derivation of specific solutions appropriate to local conditions. The recommendations have to be interpreted in different ways, sometimes a complex task, which is left to the consideration of national authorities.

Often the same issue is treated by different international bodies, with partially overlapping participation of experts and resulting in partially overlapping recommendations, in which the differences are expressed in a rather cautious manner.

2.4.1. First comprehensive guidance

The basic IAEA publication on emergency planning and preparedness (Safety Series No. 55) [13] was an exception in some ways.

The problem has already been realized as a valid one, some of the background thinking and experimental evaluations have been performed in the different countries represented in the working group, but the conclusions have not yet been included in national regulations. There were no established national schemes in force and there were no binding or limiting international recommendations on the issue. Therefore, as the field was almost free, the participants' views were rather freely introduced.

The resulting document was the first one authored by experts from the nuclear establishment which seriously assumed and acknowledged that major reactor accidents represent a real, although remote, possibility and outlined the major issues to be considered, resolved and prepared in this connection. The document was ready for print in 1978 but due to the Three Mile Island (TMI) accident its publication was delayed two more years. Much of what developed in the field subsequently is the elaboration of issues and ideas from this publication and its updating according to technical developments in fields relevant to the subject (computers, communications, instrumentation, robotics, etc.).

The TMI accident in some ways validated the approach of the Safety Series No. 55 and, following it, emergency planning and preparedness became the fourth now acknowledged layer of the defence in depth approach, made compulsory by the regulatory authorities in most countries for every nuclear installation. This was the real breakthrough in the history of emergency planning.

2.4.2. Post-TMI years

A surge of publications on many aspects of emergency planning and preparedness followed, with the IAEA assuming the leading role among the international bodies and organizations involved. To date the IAEA has published over twenty
documents dealing with almost every aspect of this complex subject, which became one of the central ones in the IAEA nuclear safety activities (Annex 1).

The ICRP, WHO, CEC and the NEA also referred to problems of radiological emergencies, but dealt mostly with specific aspects of them, in not more than a few publications each (e.g. principles, intervention levels, etc.) (Annex 2).

Even the imagination of the group which produced Safety Series No. 55 regarding the possible scale of a nuclear emergency did not match reality as it materialized in the Chernobyl accident. This accident gave rise to a new set of problems resulting from the scale of the accident and the geographical extent of its consequences. The additional issues include the necessity of notification on the international level, the scale of intervention required, the scale of international assistance needed, the scale of decontamination to be performed, food contamination in connection with international trade and the handling of the situation in the so-called far field. Still, some further issues may arise.

It seems ironic that a document prepared by an IAEA Working Group on post-accident assessment and recovery operations in radiation environment which touched upon some of these subjects suffered the same fate as Safety Series No. 55 — it was completed in late 1985, before the accident at the Chernobyl unit 4, but unfortunately its publication has been delayed for years.

There have been a few radiation accidents (e.g. in Mexico and in Goiânia, Brazil) which showed that not only nuclear reactors, but also large radiation sources can cause significant harm to the public. A certain level of emergency planning for such sources — on a national level — should also be considered, although the scale required is smaller compared with what is needed in connection with nuclear installations. The resources prepared in connection with nuclear emergencies can satisfy these needs. However, the problems of organization and logistics must be resolved in a different manner, as the location of a possible event is not fixed, and cannot be predicted.

Emergency planning and preparedness, now well established in the regulatory framework of all countries, has become in the USA a focal point in challenging licences for NPPs, as illustrated by the Seabrook and Shoreham cases. These plants cannot assume full power operation due to concerns regarding the success of any evacuation in the event of a major accident.

3. REVIEW OF SPECIFIC TOPICS

It is important to realize that there is a significant time lag between the accumulation of new knowledge, information and experience and its incorporation into any guidance, especially at the international level. This time lag results from the slow pace of the necessary processes, the evaluation and generalization of the new information and the complex nature of developing or updating international guidance.
Most of the existing international guidance on emergency planning and preparedness reflect the post-TMI situation, i.e. the lessons from TMI have already been incorporated with some extrapolation. However, the Chernobyl experience is only partly reflected, mainly with regard to a few burning issues (e.g. the IAEA conventions on early notification and mutual assistance and the setting of limits for radioactive contamination of foodstuffs in international trade within the framework of the Codex Alimentarius). Generally, the incorporation of the Chernobyl experience will take more time. In the following, attention will be given to both aspects of development. The review of existing guidance is based mainly on Refs [14, 15].

3.1. Assessment of radiological impact

The purpose of the assessment of radiological impact for emergency planning is to provide a sound technical and scientific basis for decision making for adopting and implementing protective measures for the public. One has to distinguish between a priori and a posteriori assessment of radiological impact.

The state of the art of accident assessment is well advanced. For a priori assessment, the probabilistic risk assessment methodology is becoming very useful (for understanding the full spectrum of possible accidents and their consequences. In spite of the large uncertainties inherent in this methodology, the emergency planning can be based on informed judgement on the probabilities of different accident scenarios and their associated consequences. As the number of completed and reviewed probabilistic risk assessment (PSA) studies increases, the databases, the models used and the methodology itself are improving.

The most critical parameter during an accident is the source term and the probable time of release. The source term is defined as the magnitude, composition, heat content, timing and other characteristics of the release of radioactive material into the environment. Knowledge of source term for severe accidents has progressed appreciably, although it is far from complete. However, severe accident scenarios can now be better described for a priori assessment of the radiological impact on the environment. An attempt could be made to introduce guidance on accident scenario based handling of emergencies.

The methodology for estimating source terms (a posteriori) on the basis of field measurements has also been developed further. A capable system of measurements of radioactivity levels in the plant and other monitoring data of accident conditions is required for this purpose. Significant advances have been achieved in the field of remote sensing, monitoring and data acquisition systems. Such countrywide systems are being installed at present in several European countries. The required off-site field and laboratory measurement capabilities are described in detail in Safety Series No. 86 [14].

Data handling and analysis, and communication, are of vital importance for making proper decisions on protective measures.
As a large amount of specific data about the condition of the plant and its environs has to be acquired, stored and displayed, computer aids are inevitable. Today, workstations have reached such a stage of development that the use of a computer for emergency planning and handling is undoubtedly cost-beneficial. Suitable hardware is available on the market; however, further development of realistic codes tailored to specific needs (such as real time assessment) is still necessary. It has now become easier to make parametric emergency planning studies and to train decision makers and technical assessors through exercises. The next development may be the introduction of computerized decision support systems, using artificial intelligence, which can help decision makers and technical assessors to perform their tasks more effectively and confidently during the critical and stressful situations of a real accident.

3.2. Post-accident assessment and recovery operations in a contaminated environment


This publication reviews the radiological aspects of the late phase which is characterized by the contaminated environment and chronic exposure pathways. The chronic exposure pathways deal with the transport and accumulation of long lived radionuclides in ecosystems and with internal exposure following the ingestion of contaminated foodstuffs. General techniques for long term environmental monitoring, late phase protective measures, criteria needed to control radiation exposure of different groups of population, methods for interdiction and techniques of decontamination and fixation, socioeconomic considerations and decision making about protective measures and recovery operations are extensively discussed. Guidance is also given on organizational aspects of assessing, evaluating, managing and mitigating the consequences over a long period of time.

It would be worthwhile comparing the advice and recommendations regarding the late phase covered in this publication with the experience gained in the aftermath of Chernobyl.

3.3. Intervention levels

Intervention levels deal with the course of actions to implement protective and restorative measures and are the key elements of radiological emergency management. When an accident occurs the source of exposure is, by definition, not under control. Therefore, any consequential exposure can be avoided to a great extent only by some form of intervention, which interferes with normal living conditions and involves risks other than those from radiation.
There is general agreement on the basic principles for establishing intervention levels, as follows:

- Serious immediate (early) health effects, so called non-stochastic effects, must be avoided as highest priority.
- Stochastic health effects such as cancer which may result from low doses should be limited to the extent possible, taking into account economic and social considerations as well as associated concurrent risks.

The common practice is to set the intervention levels in terms of dose levels and the derived ones in terms of contamination levels in the relevant media. As the level of interference with normal living conditions and the associated risks and costs depend on the protective measure considered, often specific levels or ranges are established for each measure. These levels are related to the radiation risk via different models and were set at levels deemed to represent acceptable risks in the event of a major nuclear accident, a proper balance between health and other considerations. Reference [15] contains a detailed discussion of these issues. The Chernobyl experience showed that these levels were appropriate or even somewhat low for an indigenous event. However, outside the USSR the system succumbed to social and political pressures and to the will of the decision makers to act positively. Consequently, in different countries different countermeasures have been taken, some of these of limited usefulness, and confusion and anxiety arose. The most striking differences were those related to the contamination levels in foodstuffs, derived intervention levels (DILs).

Several international organizations have therefore been working in the past few years on the international harmonization of intervention levels, accounting inter alia also for the shift in the risk appraisal, with greater emphasis on sociopolitical considerations.

It is important that the outcome of these efforts be incorporated into existing international guidance as soon as possible to improve the management of radiological emergencies, the understanding of the public, and to restore public confidence in the appropriateness of the protective and restorative measures.

3.4. Organizational aspects

The need for preplanning for the late phase, the recovery operations and the restoration of normal living conditions, had not always been recognized, probably due to the lack of realization of the possible magnitude of such an undertaking. The recovery phase from the Chernobyl accident, which became a multibillion dollar project employing huge human and material resources, probably indicates the limits. A large number of issues must be addressed, each one in itself an immense task, which may last for many years. Reference [15] deals concisely with many of these issues.
The lesson from Chernobyl in this context is that the possible scale of recovery operations from a major nuclear accident must be reassessed. The emergency planning should define among other issues also an organizational frame able to handle a complex undertaking of enormous scale. This organization must have wide powers, enabling it to co-ordinate and to control all the relevant activities, including final decision making on priorities and actions and on the allocation of resources.

Depending on the country, certain actions possibly needed in the late phase (the commandeering of land, for example, with or without compensation) will require a legislative and/or regulatory basis, which can be prepared in advance. The preplanning can help the recovery organization to respond smoothly to the socioeconomic and ethical problems associated with the different protective measures or recovery activity.

3.5. Decision making on protective measures in an emergency

Decision making in an emergency related to protective measures and to their implementation is the culmination of various technical assessment processes.

In general terms, the basis for decision making about protective measures is summarized below, in line with Chapter 9 of Ref. [14].

3.5.1. Information assessment and potential radiological effects

All relevant information related to the nature and the course (past and projected) of the accident must be assessed and summarized. This entails gathering all the technical data bearing upon the accident and evaluating the information using the best available techniques.
The results of this assessment must be prepared and presented to the decision maker in a comprehensive, yet easily understandable fashion.

The assessment must be updated at regular intervals and whenever a change in the conditions is observed, using sound technical expertise and models, since it is the basic input into the decision making process.

Understanding of the radiological hazard is also essential. This includes both acute as well as chronic effects associated with the potential release, the geographical extent of each risk level and the number of individuals exposed to each.

3.5.2. **Frame for protective action decision making**

The protective action decision making process for mitigating off-site radiological consequences of accidents in nuclear installations is based on an initial assessment of projected consequences and its updates. The decision making strategy can be aided by using three basic frames described below.

**The time frame**

The different types of decisions related to protective action strategies become clearer when the development of a radiation emergency is divided into separate time phases [13-15], namely:

1. **The pre-release phase** — The precautionary protective measures should, if possible, be implemented prior to release of any radioactivity. Once a release is seen to be unavoidable, a ‘pre-release phase’ begins in which decisions on protective measures will be needed if the aim of dose avoidance (or reduction) for the public is to be achieved.

2. **The release and post-release phase** — This phase can be divided into three sub-phases. In the early phase, extending for several hours to days, the main risks are due to inhalation of, or irradiation by, the passing plume.

3. **The intermediate phase**, which begins when the airborne plume has passed, extends for several days to weeks, and the risks are due to direct external radiation from radionuclides deposited on the ground, inhalation of resuspended particulates, and internal exposure via the ingestion pathway.

4. **The late phase**, which may extend from weeks to years. The risks are mainly due to internal exposure via ingestion, and external radiation from the contaminated environment.

**The geographical frame**

The area surrounding the facility should be divided into Emergency Planning Zones (EPZs), usually defined by circles centred on the plant. The size of the EPZ
is governed by the magnitude of potential releases, the local demographic and geographical features and the national policy. These zones provide a basic geographical frame for protective action decision making, both within and outside the EPZs. Sectors and segments within the EPZs constitute the next level of geographical subdivisions around the facility. A given protective measure may thus be taken to apply uniformly to a whole EPZ or to one or more sectors or segments of an EPZ depending on the (projected) plume trajectory according to the real time meteorological conditions.

The technical frame

Decisions should be based on a set of planned (but tentative and flexible) responses to certain scenarios or postulated events. These intervention criteria must be prepared as part of the emergency planning process.

Two types of emergency reference criteria can be envisaged, namely: (1) in-plant emergency reference condition (ERC), and (2) intervention levels.

Accident precursor conditions within the plant may call for specific actions on and/or off the site. These ERCS can be expressed as broad categories of plant conditions, which, if reached, should trigger decision making based on the likelihood, the timing and the projected consequences of the potential release.

The alarm point, or warning of an ERC, would then be set at a level chosen to correspond to a selected dose that might be received by an individual at a selected off-site location under conservatively chosen dispersion conditions. An 'attention point' may also be useful to warn of plant conditions or changes preceding an emergency.

In such situations it may be difficult to select the optimal responses due to the probabilistic nature of the available projections. However, to take preparatory, easily reversible steps may be appropriate, with their extent depending primarily on the estimated likelihood of a release following the ERC. These steps might include alerting the emergency organization to preset levels, warning the public or parts of it in the vicinity of the site, establishing communication channels and seeking outside assistance or additional resources for the implementation of protective measures and of other required activities. The existing guidance does not deal with this issue in detail.

Intervention levels pre-establish a range of dose values, defined by an upper and a lower bound. Below this range, no action would normally be necessary. Beyond the range, action must be taken to reduce exposure and/or to minimize the consequences of unavoidable exposures, by means of protective measures. Between the two levels some actions may or may not be taken, depending on the circumstances. A flexible specification of intervention levels is more useful than a rigid single number.
It is important to note that different intervention levels are required for emergency workers as opposed to the general public.

3.5.3. Appreciation

Any decision making process which has multiple objectives becomes complex. Since in the late phase socioeconomic considerations the problem of psychological impact on the public and political implications will inevitably enter the decision making process (besides the technical and health aspects), it turns into a very complex one in which values and trade-offs must be taken into account and arising conflicts resolved.

At the technical level, conflicting recommendations may arise from:

(1) Different assessment or appreciation of the same information; or
(2) Different assessment or appreciation arising from different data.

The resolution of conflicting recommendations is, essentially, the heart of the decision maker’s task. At the forefront of his or her consideration must always be the need to ensure public health and safety.

For the implementation of decisions, resources (human and material) are needed; without them the implementation is not feasible. The possibly required resources should be assessed in the plan, they should be identified and their availability assured.

In a real situation the available resources might not be sufficient for various reasons, imposing constraints on the decision. The decision maker’s task in such cases is to establish priorities and to optimize the allocation of the available resources accordingly.

The IAEA’s Safety Series No. 86 [16] describes the various aspects and attributes of protective measures and also provides examples of situations in which the decision making regarding protective measures is difficult.

The same publication contains a useful example of a procedure (checklist) for decision making regarding the implementation of protective measures. However, in this example neither constraints nor multiple objectives are considered. In the recovery phase both of these will probably be important, therefore the decision making will be much more complex. Reference [15] deals with some of these issues. Appropriate checklists can be developed for this phase.

It should be emphasized that decision making is a continuous activity. Many decisions will be needed, often separately with regard to each affected locality or area; therefore the formalization of the process could be beneficial.

The complexity of decision making in emergency situations justifies training and exercises in simulated conditions. This should be considered an important part of maintaining emergency preparedness. Such exercises can also be used to develop decision aiding techniques and aids, including computerized ones.
3.6. Communication with the public and the media

The importance of the public communications issue has to be stressed repeatedly. It is essential to create mutual confidence and trust between the public, the authorities, the emergency organization and the facility's operating organization. A public communications programme should be developed and implemented. The programme's purpose is to educate the public. It would be prudent to introduce this programme with the help of the media, because the media can influence public attitudes and Government's actions considerably. It should cover beside the fundamentals of radiation protection and nuclear safety issues an explanation of the emergency arrangements, in the facility and off the site, the rationale of the countermeasures planned and the communication channels to be used for alerting and informing the public.

4. RECOMMENDATIONS ON FURTHER DEVELOPMENT OF GUIDANCE

A great number of publications (over 25) on emergency planning, preparedness and handling from different international organizations are available, dating from 1980 to the present. These represent the international guidance on the issue. The information base for emergency planning is now broad enough. However, it is difficult to digest all the old and new information. There is also some concern about differences and inconsistencies between some of the documents.

The time has now come to publish a unified document (such as an emergency handbook or emergency planning guidebook) dealing with all aspects of emergency planning and the associated implementation procedures to be published. Such a book is badly needed. This publication may replace the existing ones and avoid possible inconsistencies and overlapping. For this reason feedback from Member States on the existing documents would be desirable. The envisaged document should also reflect the experiences gained and lessons learned from recent accidents. A large amount of valuable information and data on intervention levels, evacuation, relocation, decontamination and other new issues, such as medical treatment of radiation injuries and long range atmospheric transport and dispersion, has been accumulated in the USSR from the Chernobyl experience. It is recommended that this information and data should be reviewed by teams of experts, in order to incorporate these into an up-to-date guidebook.

Recommendations should be developed in the area of computer based decision support systems to improve emergency preparedness and emergency handling in a real situation. Financial and expert support for the development of such a system would be beneficial to all countries.

The guidances given in such documents can be 'living' in nature only if the IAEA continues with systematic training courses to promote their implementation on
the national levels. In order to facilitate mutual emergency assistance, especially between neighbouring countries, consistency must be achieved in national emergency planning (in many cases down to the level of basic procedures).

Much public confusion and concern has arisen from the different guidance given on the consumption of contaminated foodstuffs between countries, and in some instances within countries. This should not occur again because any credible emergency planning rests upon public confidence. The basic principles of intervention criteria or levels should be common to all emergency planning. A general consensus on the rationale behind the introduction of protective measures in the different time phases of an accident has to be reached — the earlier the better.

The development of messages that will inform the public about risks to their health and encourage them to follow the emergency response instructions is a vital step in the communication process. An informed public will be in a better situation to understand and to follow instructions regarding protective measures in case of need, to realize their merits and accept their weaknesses. Public education in understanding the concept of emergency planning should be a short term goal. Effective communication strategies for reaching and communicating with the public and the media are urgently needed.

Several major and serious industrial accidents which have occurred in recent years have demonstrated that environmental risk problems call for the utmost care and vigilance in managing them, both by preventive tools and emergency preparedness for restorative measures. Internationally concerted research efforts for effective decontamination techniques of the environment and new methods of cultivation of low level contaminated land areas are highly recommended.

REFERENCES


Annex I

LIST OF SELECTED IAEA PUBLICATIONS RELEVANT TO EMERGENCY PLANNING AND PREPAREDNESS (POST-1980)


LIST OF SELECTED PUBLICATIONS OF OTHER INTERNATIONAL ORGANIZATIONS RELEVANT TO EMERGENCY PLANNING AND PREPAREDNESS


WORLD HEALTH ORGANIZATION, Derived Intervention Levels for Radionuclides in Food, Guidelines for Application after Widespread Radioactive Contamination Resulting from a Major Radiation Accident, Geneva (1988).


THE PSYCHOLOGICAL IMPACT OF THE RADIOLOGICAL ACCIDENT IN GOIÂNIA

A.B. CARVALHO
Comissão Nacional de Energia Nuclear,
Rio de Janeiro,
Brazil

Abstract

THE PSYCHOLOGICAL IMPACT OF THE RADIOLOGICAL ACCIDENT IN GOIÂNIA.

Since Nagasaki and Hiroshima, nuclear energy and radioactivity have been associated with collective annihilation, so any accident involving nuclear or radioactive artifacts causes intense fear. In September 1987, a caesium medical source was dismantled in the city of Goiânia, Brazil, by two scrap dealers. As a result, four people died and 183 were internally and/or externally contaminated. The victims belonged to a low socioeconomic class and did not understand the meaning of radioactivity. Fears started when the first deaths occurred. The radioactivity syndrome is similar to AIDS because of the isolation, stigmatization, and the fact that the nursing staff wear a mask. During hospitalization, serious behavioural disturbances were triggered. The hospital staff had only theoretical information and were shocked at the sight of radiodermatitis. Some felt panic, were unable to think clearly, felt impotence or fled, while others felt scientific curiosity. Social workers helped the personnel of the Comissão Nacional de Energia Nuclear coping with the population but were stressed by fear and sorrow. A civil defence officer accustomed to catastrophes was shocked at the segregation suffered by the victims. The emotions of the specialists who worked on decontaminating the city varied. Many felt impotence dealing with the population’s panic; some exposed themselves unnecessarily to radiation, while others had phobia inside the protective clothing and masks. A few stated that they felt nothing. The fear of the population was fuelled by the press, rumours and lack of information and in consequence there was much discrimination. Some people presented blisters and red skin and yet had never been near the site of the accident. This could be explained as the onset of disorders in people who already had a predisposition. Psychologists trained in the basics of radiation should have been present to help these people cope with their emotions, from the beginning of the accident.

1. INTRODUCTION

"It would have been good to have had a psychologist with us here in Goiânia, from the very beginning; not just to give support to the CNEN people, but to investigate and understand everything that was happening with the people’s feelings, their emotions. So many things happened, and these things need to be documented, not as journalism, but as a study of human behaviour. The pressure was too much to stand..." [1].
It can be supposed that the terms ‘nuclear’ or ‘radioactive’ are associated with collective annihilation of life on the planet and for this reason they inspire dread. As a consequence, an accident deriving from nuclear/radioactive materials or structures creates a strong impact on the public. The danger is invisible; it cannot be sensed or fled from and its effects are still not entirely known but include manifestations that have no time limit, such as cancer or genetic mutations [2].

In mid-September 1987, in Goiânia, State of Goiás, Brazil, two scrap dealers removed part of a $^{137}\text{Cs}$ teletherapy unit from the abandoned and partly demolished Goiânia Radiotherapy Institute, causing an accident that killed a child, a woman and two men, while 183 people were internally and/or externally contaminated [3]. No one was prepared for a disaster of such magnitude and the greatest shock was suffered by those who first came into contact with its victims.

This paper is a collection of reminiscences from the Goiânia accident. Its objectives are to describe the psychological impact caused by it, describe the behavioural disturbances arising from this impact and demonstrate the need for psychological preparedness, that is, for creating mental health teams to integrate from the beginning with other working groups, in any nuclear accident or radiological emergency.

Throughout the text, repetitions of the same topics are found and these have the purpose of emphasizing the importance of facts that kept repeating themselves.

The first selection is a description of the impact on the victims. In succession there are descriptions of the emotions suffered by the hospital staffs, by the support groups, by the personnel of the Comissão Nacional de Energia Nuclear (CNEN) and finally by the population. In each section there are psychological lessons to be learned from Goiânia. Conclusions are presented at the end of the paper.

2. METHODOLOGY

The subjects were interviewed in their working areas, or in the case of the victims, wherever they were situated at the time. All interviews were conducted individually, with the aid of a tape recorder. When no tape recorder was used, the interviews were transcribed afterwards, from notes. In some cases a questionnaire was used.

3. IMPACT ON THE HOSPITALIZED VICTIMS

“...L is a child, a boy of 14, who seems younger than his years both physically and mentally. One day, in the hospital, he got hold of a magazine which said that caesium causes leukaemia. He read this and didn’t want to eat anymore. He didn’t want anything, he just cried...” [4].
Most of the people who had direct or indirect contact with the radioactive material belonged to a very low socio-economic-cultural stratum, existing under difficult conditions. They were odd-job junkmen with ages ranging from 14 to 40 and some presented low intelligence coefficients or personality disturbances [4].

After the triage at the stadium, those found to have light external or internal contamination were sent to FEBEM, the State Foundation for the Care of Minors. Most of them were teenagers, highly nervous and irritable, full of fears of the invisible and incomprehensible 'something', which was causing progressive physiological effects. This generated aggression expressed through threats to contaminate others as a vengeance reaction used to compensate their fear of rejection and death. At the same time, the need for human contact was very strong. There was fear of the future when they would leave FEBEM, fear of never being able to go to school, to have a girlfriend or to marry. Their only thought was to leave the city, and change identity [5].

To HGG (the Goiânia General Hospital, State hospital) went the patients with light to moderate damage to the haematological system and moderate to severe radiodermatitis. As the confinement became more oppressive, anxiety increased. Behavioural disturbances started to appear, in some cases of a severe nature [6]. Some of the patients tried to run away, while others broke everything in sight and were very aggressive. A psychiatrist was called to sedate them [7] and psychologists were called to help. One of them stated that she was afraid of radioactivity, and of the patients, so that it was difficult to help the victims [8].

They were aware of the ostracism within the hospital and that the hospital as a whole was afraid of them. Visits from the merely curious scientific community, who had no connection with the hospital work groups, made them feel like guinea pigs. They were famous and at the same time discriminated against and feared. Treated always by doctors and nurses totally covered and never being able to see their faces or have a human contact, these people felt the intense hurt of segregation [6]. People would go to the front of the HGG and shout "Nuclear Aids, get out of Goiânia" [9]. In any case they were fatalistic about their situation but at the same time sought compensation from the State [4].

To HNMD, Marcilio Dias Naval Hospital in Rio de Janeiro, went the victims with severe radiodermatitis and damage to the haematological system. At first they were impatient with all the protective apparatus and procedures. The fright began with the first deaths: a clear perception of the severity of their situation. As there was the possibility of one patient transmitting an infection to another, anyone with an infection was isolated. Two patients died in isolation and fear of being isolated generated tension and depression. As it was after having a scintiscan that one patient had his forearm amputated, from then on, a scintiscan caused fear of a similar fate. Many had serious lesions on feet and hands: any thought of amputation made them stop eating and sleeping, or if they slept they were subject to nightmares. Their aggressive behaviour increased and they refused medication [10]. The need to see the
faces of the staff created anguish [11]. The chaplain was the only member of the staff to give them psychological support.

3.1. Psychological issues

Two of the patients had been admitted to psychiatric hospitals, but it was impossible to recover the histories of whatever occurrences had caused them to be admitted [7]. These pathological nuclei were triggered during the months of hospitalization. Without their past histories, comparisons could not be made for study but, the fact is that before the accident these people were leading what were considered normal lives within the social contexts [6].

Depression could be a factor which aggravated the patients’ state. An eighteen-year-old admitted to HGG cried when he learned that he was to be transferred to HNMD. He told the nurses that he knew he was going to die, he wished to die in Goiânia and so tried to throw himself out of the window [11]. He arrived at HNMD in a state of extreme depression and died shortly after of opportunistic infection because his immunological system failed to react. This patient had radiodermatitis on his genitalia [10].

3.1.1. Recommendations

When the patient is admitted, immediate psychological intervention should be carried out together with clinical treatment to help the patient deal with the syndrome and to make him understand that he is part of a team whose goal is his recuperation [12]. As depression may contribute to decreasing immunity [13], this challenge could delay the process.

The presence of a chief of the medical group who is in daily and frequent contact with the patient and on whom the patient depends for support may give the feeling of security [14].

The radioprotection procedures will be better accepted if the reasons for them are explained to the patient [6].

The psychological concept of ‘secondary benefits’ plays a major role in the treatment. They must not be put aside, for they are symbolic [15].

Crisis intervention, giving an incentive to the patient to talk about his emotions and to feel that the psychologist understands him, can be useful. Relaxation techniques can also be useful when the pain is extensive. Psychiatric intervention may be needed when suicidal tendencies appear.

The patient whose treatment is not effective, and whose syndrome progresses toward death, confronts all the sentiments inherent in his state, i.e. negation, fear, hate, depression, ambivalence and the search for a meaning to existence. Usually, hospitals have a chaplain and at such a moment, depending on the patient, this intervention can be very beneficial and soothing [14].
It is easier to work with patients who are psychologically more stable than with patients with psychiatric disorders [16].

Group dynamics can be helpful when there are many victims. It diminishes the tension during hospitalization when there is an exchange among the patients about their fears and feelings [14]. Concern about their bodies, the fear of mutilation and of amputation should be worked through [11].

If the patient loses a limb, work must be done toward acceptance of this fact and later, toward readaptation to the body image.

The social worker can be of help in the hospital, working together with the mental health team, for this professional will deal with services of a practical nature [14].

Finally, the mental health team should have a voice in the clinical treatment, so that decisions may be weighed jointly as to when a medical intervention might be more prejudicial than beneficial for the body-mind system of the patient [11]. It may be supposed that depression or anger may diminish the patient's involvement in the battle against the syndrome [13]. The team could also control the visits of curious professionals, thus helping the patient to avoid the humiliation of feeling like a guinea pig [6].

The re-entry into society and adaptation to a new life should have a psychological follow-up to re-establish self-control, self-esteem, and to confront rejection.

The mental health team should be trained to enter into action with confidence. Training should cover basic knowledge of radiation medicine. This team should include the hospital psychologist, psychologists with special training in behavioural therapy, in group dynamics and in family therapy, a social worker and a hospital chaplain.

4. IMPACT ON PHYSICIANS AND HOSPITAL STAFF

"... We had psychiatrists working with us, but we ourselves had no psychological follow-up. I believe that the group could have accomplished more if there had been such a thing, because some of us were at our limits ..." [17].

The victims were admitted on 29 September but their treatment only began in the evening of 30 September, when physicians from Rio de Janeiro arrived in Goiânia: one from Furnas Electric Power Station S/A, one from NUCLEBRÁS (Brazilian Nuclear Companies S/A) and one from CNEN.

The HGG's director had arranged for the evacuation of one ward to receive the victims and there they were left without attention, either because of fear or because of the hospital personnel's lack of preparation.

The physicians from CNEN, NUCLEBRÁS and the radioprotection personnel from CNEN were profoundly shocked when they entered the ward and found twelve patients with severe lesions left without treatment, food or adequate clothing. Later
on the same evening, the physician from FURNAS arriving at HGG discovered, abandoned on a stretcher inside an ambulance, a naked patient presenting severe radiodermatitis, dehydrated and gasping for air. His first task was to take the patient inside the hospital [18].

The conditions in which the patients were found, the problems of the hospital, the fear of the local doctors and the lack of assistance from the nursing staff — all these, added to the feeling of impotence due to their own lack of practical experience, brought the doctors from Rio de Janeiro almost to the breaking point when two hours of sleep per night was average. Only after the passage of weeks did a small part of the medical community respond, nurses volunteered and they united into a small multidisciplinary team.

One physician [19] relates:

"... After the first 48 hours of intense work the tension was so great that I began to present psychosomatic symptoms imitating radiation syndrome: fever, headache, vomiting..."

They point out that there exists a difference between studying the biological effects of radiation and actually encountering radiodermatitis, that in the beginning they were upset and confused, only later on becoming more involved with treatment techniques and thus less emotionally affected [18, 19].

At the HNMD less stress was experienced because the hospital was better prepared to receive the victims and there was international assistance. None the less there was a special emotional involvement with the patients [17, 20].

4.1. The hospital staffs

At the HGG there was a great deal of tension among the few nurses who volunteered. The patients were agitated, rebellious and aggressive, which exhausted the nurses physically. But worst still they were rejected wherever they went, even by doctors of the HGG who called them suicidal and told them that they would contaminate their children, who might get cancer. The personnel of laboratories, laundries, administration and others who had contact with the victims suffered the same fate [21]. It was also difficult to work with so many radioprotection rules and the protection clothes, for it was very hot [22, 23].

At the HNMD there were nurses who had taken specialized courses in radioprotection, but they could not believe their eyes and the dread was intense [9].

4.2. CNEN’s radioprotection team in the hospitals

At the HGG, after two weeks, the team was completely exhausted with overwork and emotional stress. These people were forced, because of the lack of hospital personnel, to assume the role of nurses and also of guards, because some of the patients in extreme agitation rioted inside the ward. At this point the team started
to present behavioural disturbances, insomnia or gastric problems. Later the work was lighter as things started to adjust themselves [21].

The radioprotection team at HNMD say that they were sufficiently prepared and felt no particular impact. None the less, they became deeply involved and found difficulty in accompanying the physical and mental suffering of the patients. Others were affected more strongly and wept. One of them said that he never thought that there would come a day when he would have to screen the body of a victim, a small child [24].

The team that dealt with excreted waste felt themselves under tension and suffered from exhaustion and psychosomatic problems [25].

4.3. Psychological issues

This description of the Goiânia medical community's behaviour clearly shows the fear that radioactivity can produce, even in people with specific scientific knowledge. Trained specialists in radiation tend to experience an emotional impact when facing the real thing. Radioprotection teams are entirely unaccustomed to working in hospitals and dealing with sick people, monitoring bodies or participating in an autopsy.

4.3.1. Recommendations

A lesson learned in Goiânia is that the hospital staff working with irradiated or contaminated victims need psychological support.

Diminishing tension, fear and emotional overload may be achieved through group dynamics or individual crisis intervention. Various items can be dealt with, such as:

— low morale when facing the imminent death of a patient and frustration with the uselessness of all efforts;
— sensation of defeat experienced when a patient dies;
— facing the difficult task of communicating the news of a death to other patients;
— understanding the patients in their fear and learning how to give them emotional support;
— being able to cope with their emotional involvement with the patients and learning how to deal with it.

Informal, multidisciplinary encounters involving the whole staff might result in a better rapport between the teams, insuring that all speak the same 'language' so as to overcome difficulties without friction [14]. The mental health team should be included with the group, for there is also much tension in it.
470

CARVALHO

5. IMPACT ON THE SUPPORT GROUPS

"... There was fear everywhere: fear of sitting or leaning against anything, or taking hold of anything, because nuclear energy is something more or less unknown to the general public ..." [26].

The panic and confusion that took over the city were such that social assistance groups had to be organized in order to give information and support to the population. The CNEN needed people to accompany the teams searching for foci and going inside contaminated areas because the population there was hostile and terrified. Information, when there was any, was contradictory and the teams claimed that it was taking hours to do one hour's work because of the time lost trying to calm the people and giving information. At the same time, the population feared the CNEN workers, with their overalls, masks and strange apparatus. The presence of the social worker helped to mitigate this fear and enabled the teams to approach people directly and create confidence through conversations and explanations [26].

A social worker [26] relates:

"... The people would ask if we were from Goiânia or from Rio de Janeiro. When we said that we were from Goiânia, they asked if we had confidence in the CNEN staff. It was easy, they used to say, for an outsider to feel confidence, because he could leave at any time. It was important for them that we were from Goiânia also ..."

It was almost impossible to find volunteers among social workers, but those who presented themselves participated fully, doing much toward solving the population's problems. A telephone system was organized whereby people might call and advise the authorities of any foci of contamination. A team together with two social workers would confirm the veracity of such a report. The telephone was also used to answer questions from the population.

These social workers felt an intense emotional impact. One of them [27] states:

"... We went into that house with the demolition group to take an inventory of the belongings. I felt so full of pity and upset because I was invading someone's privacy. The house of that woman represented her entire life. Even today I am upset when I remember ..."

One of the first outsiders to offer help to the city was a Civil Defence chief, Colonel Silva, who was at that time in charge of Community Support in the Rio de Janeiro Civil Defence Department. Accompanied by three sergeants from Rio, he arrived at the site of the accident and set up a support system together with other government groups. Accustomed to dealing with catastrophes, he felt no major impact from this one. What impressed him, however, was the violent rejection suffered by the victims [28].
5.1. Psychological issues

The support group is a part of the mental health team and should have the same treatment and training.

6. IMPACT ON THE SPECIALISTS

"... Driving a backhoe tractor, demolishing that house, I cried. I couldn’t see what I was doing because of the tears, but I couldn’t wipe my eyes because the mask over my face was fastened to my cap and overall ..." [29].

Included among the approximately 700 people who worked together during the accident were scientists and technicians from CNEN, NUCLEBRÁS, private companies as well as the Armed Forces and the Civil Defence. While these latter were trained to deal with disasters, the staff from CNEN, whose academic qualifications ranged from technical school certificates to PhD, were used to work with radiation within determined norms and under laboratory conditions. Their training referred to safety standards and accident scenarios appropriate to possible accidents in radiation laboratories and reactor plants. These specialists, whose aptitudes are for the exact sciences, had never had to deal with the social and psychological aspects of this type of emergency [30].

The first staff arriving in Goiânia was appalled by the dimensions of the accident, coupled with the lack of an infrastructure. The first groups worked from 12 to 15 hours in the discomfort of the unfamiliar protective clothing, which caused dehydration because of the temperature, which at times rose to 40°C. Along with the difficulties inherent in their technical work they had to confront the population and the persecution of the press avid for information. The situation was chaotic [30].

Some technicians became so involved with their work that they neglected to take precautions necessary for their own safety. Some showed excessive courage while others felt panic and fled. It was stated that 20 days was the maximum work period that one could take in Goiânia. After that the stress began to generate psychosomatic problems such as acute radiation syndrome and, in some people, pain in the shoulder muscles, which caused rigidity [31].

The impact derived from facing a major and unpredictable accident varied widely because of personality traits. There were changes in behaviour, a few people collapsed and had to give up entirely; some felt faint wearing overalls and masks or fainted, while others said that they felt nothing whatsoever [32].

During the months of October, November and December 1987, January and February 1988, these people had to do everything and anything: screen, decontaminate, drive demolition tractors, use pick and shovel, clean cesspools in hospitals and houses, tear down walls and try, without know-how, to give moral and psychological support to the population [32].
In spite of knowing nothing about psychology, I tried to pass to the people a calmness that I wasn’t feeling myself ... I felt emotionally very small, so horrified by their suffering ..." [32].

The people of Goiânia watched their every move and reaction to judge the seriousness of the situation. The teams had to eat and drink in their houses to prove that there was no danger [33].

Many novel and difficult situations had to be dealt with, as for example the violent rejection of the people of the work groups, which at times resulted in physical aggression and even threats to their lives. This gave rise to improvisation and creativity in dealing with endless problems [1].

6.1. Psychological issues

Examples of the emotions felt by the CNEN personnel:

— feeling of pity for the victims; emotional involvement with them
— sadness and pity at seeing the panic of the population
— surprise at the lack of information about radioactivity
— emotion at being able to help the population, when possible
— sensation of invading the privacy of the victims when entering their contaminated homes
— surprise and pity at seeing the people suffering discrimination
— feeling of impotence in facing appeals for material or emotional help.

6.1.1. Recommendations

The task of the psychologist in working with field personnel is to ease stress, analyse the attitude of any member of the group who deviates from his usual behaviour, diagnose panic situations or those involving negligence in carrying out personal security measures, including the implications of this neglect. It is also to assist team leaders in deciding which members are psychologically equipped to participate in the more dangerous operations and evaluate leadership functions. It is important to apply pre-employment psychological and personality tests to know whether the worker is psychologically suited for the work or how he will react to stress [34]. It was seen in Goiânia that some people were not suited for work during an accident, so psychological tests could be helpful for those who have field work to do.

To cope with stress and emotional crises, individual interviews, crisis intervention or group dynamics could be used.

To analyse situations involving panic or neglect of security measures, a psychologist should follow the personnel during field work. Before undertaking risk
operations, the team leaders should hold discussions with small groups of workers for risk assessment and to debate the best method of carrying out tasks.

Practical field training with simulation should take place with the mental health team.

7. IMPACT ON THE POPULATION OF GOIÂNIA AND BRAZIL

The news that a radiation accident had occurred in the city at first generated only apprehension. However, through the news media and the rapid spread of rumours, sufficient panic was created to mobilize 40 000 people who by the fourth day after the news were already leaving their houses, of their own volition, in the area where the accident had occurred [35].

There were many complaints about the lack of information, especially by the families of those directly affected. During the first week the government initiated the distribution of 100 000 leaflets giving orientation, asking that the isolated areas be avoided and that anyone presenting symptoms of radiation should go at once to the Stadium.

A CNEN physicist [33] relates his experience when, on 1 October, he had to inform the people who lived in the contaminated area, that they must leave their houses:

"... An avalanche of questions began from people awakened suddenly at that hour of the morning. As they began to head for the Stadium, the questions that came flooding were: What happened? What is caesium? What is radioactivity? Is it a contagious sickness? What are its effects on women and children? Does it burn? Is it true that it causes cancer? And if it does, when will we feel it? Is it dangerous where I live? Where can I go and stay? Are the animals and fowls dangerous? Will our lives be shortened? ..."

At first, the higher the person's social stratum the greater was his concern with the consequences of the accident and his awareness of the risks involved [36].

The masses of people streaming toward the Stadium to have their contamination screened were evidence of the bewilderment and fear that overtook the city. Four or five lines formed daily at the CNEN monitoring post and during the subsequent months 112 800 people were screened [3].

A physicist who worked at the Stadium relates:

"... We could see the panic of the population. They believed that this thing called radiation was a new sickness, a sort of plague, contagious, and they took off their clothes and burned them. They believed that radiation could be carried by the wind and infect everyone. Many people monitored complained of vomiting and of diarrhoea, and showed blisters, burns and reddened skin. They insisted that this was something new for them and that these symptoms appeared after the accident. These
people presented no contamination and had never been near the foci. Some people fainted in the lines, out of fear as they approached the moment of monitoring …” [3].

The physicians from Rio de Janeiro, who also attended the outside patients at HGG during October, November and December, observed this same phenomenon and diagnosed the symptoms as herpes [19].

The ostracism practised during the accident was intense. When an accident occurs so suddenly and there is no time to prepare oneself psychologically or reason logically, the instinct of self-preservation dominates. If survival is threatened, people forget about ethics. In Goiânia the situation was still worse because the danger was invisible and was connected with the nuclear/radioactive chain of associations. Most difficult of all, the victims stopped being merely victims to be cared for and became threats to others, from which everyone fled [28].

The people living near the focal points of contamination were as much victims as those who were hospitalized. They were evacuated from their homes, had nowhere to go, were ostracized by friends and family, having no luck in renting places to live or finding a room in hotels. In some cases, they had to sleep in the streets [37]. Even some relatives of the victims were treated with hostility.

Around 600 people tried to stop the funeral of the first two victims in the local cemetery. They were people from the neighbourhood, reinforced by the curious of other districts, who blocked the middle of the street so the hearse could not pass, shouting “Not here!” . The reaction of the angry mob was unprecedented and overstepped all the limits of Brazilian Catholic tradition. The caskets were stoned and in frustration at not having any better weapons, crucifixes were wrenched from nearby tombs as well as pieces of stone from the tombs, bricks or anything else that came to hand [28].

The unaffected population of Goiânia found it necessary to obtain ‘Certificates of Non-Contamination’ issued for people about to travel, people who lived in Goiânia, for school and other places and for the transportation of material from the city to other States [38].

In the city of São Paulo, a car with a Goiânia licence plate was stoned and the same happened in the city of Brasilia [39]. The fear in other states of Brazil was stronger than in Goiânia [40].

7.1. Psychological issues

The blisters, reddened skin, vomiting and diarrhoea observed in 8.3% of the 60 000 people screened at the Stadium could indicate that the fear caused by the accident had temporarily activated neurotic nuclei in persons having such a predisposition [15].

Other manifestations presented in the Stadium included fainting, sweating, palpitations, crises of weeping and acute anxiety, all which could be symptoms of phobia, or fear, exacerbated by the influence of the press and of rumours. The onset
of panic disorders may be related to life events that are “extremely uncontrollable, extremely undesirable or causing extreme loss of self-esteem” [41].

It seems that the human being has a natural tendency to be prejudiced, giving rise to discrimination and this is fuelled by rumour [42, 43].

The whole country was afraid and prejudiced. It must not be forgotten that public anxiety increases with the distance from the accident [40, 44].

7.1.1. Recommendations

Nothing should be hidden from the population, and categorical affirmations should be avoided. Collective interviews should be given and information passed out, always through an official channel. Ambiguity should be avoided [28], for ambiguity is one of the causes of rumours [43].

8. CONCLUSIONS

The following conclusions can be drawn and lessons learned from the Goiânia accident:

— The victims need long term psychological accompaniment from the beginning of hospitalization.
— The hospital staff need psychological support during the treatment of the victims.
— The psychological support must be extended to the personnel when confronting the accident.
— The impact on the public is intense. The benefits and risks of nuclear energy must be explained.
— The mental health teams must be trained to cope with their work without fear.
— One can never be completely prepared for an accident, so improvisation and creativity are necessary.

REFERENCES

[36] MONTEIRO, P., veterinary physician (received the source from one of the victims), personal communication, March 1988.
[40] INSTITUTO VOX POPULI PESQUISA E OPINIÃO, public survey order by the Environment Special Secretariat, Ministry of Urban Affairs and Environment (January 1988).
THE ROLE OF THE UNITED STATES DEPARTMENT OF ENERGY IN RECOVERING FROM THE THREE MILE ISLAND ACCIDENT

D.J. McGOFF, N.P. KLUG
US Department of Energy,
Washington, D.C.,

W.A. FRANZ
EG&G, Idaho Falls,
Idaho

D.E. OWEN
ENCORE Technical Resources, Inc.,
Middletown, Pennsylvania

United States of America

Abstract

THE ROLE OF THE UNITED STATES DEPARTMENT OF ENERGY IN RECOVERING FROM THE THREE MILE ISLAND ACCIDENT.

Even though it resulted in minimal radioactive releases to the environment, the 1979 accident at Three Mile Island Unit 2 (TMI-2) was an event with many profound impacts upon the US nuclear power programme. The ten year involvement of the Department of Energy and its contractors in the TMI-2 cleanup and research programme has yielded many lessons of value for nuclear power programmes around the world. Some of the key lessons learned were: (1) Success at TMI-2 was generally a result of innovative engineering applied to existing technology, rather than due to the development of entirely new approaches. (2) The importance of data acquisition cannot be overstated. Through the course of the recovery, we were often tempted to bypass tedious data acquisition tasks in favour of seemingly more important ‘production line’ work. Time and again, successful data acquisition findings proved to be the turning points which led to key programme successes. (3) TMI-2 revealed that the nuclear industry needed new and improved means of collecting, concentrating, transporting and disposing of radioactive wastes generated by accidents. These improved technologies are now available for use of the nuclear community worldwide. (4) Concern for worker protection during decontamination resulted in a variety of innovations, including new surface cleaning techniques; new radiation survey equipment to quantify contamination levels; improvements in protective clothing; techniques for reducing worker heat stress; and improvements in beta dosimetry. (5) Finally, documentation mechanisms must be established and maintained so that the unique technology developed in cleanup from major nuclear accidents becomes part of the collective body of reactor safety knowledge. The lessons learned from the TMI-2 accident should help ensure that nuclear reactors are designed and operated to minimize the possibility of radioactive releases to the environment, thus assuring the health and safety of the public and operating personnel.
1. BACKGROUND/INTRODUCTION

On 28 March 1979, an accident occurred at Unit 2 of the Three Mile Island (TMI-2) nuclear power station located near Harrisburg, Pennsylvania. The accident, which was initiated while the reactor was essentially at full power, was caused by a highly unlikely combination of events involving equipment malfunction and operator error. Uncovering of the reactor core resulted, causing severe reactor fuel damage, release of much radioactive water and gas to the interior of the plant structures, and damage to the reactor plant instrumentation. With the exception of the noble gases, almost all of the radioactive fission products were retained inside the reactor primary system and the reactor structures. Less than 20 curies\(^1\) of radioactive iodine escaped to the environment out of an estimated core inventory of over 60 million curies. Much has been written concerning the accident's causes and effects, the transient conditions in the plant, and the actions taken to regain plant control. What then followed was a lengthy, technically demanding, and unprecedented cleanup programme.

The accident at Three Mile Island was an event with many profound impacts. The uncertainties during the accident deeply affected local residents, the financial consequences of the accident were significant, and the accident raised public consciousness all around the world that the risks of nuclear power needed to be considered along with its benefits. Along with these impacts, the accident also provided an unexpected but nevertheless important opportunity to learn; to apply the lessons learned towards improving the safety and operation of currently operating and future nuclear reactors; and to improve our preparedness for and ability to respond to any future major nuclear reactor accidents.

The purpose of this paper is to describe the role that the United States Department of Energy (USDOE) and its contractors played during the lengthy cleanup period, including its technology development efforts, and to present some key lessons learned from the USDOE perspective.

2. ORGANIZATIONAL ROLES IN THE CLEANUP EFFORT

During the immediate post-accident period, many organizations, including Federal and State agencies, national laboratories, reactor vendors, electric utilities, universities, and private companies, provided advice, personnel and equipment for the effort. As the magnitude of the reactor core damage and the resulting on-site radiological problem became clear, it was apparent that plant cleanup and recovery would require technical and financial resources much greater than those possessed\(^1\)

\(1\) Ci = \(3.70 \times 10^{10}\) Bq.
by the utility that owned and operated the plant. Much of the cleanup work was of a developmental nature, and was beyond the capabilities of a normal reactor plant operating staff.

Though decontamination and cleanup activities began immediately, most of the first few years were spent in assembling the technical and financial resources for cleanup operations and determining appropriate organizational roles. All agreed that the plant’s owner–operator (GPU Nuclear) retained ultimate responsibility for performing accident recovery. Other US electric utilities and private US nuclear companies, such as Bechtel Corporation, were asked to assist in the cleanup. The research arm of the utility community — the Electric Power Research Institute (EPRI) — was enlisted to co-ordinate dissemination of information to the utility industry.

The uniqueness of the accident recovery and cleanup mandated a strong Government presence at the plant. It became clear that the cleanup would be conducted under intense public scrutiny and that the agency most responsible for assuring safety, the United States Nuclear Regulatory Commission (USNRC), would have an important supervisory role over those actually performing the cleanup. It was also obvious that the accident recovery would provide significant opportunities for reactor safety research and that the Department of Energy and its contractors were best suited fully to exploit these opportunities.

In recognition of these considerations, it was decided to establish a four-party co-ordination agreement among GPU Nuclear, EPRI, USNRC, and USDOE, collectively referred to as GEND. Accordingly, in March 1980, one year after the accident, a basic agreement was signed in which the four parties agreed to co-operate in areas of common research and to disseminate fully the results of the TMI-2 recovery operations to the world.

As its contribution to the effort, the USDOE agreed to help develop the unique technology needed to enable removal and evaluation of the damaged reactor core, and to perform other safety and severe accident research. In 1982, the USDOE role in waste immobilization and reactor evaluation was substantially augmented in recognition of the unique capabilities of the Government for ensuring safe isolation and disposal of radioactive waste materials, and to conduct associated R&D that would be of general benefit. The USDOE also agreed to accept the damaged core, and transport it to its Idaho site for temporary storage and research.

The USDOE selected EG&G Idaho to manage its R&D programme and EG&G has performed a substantial part of the work — much of it at the Idaho National Engineering Laboratory (INEL). In addition, many other USDOE laboratories and contractors — including Oak Ridge National Laboratory, Battelle Northwest Laboratories, Westinghouse Hanford, Sandia National Laboratories and Los Alamos National Laboratory — have participated in the programme and provided valuable assistance.

In 1980, the USDOE established a Technical Integration Office (TIO) at the Island location to provide an on-site presence to carry out the Department’s work
more efficiently. Over the cleanup period, the TIO has served as the USDOE's primary data gathering and distribution group. Its primary tasks were:

(a) Providing technology support to GPU Nuclear for recovery operations
(b) Supporting the core debris shipping programme through on-site preparations and monitoring
(c) Providing samples and other data in support of the accident evaluation effort
(d) Disseminating technical information to the public, industry and scientific community.

For most of the cleanup, the Department maintained a relatively small agency staff and a somewhat larger contractor staff at the Three Mile Island site. These on-site technical personnel had routine access to the plant and were permitted to work directly with the organizations conducting the cleanup. The Department's manager at the Island reported to the Idaho Operations Office, but he had considerable personal budget authority and the latitude to discuss issues directly with USDOE headquarter officials in Washington, D.C. Most of the specialized equipment the Department developed for data acquisition and recovery was developed off the site and thoroughly tested in mock-up facilities before being used at the Island.

The total cost of the USDOE TMI-2 R&D programme has been about $188 million. Direct funding of research and development at the Island was approximately $78 million. Off-site technology support cost an additional $29 million, with the remaining $81 million devoted to research on the accident and its consequences. The total cost of the cleanup of the TMI-2 is expected to be close to $1 billion2.

3. RESEARCH AND DEVELOPMENT TASKS

As a consequence of the substantial research and development programme carried out by the Department and its contractors during the cleanup effort, a number of very significant technological developments and improvements were achieved. Summaries of several key areas where our programme contributed to the advancement of nuclear knowledge and technology follow.

3.1. Waste cleanup and disposal technology

Several improved or new systems and items of equipment were developed by GPU Nuclear and USDOE contractors to clean up the TMI-2 accident and primary system water, and to dispose of the generated wastes.

\[ \text{1 billion} = 10^9. \]
First, the EPICOR II system was developed to process the contaminated water from the auxiliary fuel handling building (AFHB) complex. Consistent with the design objectives of simplicity and utilization of proven technology, the EPICOR II system employed a series of disposable ion exchangers, preloaded with organic and inorganic resin media selected specifically for the physical and radiochemical characteristics of the water being processed.

The EPICOR II system processed AFHB accident generated water during the period from October 1979 to December 1980. This very successful operation established the confidence and experience that was to prove invaluable in dealing with the more highly contaminated water in the containment building basement.

In the containment building, over 600 000 US gallons (2 271 000 litres) of water had flooded the basement to a level of about 8½ feet (2.6 metres). This water was extremely radioactive, having gross fission product contamination (about 90% \(^{137}\text{Cs}\) and \(^{134}\text{Cs}\)) of about 160 µCi/mL. To address this problem, a new system, the submerged demineralizer system (SDS) was conceived in the summer of 1979, and engineered, designed, and installed, on an accelerated basis, over the next two years.

Like EPICOR II, SDS utilized disposable ion exchangers, but in this case they were smaller and loaded with inorganic zeolites, selected because of their high capacity for caesium adsorption and their long term stability under high radiation conditions. The SDS ion exchangers were designed to operate submerged, thus providing water shielding to protect operators from the very high radiation fields that accompany their use.

The SDS performed extremely well. Successful completion of the containment building basement water processing, in 1982, was a major milestone in the recovery programme.

As regards volume reduction, the use of EPICOR II reduced the volume of radioactive waste by a factor of 10, and the SDS reduced the volume by a factor of 500 over conventional waste processing systems.

The USDOE laboratories played a key role in transporting and disposing of the resins produced by the EPICOR II and SDS systems. When gas generation became a problem in the EPICOR liners stored at the Island, EG&G designed and built a device to vent them remotely and add recombiner catalyst. Later a liner was shipped to Battelle Columbus for characterization. Regarding SDS, Battelle Pacific Northwest conducted a successful vitrification experiment on two highly loaded liners.

In September 1984 the USNRC issued new requirements (IE Notice 84-72) dealing with the issue of combustible gas generation in radioactive waste containers. The rule required radioactive waste generators to demonstrate that waste containers do not contain combustible mixtures of gases by means of tests and measurements or by venting the containers before transport. Making means available to comply with this new rule involved both increased cost and radiation exposure. Based on experience with gas generation in the EPICOR II liners, TMI-2 was able to develop a computer assisted method of calculating combustible gas generation rates and, in
turn, predicting safe storage and transport periods before combustible limits were reached. The method uses existing data which were already being collected to comply with transportation and regulatory requirements.

Through USDOE sponsored TMI-2 research, improved techniques are now available to store and dispose of large amounts of radioactive waste, especially radioactive caesium. Operators of US low level waste disposal sites want to reduce the volumes of waste sent to those sites. In response to this need, US utilities have been investigating the use of resins and other absorbent materials that have increased capacity for radioactive waste. This approach decreases the volume of low level waste delivered to the waste sites by significantly increasing the radioactive loading on these materials. The techniques established in the TMI-2 programme for processing and disposing of these highly concentrated radioactive wastes have assisted utilities in developing appropriate techniques for such processing and have also assisted the waste disposal sites in establishing suitable regulations for disposal.

To cite one example of disposal technology, a first-of-a-kind, high integrity container (HIC) was developed and tested by the USDOE, and approved by the State of Washington, for use in disposing of 46 EPICOR II prefilters from TMI-2. Fifty prefilters were transported to the INEL for storage, research and disposition. The disposal demonstration of one EPICOR II prefilter contained in an HIC paved the way for TMI-2 to dispose of 46 prefilters individually contained in HICs at the waste disposal facility operated by US Ecology, Inc., in the State of Washington.

The concrete HIC, developed for disposal of EPICOR II prefilters from TMI-2, is durable, licensed, tested, and is equipped with a one-way vent system for exhausting the gases produced inside. It should be simple to adapt its design and to scale its dimensions according to need. The container can be used for both aboveground storage and underground disposal of low level radioactive wastes, because it is durable, capable of withstanding mechanical deformation, resistant to internal and external corrosion, reinforced internally to withstand high pressures (should the vent system in the lid fail), externally inspectable, and designed to provide some radiation shielding. It is a cost competitive alternative to solidification of Class C low level radioactive wastes.

3.2. Damaged reactor defuelling, shipment and storage

As a result of its TMI-2 involvement, the USDOE and its contractors undertook a major effort to develop new and/or improved technology to help defuel, safely ship and store reactor fuel debris. This included systems, tools, procedures, and operator training for defuelling; canisters and casks for shipping the debris; and equipment, procedures and facilities for safely receiving and storing damaged reactor fuel at INEL.

A key decision in the defuelling approach was to rely on manual defuelling primarily using long-handled tools. The particular concept selected utilized a rotating
shielded defuelling platform above the reactor. Access for direct defuelling by long-handled tools was provided by a slot in the platform. The tools were used to cut, pick up, scoop, or vacuum debris into 12 foot (3.7 metre) canisters. A variety of long-handled tools were used, including relatively simple gripping, scooping and lifting tools as well as more sophisticated tools. After being filled with debris, the canisters were transferred to the fuel storage pool, and subsequently loaded into casks for shipment to Idaho.

One of the main contributions made by the USDOE in the defuelling effort was the development and application of a core boring machine. This machine was initially developed by EG&G to obtain research samples from the damaged TMI-2 reactor core. Using a technique analogous to geological coring, it was designed to sample the complete spectrum of damage structures, ranging from loose fuel debris at the top of the damaged core, to oxidized and damaged fuel assemblies, to previously molten core material, to intact fuel assemblies. The core bore machine proved so successful in drilling into the cores that GPU Nuclear subsequently used the machine for critical steps in the defuelling. The machine was used to pulverize rocklike agglomerations of previously molten fuel material, and to drill through the massive stainless steel plates of the reactor core lower support assembly.

In July 1986, the first seven canisters containing core debris were transported to Idaho. Packaging and transporting radioactive core debris from the Island to INEL were complex operations, involving close co-operation and communication between a large number of parties.

To ensure public safety and confidence, USDOE authorized extra efforts in designing, fabricating, and testing a new rail cask, the NuPac 125-B, for shipping the debris. This cask was certified by the USNRC only after complete review of the application for certification, which included the Safety Analysis Report for the cask, data from drop tests using a one quarter scale cask model, data from drop tests of full-sized fuel and knockout canisters, and resolution of many design- and test-related questions from NRC. Before the application was submitted to the NRC for review and approval, it was subjected to one of the most intense reviews in the history of transporting radioactive materials. Reviewers included personnel from USDOE national laboratories, EG&G Idaho, GPU Nuclear, and several subcontractors. The scrutiny and analysis expended on the application by the USNRC were as thorough as given any application for any rail, or truck, spent fuel cask. From the first meeting with the USNRC on cask certification, until its issuance of the Certificate of Compliance, only about 23 months passed — a record for this type of activity.

The Model 125-B transport casks are 280 inches (7.1 metres) long by 120 inches (3.1 metres) in diameter and weigh approximately 90 long tons (82 tonnes) when fully loaded. Attached to each end of the casks during transport are large energy absorbers called overpacks. The overpacks, made of stainless steel, are filled with foam that would, upon impact, absorb impact energy and protect the cask body. Each cask is mounted on its own heavy duty rail car.
The cask body itself is comprised of two separate vessels to provide double containment. Two separate vessels were needed since USNRC regulations required two barriers and the TMI-2 fuel could not count upon the cladding that surrounds nuclear fuel in an intact assembly. The inner vessel, consisting of a one-inch (2.5 cm) thick stainless steel outer shell, includes a hub and spoke arrangement to support tubes which hold loaded fuel canisters. The outer vessel has a composite wall comprised of three thick layers of metal. The inner shell of the outer vessel is a cylinder of 1 inch (2.5 cm) thick stainless steel, and the outer vessel shell is a cylinder of 2 inch (5.1 cm) thick stainless steel. A 4 inch (10.2 cm) void between the two shells is filled with lead for radiation shielding. A thermal shield surrounds the cask body to further protect it in the event of an accident involving fire.

Seven canisters are loaded into each cask. The canisters are made of stainless steel and measure 150 inches (3.8 m) long by 14 inches (0.36 m) in diameter. Neutron absorber materials are built into the canisters to prevent a chain reaction (criticality) of the nuclear fuel. To prevent the formation of combustible gas mixtures, the canisters are equipped with beds of catalytic materials that recombine hydrogen and oxygen gases back into water.

At the INEL, the casks were unloaded at the EG&G Idaho operated Test Area North (TAN) hot shop, one of the largest remote handling facilities in the world. Samples for examination and analysis were taken from selected canisters. The fuel was then placed in the TAN fuel storage pool for interim storage. The planned national spent fuel repository is the intended ultimate destination for the fuel debris.

The shipping programme has been one of the most ambitious transportation campaigns in the history of the nuclear industry. The campaign has involved transport of nearly 300 000 pounds (136 tonnes) of radioactive core debris from TMI near Harrisburg, Pennsylvania, across nearly 2400 miles (3840 km) of rail track passing through 10 states, to the INEL near Idaho Falls, Idaho. Because of the high visibility of TMI in the aftermath of the Unit 2 accident in March 1979, the transportation campaign received intense public scrutiny.

3.3. Instrumentation and electrical equipment

The accident at TMI-2 provided an opportunity to evaluate a variety of instrumentation and electrical equipment for the effects of exposure to moderately severe accident conditions, including steam, spray, and radiation, as well as hydrogen burn and the resultant overpressure. The examination of this equipment over a period of several years also provided information on long term exposure to moisture. The USDOE endeavoured to recover as much of this information as possible.

Although the primary thrust was the evaluation of the survivability and performance of safety equipment, the programme revealed several weaknesses in the way both safety and balance-of-plant equipment was designed, installed, maintained, and tested. Safety related equipment performed quite well, with the only failures being
pressure transmitters and motor operated valves located in the basement, which were eventually flooded. These findings apply to all nuclear power plants, because the equipment at TMI-2 was not unique and the observed problems can arise in a normal operating plant.

The dominant failure mode of the TMI-2 instrument and electrical equipment was due to corrosion. Water and vapour intrusion into the equipment housings caused erratic readings and ultimate failure. Where a reliable seal existed at the cable entry into the instrumentation and electrical (I&E) equipment housing, the internals were generally not corroded and the instrumentation or electrical equipment was operable.

The installed configuration of some equipment clearly played a major role in its failure under accident conditions. In some cases, these problems resulted in failures during the first 24 hours after the accident. The findings demonstrate that even simple installation requirements, if incorrectly performed, can degrade the ability of the equipment to function as designed. Particular care should be taken with conduit and junction box seals, drains and vents, and the sealing of connector backshells to protect against moisture intrusion. Design practices should take into account those field activities that are critical so that installation problems are minimized. Installation practices should be controlled to ensure they do not degrade the equipment design and application engineering requirements.

To assist in the selection of electrical equipment for removal from the TMI-2 containment, EG&G developed the electrical circuit characterization and diagnostic (ECCAD) system. The ECCAD system was designed to distinguish small differences in the electrical characteristics of different circuits and can distinguish between normal and degraded systems. It is likely that ECCAD-type diagnostic systems will eventually become standard maintenance equipment at most electric utilities, both nuclear and non-nuclear.

3.4. Accident evaluation programme

The TMI-2 accident was unique in two important ways relative to severe accident and source term research. First, the accident occurred in a commercial light water reactor under thermal-hydraulic conditions typical of a large family of hypothetical severe accidents. Second, the damage to the core has been confirmed to be more severe than that characterized by the existing severe fuel damage experiment database. Because of these unique features, review of the accident has significantly increased our understanding of many severe accident and source term technical issues.

Because damage during the accident was primarily limited to the core and reactor vessel, the use of TMI-2 data has been mainly related to those technical issues associated with in-vessel core degradation and fission product behaviour before theoretical vessel failure.
To develop a consistent understanding of the accident, researchers studied samples from the core and reactor systems, data collected during the accident from on-line instruments, and independent severe accident research results. The overall goal of the accident evaluation programme was to provide a complete understanding of the accident scenario, including core damage progression and the physical processes that took place, and to ensure that the results of the programme were effectively transferred to the international nuclear power community.

Accident evaluation results showed that damage to the TMI-2 core was much more extensive than originally estimated. The core reached a peak temperature of approximately 5100°F (2800°C), at least 70% of the core was damaged, between 35-45% of the core melted, and some 20 long tons (18 tonnes) of previously molten materials flowed to the bottom of the reactor vessel. The pressure vessel remained intact because there was still water in the lower head of the reactor vessel. This would indicate that even a small amount of water can serve to cool a severely damaged core.

Research also showed that most of the high volatility fission products were retained in the reactor core, coolant systems and containment building. Of principal concern are iodine and caesium radioisotopes, which would be available for release to the environment in the event of containment building failure. Earlier projections had assumed that most of the high volatility fission products would be released to the environment in the event of severe core damage. Results from the TMI-2 programme indicate that potential radioactive releases from a severe reactor accident may not be as much as earlier postulated.

A major finding has been the demonstration that mechanisms exist which can serve to reduce greatly the release of radioactive fission products to the environment even in the event of major reactor core damage. Research programmes are building on this knowledge to ensure that future reactors are designed and operated to minimize the possibility of release of hazardous fission products.

A good understanding was developed of the condition of the damaged reactor core and plant systems and the specific causes for this condition, from which a much more realistic understanding of plant capabilities and severe accident behaviour has been derived together with the means needed to improve accident prevention, mitigation and control features of reactor design. The databases, and other documentation of this work, represent important elements in ongoing efforts to resolve outstanding severe core damage issues in the licensing of advanced light water reactor designs.

4. DISSEMINATION OF USDOE RESEARCH RESULTS

Recognizing the importance of disseminating the knowledge gained from the TMI-2 research and development programme, the USDOE decided very early in its inception to make the results available to all interested parties throughout the world. A substantial effort was undertaken to meet this goal.
Direct information dissemination has occurred through a number of vehicles including scientific meetings, technical meetings, technical journals, reports and videotapes. More than 100 GEND and EGG-TMI detailed technical reports have been published. Scores of technical reports have been given at meetings of technical societies — in the USA and abroad. Special reports have been given at the USNRC’s annual Water Reactor Safety Research Meeting, and a TMI-2 Executive Conference ‘TMI-2: A Learning Experience’ was held in 1985. Capping this whole effort was the conduct of a topical American Nuclear Society meeting on the TMI-2 accident, which was held in Washington in November 1988. Some 138 papers were presented at this meeting, the majority of which are being published in four issues of the US journal Nuclear Technology. A selected bibliography on the TMI-2 programme, with emphasis on the USDOE’s publications, is included at the end of this paper.

A special effort was made to involve foreign parties in the TMI-2 research programme. A qualified database on TMI-2 was created and made available to researchers around the world. Two international analytical efforts were conducted under the auspices of the OECD: the standard problem exercise and the sample examination programme. The standard problem was a formal exercise in which a number of US and foreign participants applied their calculational and analytical methods to the TMI-2 accident case. The sample examination effort involved distribution of TMI-2 debris samples to a number of laboratories, US and foreign, and a comparison of the results of the examinations. Of special note was the major five year co-operative TMI-2 programme conducted between the US parties and 17 Japanese utility and nuclear industry organizations.

5. LESSONS LEARNED

The ten year involvement of the USDOE and its contractors in the TMI-2 accident cleanup and research programme has yielded numerous lessons of value to nuclear power programmes around the world. This knowledge is applicable to new plant designs, modifications to existing designs, plant operations, emergency preparedness, and accident recovery. An overview of some of the more important lessons is presented below.

5.1. Need for innovative applications of existing technology

The TMI-2 cleanup presented formidable technical challenges, particularly in the areas of facility decontamination, reactor core defuelling and shipment of core debris. In many instances the equipment did not exist to deal with these challenges. Success in these areas at TMI-2 was generally a result of innovative engineering applied to existing technology. Building on existing technology allowed the cleanup to proceed in small, steady, incremental steps. In most instances, this proved to be faster and to cost less than to try to engineer entirely new approaches to the
problems. The approach at TMI-2 was to try to use existing technology in a creative manner. This produced some simple and clever solutions to intimidating problems.

The small step at a time or 'learn as you go' approach was successful because there were so many surprises during the cleanup programme. Looking back, one finds that many of the initial assumptions, calculations and estimates of some aspects of the TMI-2 accident were wrong, even though some of the most knowledgeable and experienced people in the industry were involved. If large scale equipment design and fabrication had been based on some of the early evaluations, there would have been many more wasted steps. Consider the following important areas where initial projections or estimates were subsequently shown to be wrong:

- Extent of fuel fragmentation (more than projected)
- Extent of core melting (more than projected)
- Defuelling (more difficult than projected)
- Worker radiation exposure (less than projected).

If engineering had been based on some of the initial projections, then the resulting equipment would have been either inadequate to the task or unnecessarily complicated and expensive. Engineering based on the incremental approach allowed full advantage to be taken of new knowledge as it became available.

5.2. Importance of data acquisition

The above lesson leads directly to the importance of adequate data acquisition. Numerous specific data acquisition tasks were performed during the course of the cleanup, often at substantial expense and causing a delay in plant cleanup operations. It was often tempting to forego the data acquisition in order to demonstrate more conventional measures of progress (pounds of fuel removed, square feet of surface decontaminated, etc.). But it was these individual data acquisition tasks that time and again shed new light or revealed new information. It was these small but steady increases in knowledge that provided the basis for the 'learn as you go' engineering described above. Consider the following example. It was specialized data acquisition tasks (neutron measurements and video camera inspections) that provided the first evidence that a large quantity of molten fuel had relocated onto the bottom head of the reactor. These tasks indicated this fact over two years before it would have been revealed by the defuelling operation. As a result of this prior knowledge, the planning, tools, and techniques were in place to remove this material effectively once the defuelling operation reached this region of the reactor vessel.

5.3. Need for new waste technology

TMI-2 revealed that the nuclear industry needed new and improved means of collecting, concentrating, transporting and disposing of radioactive wastes generated
by accidents. Liquid waste management proved to be particularly difficult. The large quantity of water, its high initial concentration of fission products, and public concern over its location (on an island, in a river) prompted extensive efforts in this area. As discussed in Sections 3.1 and 3.2, TMI-2 both extended and pioneered the technologies of resin and zeolite ion exchange, handling combustible gases in wastes, techniques for dealing with radioactive wastes, transportation of damaged fuel, and high integrity containers for waste storage and disposal. These technologies are now available to operating plants for their low level waste programmes and to the accident response capabilities of the nuclear industry worldwide.

5.4. Need for improved decontamination technologies

The TMI-2 accident produced contamination levels unprecedented for commercial reactors. The extensive contamination and the concern for worker protection during decontamination activities prompted a variety of innovations, including:

— new surface cleaning techniques
— new radiation survey equipment to quantify contamination levels
— improvements in protective clothing
— techniques for reducing worker heat stress
— improvements in beta dosimetry.

One measure of TMI-2 success in this area is their radiation exposure record. Even though decontamination has been an ongoing effort through virtually the entire decade-long cleanup, personnel radiation exposure is less than half of the initial estimates of the USNRC.

Among the most important lessons of TMI-2 in the decontamination area is in the use of robotics. From their initial steps using simple commercial devices and borrowed DOE robots, TMI-2 has developed unique robots designed for specialized accident recovery tasks, including radiation measurements, video camera surveys, data acquisition, sample acquisition, high pressure water decontamination, concrete core drilling, and sludge vacuuming. Robotics research at TMI-2 has produced innovations in robot design, deployment, and operator training, all of which can be applied to the use of robots in operating power plants.

5.5. Need for technology transfer

The wealth of information stemming from the TMI-2 cleanup and research programmes has been disseminated throughout the world. The USDOE recognized early that among its more important responsibilities was the collection and publication of this unique information. As discussed in Section 4, USDOE, as well as other organizations such as EPRI, employed a variety of technology transfer vehicles. Among the technology transfer innovations at TMI-2 was the decision to combine the techni-
cal reporting of a number of organizations under the GEND report format, providing a single definitive source for research data.

International involvement in TMI-2 research was encouraged and welcomed by the USDOE from the early days of the programme onward. Some twelve countries have been directly involved in some aspect of TMI-2 research, ranging from financial participation, to sample analysis, to computer code analysis, to assignment of their engineers to the project. Numerous other countries have had the benefit of TMI-2 research through their participation in international scientific meetings. As a result of this technology transfer, the nuclear community worldwide has a better understanding of reactor safety issues.

6. CONCLUSION

The accident at Three Mile Island Unit 2 was an event with many profound impacts for the USA. The Federal Government, together with industry, made a concerted effort to learn from the accident to improve the safety of our nuclear power plants. New and improved techniques have been developed to disassemble safely a damaged reactor and dispose of the waste products. Lessons from this accident are also being used to ensure that future nuclear reactors are designed and operated to minimize the possibility of release of hazardous fission products to the environment, thus assuring the health and safety of the public and operating personnel.

SELECTED BIBLIOGRAPHY

EG&G, History and Evaluation of TMI-2 Core Debris Transportation and Storage Campaign, Summary Report prepared for DOE by EG&G, ID (to be published).


TMI-2 Standard Problem — Final Report, prepared for DOE by EG&G Idaho (to be published).
LE PROGRAMME DE RECHERCHE RESSAC*

A. L'HOMME
Département d'analyse de sûreté,
Institut de protection et de sûreté nucléaire,
Commissariat à l'énergie atomique,
Fontenay-aux-Roses

N. PARMENTIER, B. LEGRAND
Département de protection sanitaire,
Institut de protection et de sûreté nucléaire,
Commissariat à l'énergie atomique,
Fontenay-aux-Roses

P. FACHE
Département d'étude et de recherche en sécurité,
Institut de protection et de sûreté nucléaire,
Commissariat à l'énergie atomique,
Saint-Paul-lez-Durance

France

Abstract-Résumé

PRESENTATION OF THE RESSAC RESEARCH PROGRAMME.
If, despite the precautions taken at nuclear facilities, a serious accident should occur in France resulting in substantial releases of radioactivity into the environment, emergency plans exist which will permit rapid decisions concerning the immediate protection of the population: confinement, evacuation, taking of stable iodine, and so on. However, in the medium and long term, action must be taken to decontaminate the affected zones and to limit subsequent contamination of the food-chain so that the population can return to normal life. These actions would concern — if we adopt the terminology used by WHO and the IAEA — in the first instance the near field close to the accident site and then the far field directly affected by fallout, the aim being to reduce external exposure resulting from deposits and internal exposure due to inhalation of radioactive products resuspended in the atmosphere and to ingestion of food products. Under the research and development programmes relating to serious accidents carried out by the Institute for Radiation Protection and Nuclear Safety, the RESSAC programme, set up in 1985, focuses on the rehabilitation methods and equipment required for the near field and on the monitoring of problems characteristic of the far field.

* Réhabilitation des sols et surfaces après accident.
The RESSAC programme is being carried out at the Cadarache Nuclear Research Centre and its main areas of emphasis are: evaluation of the fate of radionuclides deposited on soil and vegetation, determination of priorities and intervention methods and management of the waste generated.

PRESENTATION DU PROGRAMME DE RECHERCHE RESSAC.

Si, en dépit des précautions prises sur les installations nucléaires, un accident grave survenait en France, entraînant des rejets radioactifs importants dans l'environnement, des plans d'intervention permettraient des prises de décisions rapides relatives à la protection immédiate des populations: confinement, évacuation, prise d'iode stable, etc. Mais, par la suite, des actions à moyen et long terme devraient être menées pour décontaminer les zones touchées et limiter la contamination ultérieure de la chaîne alimentaire, afin de permettre le retour des populations à une vie normale. Ces actions concerneraient, par ordre de priorité décroissante et en reprenant les définitions données par l'OMS et l'AIEA, la zone proche du site de l'accident et la zone lointaine subissant l'impact direct des retombées et devraient viser à réduire l'exposition externe due aux dépôts et l'exposition interne par inhalation des produits radioactifs remis en suspension dans l'atmosphère et par ingestion des produits de consommation. Dans le cadre des programmes de recherche et développement menés sur les accidents graves par l'Institut de protection et de sûreté nucléaire a été défini, en 1985, le programme RESSAC, consacré à l'étude des méthodes et moyens nécessaires pour la réhabilitation en zone proche et le contrôle des problèmes concernant la zone lointaine. Le programme RESSAC est en cours de réalisation au Centre d'études nucléaires de Cadarache et ses axes principaux sont les suivants: l'évaluation du devenir des radionucléides déposés sur le sol et la végétation, la détermination des priorités et des moyens d'intervention, et la gestion des déchets générés.

1. INTRODUCTION

Malgré les précautions prises lors de la conception, de la construction et de l'exploitation des réacteurs nucléaires, on ne peut pas exclure totalement l'éventualité d'accidents graves entraînant des rejets significatifs de produits radioactifs dans l'environnement. Il a donc été jugé nécessaire, en France, d'établir des plans de secours destinés à protéger la population des effets de telles situations accidentelles. Concernant la phase accidentelle proprement dite et le court terme après l'accident, ces plans sont actuellement opérationnels: il s'agit du plan d'urgence interne (PUI), relatif à chaque installation nucléaire et dirigé par le chef de centrale, et du plan particulier d'intervention (PPI), relatif à l'environnement proche de chaque site nucléaire et dirigé par le préfet du département concerné. Concernant le plus long terme après l'accident, les pouvoirs publics français élaborent actuellement le plan d'action post-accidentel (PPA), dont les objectifs sont de permettre le retour à une vie normale dans les zones touchées par l'accident par réduction de l'exposition externe due aux dépôts de produits radioactifs et de l'exposition interne due à la remise en suspension des produits radioactifs, et de limiter la contamination ultérieure de la chaîne alimentaire.
L’élaboration du plan d’action post-accidentel nécessite un ensemble de données techniques, notamment sur les méthodes et les moyens à mettre en œuvre pour atteindre les objectifs définis ci-dessus: le but du programme de recherche RESSAC est de fournir de telles données.

La définition de la phase actuelle du programme de recherche RESSAC, qui est en cours de réalisation au Centre d’études nucléaires de Cadarache, a été effectuée en 1985. Certaines de ses composantes sont co-financées par la Commission des Communautés européennes (CCE). Pour cette phase, on a supposé que les dépôts de produits radioactifs résultent d’un accident grave intervenu sur un réacteur à eau sous pression (REP) d’Electricité de France (terme-source de référence S3 des REP [1]). Il est à noter cependant que ceci n’est pas de nature à entamer la généralité de l’application des résultats, car la plupart des nucléides étudiés seraient également relâchés lors d’un accident grave intervenant sur un autre type d’installation nucléaire. Pour cette phase également, les sols retenus pour l’expérimentation sont des sols naturels considérés comme représentatifs de l’environnement proche des sites de REP en France.

Etant donné les objectifs définis plus haut, les principales questions auxquelles le programme RESSAC doit permettre de répondre sont les suivantes:

— l’évaluation du devenir des radionucléides déposés sur le sol et la végétation;
— la détermination des priorités d’intervention, en fonction par exemple de l’occupation des sols et de la vulnérabilité des nappes d’eau souterraine;
— le choix des moyens d’intervention, en fonction du champ d’application (terres cultivées, forêts, voies d’accès, etc.) et de la disponibilité des matériels;
— la gestion des déchets générés.

Nous verrons en détail plus loin quelles actions de recherche ont été entreprises pour apporter les réponses à ces questions. Auparavant, nous allons préciser brièvement le cadre prévu pour l’utilisation des résultats du programme RESSAC, car cela conditionne la nature et la forme désirées pour les résultats.

2. CADRE D’UTILISATION DES RESULTATS DU PROGRAMME RESSAC

En cas de crise nucléaire, l’Institut de protection et de sûreté nucléaire (IPSN) intervient comme appui technique de l’autorité de sûreté pendant la phase accidentelle (domaine d’application du PUI et du PPI) et, plus globalement, de l’ensemble des pouvoirs publics pendant la phase post-accidentelle (domaine d’application du PPA), en assumant en particulier la coordination des moyens d’intervention sur le terrain du Commissariat à l’énergie atomique (CEA). Un centre technique de crise (CTC), implanté sur le Centre d’études nucléaires de Fontenay-aux-Roses, permet de répondre à ces missions.
Pendant la phase accidentelle, l’objectif final du CTC est de fournir une prévision des rejets radioactifs dans l’environnement, suffisamment anticipée pour permettre la bonne application des contre-mesures destinées à protéger les populations [2]. Pour ce faire, le CTC dispose de moyens divers et performants de communication, de documentation, d’évaluation et de prévision. Par exemple, on a installé au CTC un outil rapide et précis d’évaluation et de prévision des conséquences radiologiques — le code CONRAD [3] —, tournant sur un système informatique dédié. Ce dernier permet également de visualiser la répartition des populations autour d’un site, en utilisant les dernières données du recensement national.

Pendant la phase post-accidentelle, le CTC, de par les compétences étendues de l’IPSN, n’a pas seulement un rôle de pure logistique (coordination des moyens d’intervention du CEA sur le terrain mis à la disposition des pouvoirs publics), mais doit largement participer à l’orientation des opérations : à l’instar de ce qu’il effectue pendant la phase accidentelle, il doit fournir des éléments techniques permettant les prises de décisions concernant les actions à effectuer. Aussi le programme RESSAC n’est-il pas conçu seulement comme devant fournir une base technique pour l’élaboration du PPA, mais aussi comme devant aboutir à la création d’outils d’interprétation et d’évaluation (banques de données, logiciels) des actions de réhabilitation des sols.

Par ailleurs, la pratique des exercices de crise, effectués depuis plusieurs années avec les exploitants, les autorités de sûreté et, plus globalement, l’ensemble des pouvoirs publics, montre que les évaluations issues du CTC sont d’autant mieux comprises et mises à profit par les décideurs qu’elles sont fournies sous forme de cartes renseignées. Ceci a conduit au projet d’équiper le CTC d’un système de cartographie (projet CART) ayant pour fonction de visualiser et de superposer des informations sur des fonds de carte, d’effectuer des calculs «simples» sur les données visualisées (tracé de courbes isodoses, calcul du nombre de personnes concernées par une contre-mesure à l’intérieur d’un périmètre donné, calcul du volume de déchets créés par une action de réhabilitation sur une surface donnée, etc.) et de préparer les données pour l’utilisation de codes spécifiques (par exemple, pour l’évaluation de la migration des produits radioactifs dans le sol). Une maquette de ce projet a été établie en 1989 et a été utilisée avec profit lors de deux exercices de crise. Dans un proche avenir, les cartes résultantes seront transmises aux partenaires éloignés du CTC par l’intermédiaire de micro-ordinateurs portables. Il est clair que les outils résultant du programme RESSAC doivent s’intégrer facilement dans l’environnement informatique ainsi défini.

3. DESCRIPTION DU PROGRAMME RESSAC EN COURS

Le programme de recherche en cours comprend trois volets qui sont présentés ci-après.
3.1. Préparation à l'intervention pour la zone proche

Des bases de données sont constituées pour le champ proche autour des sites de REP (rayon de 10 km). Elles s'appuient sur trois éléments:

a) des enquêtes pédologiques visant à recenser les différents types de sol et à déterminer ceux sur lesquels les essais de laboratoire décrits plus loin au paragraphe 3.3 doivent avoir lieu;
b) des cartes d'occupation des sols qui sont élaborées à partir d'enquêtes sur le terrain et qui définissent des zones d'occupation ou d'emploi homogènes des sols. A ces cartes sont associés des fichiers indiquant:

- la destination des sols: surfaces agricoles, surfaces non agricoles (surfaces urbaines, en eau, incultes), surfaces boisées (feuillus ou conifères);
- le cas échéant, le type de culture, en adoptant le classement en usage au niveau de la communauté européenne;
- les superficies concernées.

c) des cartes de vulnérabilité des nappes d'eau souterraine, élaborées en coopération avec le Bureau de recherches géologiques et minières (BRGM). Elles comportent une indication de vulnérabilité par zone, évaluée à partir de divers paramètres, tels que la profondeur de la nappe et la perméabilité de l'aquifère. Y sont également indiqués les différents points de captage et leur usage (alimentation en eau potable, irrigation des cultures, usage industriel).

Comme déjà signalé, ces banques de données caractérisent le champ proche autour des sites de REP. Par ailleurs, elles sont le reflet d'une situation à un moment donné, susceptible de varier dans le temps. Il se pose donc deux types de problème, qui ne sont pas actuellement résolus par le programme RESSAC: la caractérisation du champ lointain et l'actualisation des banques de données au cours du temps. Ces deux problèmes ont été examinés dans le cadre du projet CART évoqué plus haut. La première conclusion est que, pour le champ lointain, on peut se contenter de données moins fines: des fichiers existants, par exemple sur l'occupation des sols, établis et actualisés au niveau national ou européen, peuvent être utilisés valablement. La seconde conclusion est que, en reprenant l'exemple majeur de l'occupation des sols, il n'existe actuellement pas de moyen organisé pour réactualiser les données du champ proche (il n'est évidemment pas question de renouveler tous les ans les enquêtes faites dans le cadre du projet RESSAC). Cependant, une voie de recherche a été identifiée: elle consisterait à exploiter les images du satellite de télédétection SPOT et les logiciels de traitement de ces images mis au point par la société de commercialisation des produits de SPOT. Cette voie de recherche sera explorée dans le futur, probablement dans le cadre du projet CART.
3.2. Essais in situ des techniques et moyens de réhabilitation des sols

Les actions de réhabilitation des sols peuvent aller du retrait de la végétation (cultures, prairies, forêts) au décapage du sol. Elles sont peu dépendantes du type de sol naturel.

Les essais doivent évidemment être effectués sur un terrain d’expérimentation assez étendu et comportant un couvert végétal varié: un champ d’expérimentation de plusieurs hectares, sur les terrains appartenant au Centre d’études nucléaires de Cadarache, est attribué au programme RESSAC et permet de cultiver de nombreux types de végétaux (prairie, produits agricoles). Concernant les zones boisées, un accord a été passé avec le Service national des eaux et forêts. Les dépôts radioactifs sont simulés par émission, au-dessus de la surface étudiée, d’un produit fluorescent non radioactif, sous forme d’aérosols secs de diamètre moyen voisin de 1 \( \mu \text{m} \).

Deux types d’essais sont effectués en parallèle.

3.2.1. Mesure du facteur d’interception des végétaux

Divers types de végétaux, caractéristiques de l’environnement des sites de REP en France, sont cultivés: prairie, céréales, légumes. A différents stades végétatifs, on simule une contamination par la méthode indiquée au début du paragraphe 3.2 et on mesure le facteur d’interception par les végétaux (rapport de la quantité déposée sur les végétaux à la quantité déposée totale, par unité de surface du sol et en notant la masse de végétaux par unité de surface du sol).

Ces résultats permettront d’établir une banque de données chiffrant le gain obtenu par simple enlèvement de la végétation, en fonction du type de végétation, de sa maturité, du rendement par unité de surface.

3.2.2. Qualification des moyens et techniques de fixation et de retrait des dépôts

Pour des raisons pratiques et économiques évidentes, les matériels (machines, produits) à employer pour fixer et retirer les dépôts doivent être, autant que faire se peut, des moyens existants et répandus, couramment utilisés pour d’autres usages. En outre, les adaptations des matériels et les conditions particulières de leur emploi pour les actions de réhabilitation des sols doivent être aussi réduites que possible, dans le souci d’une mise en œuvre rapide.

Différentes techniques de fixation et de retrait de la contamination sont expérimentées, suivant la nature du couvert végétal.

a) Couvert végétal ras

Une première action peut consister à fixer la contamination en surface à l’aide de peintures ou de mousses thermoexpansible pelables. En URSS, on a utilisé des
peintures pelables après l'accident de Tchernobyl et on annonce un facteur d'efficacité de décontamination variant entre une et trois décades. Ces produits ont deux intérêts principaux : d'une part, ils immobilisent les produits radioactifs (pas de migration dans le sol ou d'entraînement sur le sol sous l'effet des précipitations) et, d'autre part, ils permettent, en pénétrant dans la porosité du sol sur une certaine épaisseur (ceci est surtout vrai pour les mousses), de retenir des produits radioactifs qui ont déjà pénétré dans le sol. L'épandage des mousses peut être fait à l'aide d'un matériel agricole couramment utilisé pour l'aspersion des engrais liquides, moyennant quelques adaptations simples.

Une seconde action consiste à décapé le sol, couvert ou non de peinture ou de mousse, sur une certaine épaisseur, à l'aide d'engins lourds usuels de terrassement (scrapers).

b) Couvert végétal plus épais (cultures)

L'effet néfaste à craindre lors du retrait de la végétation est une certaine remise en suspension des produits radioactifs dans l'atmosphère. Le choix des moyens à utiliser doit donc tendre en premier lieu à minimiser cet effet. Par ailleurs, les opérateurs doivent être protégés (port d'un masque respiratoire, aménagement de la cabine de pilotage des engins).

Il semble à priori que les machines agricoles courantes du type ensileuse ou rotovator avec aspiration dans une remorque couverte permettent le retrait de la végétation avec une efficacité satisfaisante et une remise en suspension minimale.

c) Couvert boisé

Pour les arbustes, les buissons et les taillis, on peut employer le matériel courant du Service des eaux et forêts utilisé pour les opérations de débroussaillage destinées à éviter la propagation des feux. Les engins hachent les produits ligneux et aspirent copeaux et feuilles dans des remorques. Pour les arbres, une défoliation non létale peut être pratiquée, suivie d'un ramassage des feuilles par les moyens forestiers classiques.

Les mêmes précautions qu'au paragraphe b) sont à prendre (opérateurs munis de masques respiratoires, cabines aménagées).

De tous les essais évoqués dans le paragraphe 3.2.2 doit résulter une banque de données déterminant, pour chaque cas d'application :
— le type de matériel le plus adapté,
— les procédures d'utilisation, y compris la protection des opérateurs,
— l'efficacité, la durée et le coût de l'intervention.
3.3. Essais techniques en laboratoire sur la migration des produits radioactifs déposés

Les actions de réhabilitation des sols doivent être entreprises dans un certain ordre de priorité dépendant de paramètres économiques, sociaux et techniques. Un paramètre technique important à considérer est la possibilité de migration des produits radioactifs déposés sous l’effet du temps et des conditions météorologiques, à l’intérieur du sol et des végétaux. Le troisième et dernier volet du programme RESSAC actuellement en cours de réalisation a en conséquence pour ambition d’établir une modélisation (données et logiciels) de la migration des produits radioactifs dans les sols et les plantes et de la valider dans des conditions réalistes. Deux types d’essais sont effectués: des essais analytiques destinés à établir la modélisation, des essais globaux destinés à valider les données et logiciels obtenus dans des conditions réalistes. Les nucléides étudiés sont les principaux radioéléments à vie longue (isotopes de Cs, Sr, Ru et Te).

3.3.1. Essais analytiques

a) Etude de la migration dans les sols


b) Etude du transfert racinaire sol-plante

Ces essais, dits «en pots de fleurs», sont réalisés en milieu confiné. Ils peuvent être nombreux, car de faible coût. L’approche rationnelle de l’étude est réalisée de la façon suivante:

— une méthode est mise au point pour déterminer la fraction biodisponible d’un radionucléide contaminant un sol, en fonction des caractéristiques agronomiques du sol, par exemple du taux de présence de cations analogues (le calcium pour le strontium, le potassium pour le césium);
— les facteurs de biodisponibilité sont déterminés pour un radionucléide et une plante donnés, en cultivant la plante sur une solution nutritive où la totalité de la quantité de radionucléide présente dans la solution est considérée comme biodisponible.
La décomposition de l'étude du transfert sol-plante en ces deux étapes diminue considérablement le nombre de combinaisons paramétriques à étudier.

3.3.2. Essais globaux

Comme il a été indiqué au début du paragraphe 3.3, l'un des objectifs de ces essais est la validation de la modélisation établie à partir des résultats des essais analytiques décrits au paragraphe 3.3.1. Un second objectif est de confirmer, pour une contamination radioactive, l'efficacité des méthodes de réhabilitation des sols expérimentées avec un simulant non radioactif dans le deuxième volet du programme défini dans le paragraphe 3.2.

Ceci impose le plus grand réalisme possible à trois niveaux.

a) Les sols

Des blocs de sol non remanié, de grandes dimensions (cube de 2 m de côté), sont prélevés dans l'environnement des sites de REP. La synthèse des enquêtes pédologiques annoncées au paragraphe 3.1.a) permet de limiter le nombre de tels prélèvements à une valeur acceptable, au double point de vue de l'économie et de la représentativité.

b) L'ambiance

Ces blocs de sol sont placés dans des cuves lysimétriques1 permettant la simulation de l'environnement climatique du site (ensoleillement, température, teneur en gaz carbonique, précipitation, hygrométrie) et des mouvements verticaux de la nappe d'eau souterraine (régulation du potentiel hydrique). Différents types de plantes sont cultivés dans ces conditions sur les blocs de sol.

c) La contamination

La contamination des lysimètres est assurée au moyen du générateur POLYR (pollution des lysimètres RESSAC). Ce dispositif permet la production et le dépôt sur les sols d'aérosols radioactifs dans des conditions représentatives de ce qui se passerait lors d'un accident grave survenant sur un REP (formes physico-chimiques, concentration).

1 Un lysimètre est un dispositif expérimental qui permet d'étudier, en simulant des conditions aussi naturelles que possible, les échanges de toute nature entre un prélèvement de sol avec sa couverture végétale et le milieu physique d'origine.
On a donc été conduit, pour abriter ces cuves lysimétriques, à prévoir la construction, sur le Centre d'études nucléaires de Cadarache, d'un bâtiment spécialisé, soumis aux règlements français concernant les installations classées pour l'environnement (ICPE). Ce bâtiment comprendra quatre serres abritant chacune trois lysimètres, le générateur POLYR, divers laboratoires et locaux de service.

4. CONCLUSION

Les résultats de la première phase du programme RESSAC définie dans cet exposé permettront de répondre dans un proche avenir à bon nombre de questions qui pourraient se poser pour la réhabilitation des sols à la suite d'un accident grave intervenu sur un REP. En fait, la réalisation du programme a débuté en 1986 et beaucoup de résultats ont déjà été obtenus et utilisés pour la rédaction du PPA et l'équipement technique du CTC de l'IPSN. Un autre exposé [4], prévu dans le cadre de cette conférence, fait d'ailleurs le point sur les résultats actuellement obtenus et sur le calendrier du reste des essais.

Cette première phase ne permet cependant pas de répondre à toutes les questions qui peuvent se poser.

Le principal problème actuellement non couvert par le programme RESSAC concerne le devenir des déchets générés. En effet, les interventions de réhabilitation seraient génératrices de quantités importantes de déchets différant à la fois par leur degré de contamination et leur nature (terres de décapage, mousses fixatives, végétaux, etc.). Ils nécessiteraient inévitablement la création d'un stockage provisoire. Pour en réduire les coûts et les nuisances, comme par exemple le risque de contamination des eaux de surface ou souterraines, il sera nécessaire de trouver des méthodes permettant de diminuer graduellement le volume de ces déchets par des opérations de décontamination, compactage, etc. Certains phénomènes liés à la nature organique des produits, comme la fermentation, empêcheraient de leur appliquer les méthodes habituelles de conditionnement des déchets nucléaires et de différer longtemps leur traitement. En revanche, certains procédés permettant d'accélérer la décomposition naturelle pour obtenir rapidement un concentrat pourraient se révéler bénéfiques. La réflexion sur la définition des études à effectuer dans ce cadre a bien sûr déjà été engagée et on peut dès à présent penser que le problème du devenir des déchets constituera une première extension logique du programme RESSAC.

D'autres extensions pourraient également être envisagées (accident grave intervenant sur une installation nucléaire autre que REP, contamination d'un milieu urbain). Il nous semble toutefois que le programme RESSAC actuel d'une part (extension «devenir des déchets» comprise), et les acquis de la communauté internationale d'autre part (en particulier concernant la réhabilitation d'un milieu urbain) permettent déjà de répondre en grande partie à ces dernières questions.
Pour terminer, nous signalons que les problèmes afférents à la contamination de l’eau par voie hydrogéologique ou hydrologique ne sont pas oubliés et sont actuellement étudiés dans un autre important programme de recherche de l’IPSN.

REFERENCES


PREMIERS RESULTATS EXPERIMENTAUX DU PROGRAMME RESSAC SUR LES ESSAIS IN SITU DE DECONTAMINATION/FIXATION ET ETUDES DE MIGRATION DES RADIONUCLEIDES DANS LES SOLS

B. LEGRAND, P. FACHE, M. HAMONIAUX, H. CAMUS, D. GAUTHIER
Institut de protection et de sûreté nucléaire,
Commissariat à l'énergie atomique,
Saint-Paul-lez-Durance, France

Abstract–Résumé

FIRST EXPERIMENTAL RESULTS OF THE RESSAC PROGRAMME FROM IN SITU DECONTAMINATION/FIXATION TESTS AND STUDIES OF RADIONUCLIDE MIGRATION IN SOILS.

The paper presents the first experimental results obtained from the RESSAC research and development programme, a general description of which is given in another paper presented at this Symposium. These preliminary results relate to: (1) improved knowledge of plant radionuclide uptake factors derived from in situ tests; (2) a feasibility study on different types of agricultural equipment for operations to remove plant cover or to fix radionuclides on the surface; and (3) analytical studies on the migration of radionuclides in representative soils, permitting the development of easy-to-use calculation software.

1. INTRODUCTION

Dans cette communication sont présentés les premiers résultats expérimentaux menés dans le cadre du programme de recherche et de développement RESSAC, dont la description générale fait l'objet d'une autre communication à ce même colloque. Les résultats préliminaires obtenus portent sur: 1) l'amélioration de l’état de la connaissance des facteurs de captation des radioéléments par les végétaux, au moyen d’essais in situ; 2) des études de faisabilité d’emploi de divers matériels agricoles pour des opérations d’enlèvement du couvert végétal ou de fixation des radioéléments en surface; 3) enfin, des études, dites analytiques, sur la migration des radioéléments dans des sols représentatifs, permettant la mise au point de logiciels de calcul simples d’emploi.

Dans la zone proche du site nucléaire, la réhabilitation des terrains après un accident nucléaire grave serait d’autant plus aisée que la fraction des aérosols
interceptée par les végétaux sera importante: en effet, le simple enlèvement, même partiel, du couvert végétal constitue une première parade pour abaisser les niveaux de contamination. Du point de vue opérationnel, il est donc intéressant de bien connaître les «facteurs de captation» des aérosols radioactifs par les végétaux, spécifiques des catégories de plantes habituellement rencontrées en France autour des sites électrogènes. Il a été convenu que l'étude se porterait prioritairement sur les sites équipés de réacteurs à eau sous pression (REP).

De même, il est intéressant de privilégier les méthodes d'enlèvement du couvert végétal qui font appel à des matériaux courants, aisément disponibles, comme le sont les machines agricoles.

Enfin, si les techniques précédentes n'ont pu être mises en œuvre que tardivement ou se sont révélées insuffisantes, il est nécessaire de prévoir la migration des radioéléments dans les sols les plus représentatifs possible des sites électronucléaires français, et de disposer de méthodes de calcul adaptées (logiciels informatiques).

C'est pourquoi il a été décidé dès la mise en route, en 1985, du programme de recherche et de développement RESSAC (réhabilitation des sols et des surfaces après accident) qu'une partie des investigations porteraient sur:

— l'amélioration de la connaissance des facteurs de captation des radioéléments par les végétaux, par le biais d'essais in situ;
— des études d'emploi de divers matériaux agricoles pour des opérations d'enlèvement du couvert végétal ou de fixation de radioéléments en surface;
— enfin, des études dites analytiques sur la migration des radioéléments dans des sols représentatifs, permettant la mise au point de logiciels de calcul simples d'emploi.

2. ESSAIS IN SITU

2.1. Mesure des facteurs de captation

L'efficacité du processus de captation des aérosols radioactifs par les végétaux dépend de nombreux facteurs tels que le type, la densité, les caractéristiques physiques et physiologiques des végétaux, les propriétés physiques et chimiques des aérosols et les conditions climatiques.

Chamberlain [1, 2] a le premier proposé une relation empirique entre la fraction interceptée $F$ et le rendement cultural $R$, exprimé en kg·m$^{-2}$ de poids sec, de la forme

$$F = 1 - \exp(-m \cdot R)$$

où $m$ représente le coefficient d'interception. D'après ces travaux, réalisés principalement sur des surface en herbe, le coefficient $m$ varie de 2,3 à
3,3 m^2·kg^{-1} sec. Plus récemment, Pinder et al. [3] ont compilé les données de la littérature concernant le facteur de captation et déterminé sa valeur pour d'autres végétaux.

Si la valeur du facteur de captation en fonction de la densité du végétal est relativement bien établie pour certains végétaux tels que l'herbe, il apparaît que les données de la littérature concernant, d'une part, la valeur de ce facteur pour d'autres plantes, tels le tournesol, le colza, etc., et, d'autre part, sa variation en fonction du stade végétatif de la plante, restent peu connues, particulièrement dans le domaine des aérosols microniques.

Des essais systématiques de mesure du facteur de captation des aérosols par les végétaux, pour différentes plantes caractéristiques des sites français et différents stades de croissance, ont été entrepris dans le cadre du programme RESSAC. Ces expériences sont menées en plein champ dans un souci de réalisme.

La simulation de la contamination radioactive est obtenue au moyen d'un aérosol sec fluorescent, non radioactif, dont la granulométrie moyenne est de 1,5 μm et l'écart-type géométrique de 2,2.

L'appareillage d'émission (générateurs ultrasoniques) est déplacé le long des parcelles expérimentales, en amont du vent. La distribution verticale des concentrations est homogène, du moins jusqu'à une hauteur de 2 m.

La mesure des quantités déposées sur le sol (recueillies sur des jauges de dépôt en acier inoxydable) et sur les végétaux est obtenue par fluorométrie des eaux de lavage des échantillons.

Les principaux résultats préliminaires sont présentés dans le tableau I pour plusieurs végétaux et divers stades de croissance: avant la montaison (stade A), à la montaison (stade B), adulte en fructification/épiaison (stade C), et à la maturité commerciale (stade D).

Les plantes à fort développement vertical, comme les céréales (maïs) et le tournesol, conduisent à des facteurs de captation qui peuvent atteindre environ 80% ou plus.

A rendement cultural donné, les valeurs mesurées du facteur de captation pour l'herbe et le maïs semblent montrer un bon accord avec les valeurs calculées au moyen des formules de Chamberlain et de Pinder.

Il faut noter que la première méthode de production des aérosols employée conduisait à des écarts-types importants; ils ont pu être réduits avec la méthode actuelle (voir ci-dessus). La relative dispersion des valeurs du rapport F/R (facteur de captation ramené à une densité culturale unitaire) est consécutive aux variations des rendements des productions agricoles auxquelles sont inévitablement soumis des essais in situ s'échelonnant sur plusieurs années.

Sous réserve de confirmation par un traitement statistique des données plus fouillés, il apparaît que certaines mesures des facteurs de captation présentent une valeur de deux à trois fois supérieure aux valeurs de 20 à 30% souvent considérées en tant que valeurs à retenir par souci de conservatisme [4].
TABLEAU 1. FACTEURS DE CAPTATION (RESULTATS PRELIMINAIRES)

<table>
<thead>
<tr>
<th>Végétal</th>
<th>Stade</th>
<th>Rendement sec (R((g/m^2)))</th>
<th>Facteur de captation du végétal Mesuré (F(%)</th>
<th>Ecart-type</th>
<th>Calculé (F(%))</th>
<th>Facteur normalisé F/R (m²/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Herbe</td>
<td>B</td>
<td>492</td>
<td>73,0</td>
<td>40,5</td>
<td>76(^a)</td>
<td>1,48</td>
</tr>
<tr>
<td>Prairie</td>
<td>D</td>
<td>354</td>
<td>87,5</td>
<td>3,9</td>
<td></td>
<td>2,47</td>
</tr>
<tr>
<td>Blé</td>
<td>C</td>
<td>839</td>
<td>67,0</td>
<td>3,6</td>
<td>0,80</td>
<td></td>
</tr>
<tr>
<td></td>
<td>C</td>
<td>1062</td>
<td>58,3</td>
<td>3,8</td>
<td>0,55</td>
<td></td>
</tr>
<tr>
<td></td>
<td>C</td>
<td>963</td>
<td>76,0</td>
<td>1,7</td>
<td>0,79</td>
<td></td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>1017</td>
<td>78,3</td>
<td>4,2</td>
<td>0,77</td>
<td></td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>1228</td>
<td>66,0</td>
<td>32,0</td>
<td>0,54</td>
<td></td>
</tr>
<tr>
<td>Maïs</td>
<td>B</td>
<td>47</td>
<td>10,5</td>
<td>6,1</td>
<td>15(^b)</td>
<td>2,23</td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>212</td>
<td>47,2</td>
<td>4,6</td>
<td>53(^b)</td>
<td>2,22</td>
</tr>
<tr>
<td></td>
<td>B/C</td>
<td>432</td>
<td>76,0</td>
<td>2,8</td>
<td>79(^b)</td>
<td>1,76</td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>1666</td>
<td>83,8</td>
<td>5,8</td>
<td>99(^b)</td>
<td>0,50</td>
</tr>
<tr>
<td>Tournesol</td>
<td>A</td>
<td>15</td>
<td>21,4</td>
<td>2,5</td>
<td></td>
<td>13,99</td>
</tr>
<tr>
<td></td>
<td>A</td>
<td>40</td>
<td>37,0</td>
<td>7,5</td>
<td>9,25</td>
<td></td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>58</td>
<td>40,5</td>
<td>18,6</td>
<td>6,98</td>
<td></td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>73</td>
<td>63,3</td>
<td>7,0</td>
<td>8,66</td>
<td></td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>109</td>
<td>26,2</td>
<td>9,0</td>
<td>2,40</td>
<td></td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>598</td>
<td>39,0</td>
<td>13,8</td>
<td>0,65</td>
<td></td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>350</td>
<td>62,0</td>
<td>10,0</td>
<td>1,77</td>
<td></td>
</tr>
<tr>
<td>Colza</td>
<td>B</td>
<td>750</td>
<td>82,5</td>
<td>2,0</td>
<td>1,10</td>
<td></td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>1240</td>
<td>80,2</td>
<td>6,3</td>
<td>0,65</td>
<td></td>
</tr>
<tr>
<td>Pommes de terre</td>
<td>A</td>
<td>10</td>
<td>14,9</td>
<td>1,3</td>
<td>14,32</td>
<td></td>
</tr>
<tr>
<td></td>
<td>B</td>
<td>39</td>
<td>43,2</td>
<td>3,8</td>
<td>11,16</td>
<td></td>
</tr>
<tr>
<td></td>
<td>C</td>
<td>205</td>
<td>70,8</td>
<td>7,1</td>
<td>3,45</td>
<td></td>
</tr>
</tbody>
</table>

\(^a\) Selon Chamberlain et al.: F = 100 \times (1 - \text{Exp}(-2,91E-3 \times R)).

\(^b\) Selon Pinder et al.: F = 100 \times (1 - \text{Exp}(-3,60E-3 \times R)).
Les essais de modélisation se heurtent actuellement au choix des paramètres les plus significatifs: ainsi, il a pu être vérifié que la définition agricole du stade végétatif reste un paramètre commode mais imprécis. A l’opposé, la hauteur du végétal (et donc le volume offert et la surface utile) devrait jouer un rôle important dans le cas d’aérosols microniques pour lesquels le dépôt gravitaire reste faible.

2.2. Essais d’enlèvement du couvert végétal

L’ampleur des interventions de décontamination à réaliser autour d’un site nucléaire accidenté conduit à privilégier, autant que faire se peut, l’emploi de matériels sans doute moins efficaces que les moyens spécialisés dans les opérations de décontamination mais plus répandus et, surtout, plus disponibles, comme le sont les matériels agricoles.

Certains engins agricoles sont à prohiber: machines trop spécialisées dans un type de récolte, hors gabarit routier, ou, comme les moissonneuses-batteuses, productrices de poussière.

Après un examen critique du matériel agricole existant mené auprès des organismes spécialisés, la faisabilité de l’enlèvement du couvert végétal au moyen de machines du type ensileuse (à fléaux ou à couteaux) a été confirmée.

Le dispositif expérimental comprenait un train de matériel constitué d’un tracteur, d’une ensileuse et d’une remorque.

L’indice de remise en suspension, défini comme le rapport entre la quantité volumique du polluant dans l’air au niveau du conducteur et le dépôt surfacique, a été mesuré. Les valeurs du taux de remise en suspension sont de l’ordre de $1,5 \times 10^{-3} \cdot m^{-1}$ et diminuent généralement de près d’un ordre de grandeur pour les stades végétatifs finaux. A l’intérieur de la cabine du conducteur, ces valeurs sont de l’ordre de $4 \times 10^{-4} \cdot m^{-1}$, ce qui semble indiquer que l’équipement filtrant standard peut se révéler insuffisant pour assurer la protection des conducteurs vis-à-vis du risque d’inhalation de particules radioactives de l’ordre du micromètre.

Il apparaît que l’ensileuse (à marteaux ou à couteaux) est un appareil adapté à l’enlèvement de certains couverts végétaux. Le facteur limitant son emploi est la teneur en eau de végétaux broyés. Par ailleurs, il est vraisemblable que les plantes à fibres longues, comme le lin, posent d’autres problèmes.

2.3. Essais de fixation par épandage de mousses

L’emploi des peintures formant un film solide qui puisse être détaché des murs ou des sols est d’une pratique courante pour les locaux à risque de contamination. Une méthode voisine, utilisant des mousses de polyuréthane au lieu de peintures sur les surfaces «naturelles» contaminées, tels les sols nus ou à faible couvert végétal, apparaît prometteuse. Son avantage potentiel par rapport aux peintures réside dans
«l’aspiration» et le piégeage des particules radioactives dans la mousse thermodurcisée.


Le bien-fondé de ces méthodes apparaissant établi, il a été décidé d’entreprendre, dans le cadre du programme RESSAC, des essais systématiques. Dans une optique opérationnelle, les contraintes imposées étaient: d’une part, la mise au point d’une méthode qui ne fasse pas appel à un appareillage spécialisé, et, d’autre part, la possibilité de l’utiliser dans une large gamme de températures, en particulier l’hiver.

Les mousses de polyuréthane dites à cellules fermées, couramment employées dans l’industrie du bâtiment pour procéder aux isolations thermiques, ont été retenues pour les essais. Elles sont constituées de deux composants: A (polyol) et B (isocyanate).

Afin d’augmenter la gamme d’utilisation des résines, en particulier à faible température et au moyen d’un matériel agricole courant (asperseur d’engrais liquide), il a été procédé à des tests de fluidité par dilution dans trois solvants usuels: acétone, trichloroéthylène et perchloréthylène. Il en résulte qu’une dilution dans l’acétone à 10% présente le meilleur compromis:
— solvant non chloré et de moindre toxicité;
— temps de durcissement de 10 à 20 min;
— facteur d’expansion du volume variant de 17 à 25, pour des températures variant de 5 à 30°C, soit dans les mêmes gammes que pour les produits purs.

Les essais de terrain, réalisés au moyen d’un ensemble de deux pompes agricoles d’asperision d’engrais liquide, ont permis d’optimiser la qualité de l’épandage après un choix approprié de l’espacement des buses d’aspiration et de la rampe par rapport au sol, la pression de propulsion étant de quelques bars.

Une bonne couverture de sol nécessite 1,6 kg·m⁻² de mélange, valeur qui se situe dans la fourchette des données de la littérature (de 1 à 2 kg·m⁻²). Quelques tests, effectués au moyen d’aérosols fluorescents, mais qui demandent confirmation, laissent espérer une efficacité de décontamination de l’ordre de 90%.

3. ETUDES DE SOLS

3.1. Choix des sols représentatifs des sites français

Les sites électronucléaires français sont nombreux (une vingtaine) et très divers par leurs sols, leur climatologie, leur agriculture. De ce fait, il était nécessaire de réduire le nombre de cas d’études possibles.
Une sélection «de compromis» a permis de définir les cas les plus représentatifs en vue d’études globales de migration des radioéléments dans les sols et de transfert dans les plantes menées dans des lysimètres de grande dimension.

Après analyse des échantillons de sols des sites électronucléaires prélevés pendant les campagnes de prélèvement, onze types de sols ont d’abord été retenus sur la base de critères pédologiques, hydromorphiques et agroclimatiques.

Les résultats de tri statistique par classification hiérarchique ascendante, interprétés à la lumière d’une analyse complémentaire en composantes principales, permettent de distinguer:

— le groupe des sols caractérisés par une forte teneur en argile, en limons fins, en ions échangeables (K, Ca), à faible humidité à saturation, et à pH acide ou à forte humidité à saturation et pH basique;

— le groupe des sols caractérisés par une faible teneur en limons fins, en argile, en ions échangeables, et à teneur variable en fer, aluminium et manganèse, certains sous-groupes ne se différenciant que par leurs valeurs d’humidité à saturation, de teneur en matière organique, et par leur rapport C/N.

En définitive, il est donc possible de séparer les sols riches en éléments fins, tels les limons et argiles, à forte teneur en ions échangeables, des sols plutôt sableux, à faible capacité d’échange. La macroporosité du sol, ainsi que son acidité, ou la valeur des paramètres liés à la matière organique, permettent de réduire à 5 le nombre de sols les plus représentatifs des sites électronucléaires français.

3.2. Etude analytique des sols représentatifs

La migration des radioéléments dans la couche de surface des sols est un phénomène complexe, particulièrement lorsqu’ils ne sont pas saturés en eau (comme cela pourrait être le cas dans une situation réelle de contamination radioactive). Il existe actuellement des modèles de transferts et d’interaction physico-chimique de polluants dans les sols permettant de décrire une large gamme de situations, mais il n’est pas envisageable de les utiliser dans un contexte opérationnel du fait de leur complexité. Il était donc nécessaire de pouvoir disposer d’un outil informatique simple et adapté.

Dans une première étape, il s’est avéré nécessaire de se limiter à une approche locale et unidimensionnelle des phénomènes, et pour cela de travailler en laboratoire sur des colonnes de sols nus et homogènes représentatifs des sites français. Ces essais ont pour but de déterminer, d’une part, les caractéristiques de l’écoulement de l’eau, et, d’autre part, celles des transferts convectifs et dispersifs des traceurs en sols non saturés. Les essais relatifs aux transferts convectifs et dispersifs de produits chimiques ont été limités à la migration d’un traçeur parfait, le calcium.

Les sols qui sont ou seront soumis à analyse concernent les premiers horizons (1 à 20 cm) des profils, comme par exemple: sol brun lessivé à hydromorphie
temporaire, à texture sablo-limoneuse; sol brun jeune, à texture sableuse; sol lessivé à pseudo-gley, à texture argileuse; sol à rendzine grise, etc.

Un modèle de migration a été élaboré: le logiciel CATHY permet la résolution numérique des équations de convection-diffusion, régissant la migration d'un polluant dans un sol saturé ou non en eau.

Un premier module de calcul élaboré avec l’Université de Grenoble (Institut de mécanique) permet de caractériser l'état hydrique du sol [7]. Un deuxième module, élaboré avec l'Université de Nancy (ENSIC, Laboratoire des sciences du génie chimique) calcule le transport du polluant par l'eau en tenant compte des phénomènes habituels de diffusion, de convection et de sorption.

Les données d'entrée comportent un fichier de climatologie du site considéré et un fichier des caractéristiques du sol (caractéristiques hydrauliques, paramètres dispersifs) et de l'interaction sol-polluant (par le biais du coefficient d'échange représentant la capacité du sol à sorber le polluant), qui font l'objet des études analytiques de sols décrites précédemment.

Les résultats obtenus concernent, d'une part, l'hydrodynamique (profils en profondeur de débit, de pression de l'eau, de perméabilité et de teneur en eau), et,
d'autre part, la migration à proprement parler: profils de concentration du polluant dans les différentes phases du sol (phase solide, eau mobile et eau immobile).

A titre d'exemple, la figure 1 présente les profils de migration du césium et du strontium pour un sol de site nucléaire français (Cadarache) trois mois après le dépôt, tels que calculés par le logiciel CATHY pour des conditions climatiques locales moyennes. La nature du sol de Cadarache fait que le césium reste dans les premiers centimètres de sol tandis que le strontium migre beaucoup plus profondément. Le logiciel est en cours de qualification par comparaison avec les profils expérimentaux de migration des radioéléments consécutifs à l'accident de Tchernobyl.

La saisie interactive des données permettant la correction aisée des erreurs, le logiciel CATHY (Turbo-Pascal 4.0, compatible PC) peut être mis en œuvre par un utilisateur néophyte.

4. CONCLUSION

Les résultats préliminaires obtenus dans le cadre du programme RESSAC ont concerné trois domaines.

Il s'agit dans le premier de l'amélioration de la connaissance du facteur de captation des aérosols radioactifs par les végétaux. L'étude systématique des plantes les plus fréquemment rencontrées autour des sites électronucléaires français (et pour une certaine part, européens) permet de disposer d'un ensemble de données homogènes pour plusieurs stades végétatifs et d'envisager à terme une modélisation permettant l'obtention d'abaques simplifiés.

Deuxièmement, du point de vue opérationnel, la faisabilité de l'emploi de matériels agricoles tant pour l'enlèvement du couvert végétal (ensileuse) que pour l'épandage de mousses thermodurcissantes (asperseur d'engrais liquides) a été démontrée. Les adaptations du matériel existant sont mineures et restent acceptables.

Enfin, une méthode de sélection des types de sols des sites électronucléaires français a été entreprise. Sur cette base, on a mené les premières études analytiques des sols représentatifs pour lesquels une modélisation a été développée. Un programme informatique simple d'emploi a été mis au point.

Ces résultats préliminaires demandent à être approfondis.

Concernant les essais in situ, d'autres études portant sur le facteur de captation des espèces forestières ainsi que sur l'emploi d'autres machines agricoles pour l'enlèvement du couvert végétal sont envisagées. Enfin, les études de sols demandent à être complétées par une étude de sensibilité des profils de migration aux différents paramètres afin de définir l'importance des uns par rapport aux autres.
REFERENCES

[6] YURCHENKO, Yu.F., Principal methods and technologies applied in cleaning and decontamination work in the highly contaminated zone round the damaged unit of the Chernobyl nuclear power station, presented at IAEA Technical Committee Meeting on the Overall Operational Planning for the Cleanup and Control of Very Large Areas after a Nuclear Accident (AIEA, Vienne, déc. 1988).
LESSONS LEARNED DURING THE RECOVERY OPERATIONS IN THE CIUDAD JUÁREZ ACCIDENT

G. MOLINA
Instituto Nacional de Investigaciones Nucleares,
Mexico City,
Mexico

Abstract

LESSONS LEARNED DURING THE RECOVERY OPERATIONS IN THE CIUDAD JUÁREZ ACCIDENT.

On or around 6 December 1983 a non-licensed teletheraphy source containing 16.65 TBq (450 Ci) of $^{60}$Co was broken and its 6000 pellets began to be dispersed. The consequences of this accident were the contamination of thousands of tonnes of metallic products that were sold in Mexico and the USA, the contamination of several foundries, streets, and hundreds of houses. Health effects in members of the public were also found. The initial response to the accident was made by the Comisión Nacional de Seguridad Nuclear y Salvaguardias (CNSNS), the Mexican regulatory agency. In this phase of the accident, all major sources of radiation were identified and controlled. During the recovery phase, the CNSNS supervised and directed the technical work. Other governmental agencies (Ministry of Health, Ministry of Ecology and Urban Development, and the Chihuahua State Government) provided personnel and funds for contracting companies which carried out specific activities. Big amounts of contaminated soil and scrap metal were removed from the two most contaminated places: Aceros de Chihuahua foundry, and the scrapyard 'El Fenix'. An intense search covered 17 States of the Mexican Republic, and more than 17 000 buildings and houses were surveyed. 814 constructions were totally or partially demolished by September 1985. Decontamination of streets of two cities: Ciudad Juárez and Chihuahua, was performed because dozens of pellets of cobalt fell from trucks carrying scrap. Other locations required decontamination, also. In the USA the authorities located constructions made with Mexican contaminated rebar, and in a search that covered the 50 States, they recovered 200 tonnes of table bases made in Falcon de Juárez (2). The decontamination of soil including pellets was carried out using containers with 18 cm of concrete shielding. Less contaminated soil was manipulated in drums without shielding and scrap and soil with very low levels of contamination were managed as raw material. The CNSNS recommended as an acceptable limit an exposure rate of 60 $\mu$R/h within 1 m of walls in constructions made with contaminated rebar. Because of the big amounts of wastes generated during the recovery operations, a disposal site was designed and constructed approximately 70 km from Ciudad Juárez, in a place known as 'La Piedrera'. Mexico has had since 1969 a low level disposal site near a town named Maquixco, located 50 km from downtown Mexico City, but it was too far away for transporting the wastes. In 1985 the wastes were disposed at La Piedrera and since then an environmental programme has been started.
1. INITIAL SITUATION

The Mexican authorities were alerted by the Texas Department of Health on 19 January 1984 to the presence of radioactive contamination in metallic products made by Aceros de Chihuahua (ACHISA, a Mexican foundry).

On the same day, personnel of the Comisión Nacional de Seguridad Nuclear y Salvaguardias (CNSNS, nuclear regulatory agency) started the search and control of the radioactively contaminated places and materials. During the first days of the emergency, the main contaminated places were located. Table I shows these places and their estimated activity. This work was directed by Raul Ortiz Magaña. The contamination was caused by $^{60}\text{Co}$ from a non-used and non-licensed teletherapy machine.

2. ABOUT COBALT-60

This radionuclide is artificially produced by activation of $^{59}\text{Co}$ in a nuclear reactor. Cobalt has very similar physical and chemical properties to iron. Cobalt-60 has a half life of 5.26 years and emits a beta particle and two gamma rays of 1.17 and 1.33 MeV. The beta particle has an $E_{\text{max}} = 0.312$ MeV. The gamma rays are very penetrating and the exposure during the accident and during the recovery operations was external. None of the people involved showed internal contamination when they were checked with a whole body counter. At the time of the accident, the teletherapy unit had 16.65 TBq (450 Ci).

3. EL FENIX

The source attached to the internally rotating cylinder was sold to the El Fenix scrapyard. It is believed that the two stainless steel windows were broken with a screwdriver during the pick up. The cylinder, with the punctured source, was weighed in the scale used for small items in the scrapyard. From this location most of the pellets were dispersed, and the cylinder remained near the scale, because it was not attracted by the magnetic crane. The pellets on the other hand were easily attracted by the crane and dispersed in the scrap metal and the soil. The exposure rates found on the day of the arrival of CNSNS personnel at El Fenix are shown in Fig. 1.

On the same day, some effort was made to reduce the radiation levels inside the offices. The scrapyard was closed.

Initially, the pellets located near the fence were removed to allow the movement of machinery and personnel. The pellets were stored in containers made with 18 cm shielding. This work started on 8 February 1984. Outside the city, a field
### TABLE I. ACTIVITY DISTRIBUTION IN CONTAMINATED PLACES, JANUARY 1984

<table>
<thead>
<tr>
<th>Place</th>
<th>Description</th>
<th>Estimated activity</th>
<th>Contaminated materials</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pick up</td>
<td>The rear box of the pick-up used to transport the source was contaminated</td>
<td>2.22 Tq (60 Ci)</td>
<td>Rear box of the pick-up</td>
</tr>
<tr>
<td>Cd Juárez</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>El Fenix</td>
<td>Scapyard where the source was initially sold. This was the main focus of dispersion</td>
<td>5.47 TBq (148 Ci)</td>
<td>Dispersed pellets in the top soil and scrap metal</td>
</tr>
<tr>
<td>Cd Juárez</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Aceros de Chihuahua</td>
<td>Principal customer of El Fenix. This foundry produced construction reinforced bars and billet</td>
<td>0.26 TBq (7 Ci) pellets</td>
<td>Pellets in the scrap, slag and hulls all over the foundry</td>
</tr>
<tr>
<td>Chihuahua City</td>
<td></td>
<td>0.44 TBq (120 Ci) metallic products</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.18 TBq (5 Ci) solids and slag</td>
<td></td>
</tr>
<tr>
<td>Cd Juárez</td>
<td>Streets contaminated with pellets falling from trucks loaded with scrap metal from El Fenix</td>
<td>0.18 TBq (5 Ci)</td>
<td>Soil and paved streets</td>
</tr>
<tr>
<td>Cd Juárez streets</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Falcon de Juárez</td>
<td>Foundry that exported all of its table bases production to USA</td>
<td>51.8 GBq (1.4 Ci) pellets</td>
<td>Table bases, slag, pellets on the floor</td>
</tr>
<tr>
<td>Cd Juárez</td>
<td></td>
<td>0.33 TBq (9 Ci) bases 18 Gbq (0.5 Ci) slag</td>
<td></td>
</tr>
<tr>
<td>Cd Juárez streets</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Chihuahua City streets</td>
<td>Streets contaminated with pellets from the El Fenix trucks</td>
<td>37 GBq (1 Ci)</td>
<td>Soil and paved streets of Chihuahua City and the road to Cd Juárez</td>
</tr>
<tr>
<td>Fundival</td>
<td>Foundry</td>
<td>7.4 GBq (0.2 Ci) pellets</td>
<td>Scrap metal, from El Fenix</td>
</tr>
<tr>
<td>Gomez Palacio, Durango</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Duracero</td>
<td>Reinforced bar producer</td>
<td>0.89 TBq (24 Ci)</td>
<td>Rebar produced with billet bought from El Fenix</td>
</tr>
<tr>
<td>Sans Luis Potosí</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Alumetales Monterrey NL</td>
<td>Aluminium foundry</td>
<td>3.7 MBq (0.000 1 Ci)</td>
<td>Aluminium parts contaminated</td>
</tr>
<tr>
<td>Mexican Buildings and houses</td>
<td>Constructions made with contaminated rebar</td>
<td>2.4 TBq (65 Ci)</td>
<td>Houses and buildings made with contaminated rebar in 17 Mexican States</td>
</tr>
</tbody>
</table>
owned by El Fenix was selected as a temporary storage for the wastes. When the outer part of the scrapyard was free of pellets, then the decontamination started closer to the scales, until the radiation was too high to allow proper work. Then the decontamination of the scales was started. All the work was done by hand, excepting the movement of heavy objects such as containers, big metal parts, temporary shielding, etc.

By 13 April 1984 all the pellets had been removed and placed in drums with shielding in the field outside the city. At this point the radiation hazard to El Fenix's neighbours was considered negligible, and it was decided to stop the work there until a final disposal site was available to receive the wastes in raw form.

The decontamination of slightly contaminated soil and scrap began in January 1985. The levels left were essentially the background.

4. ACEROS DE CHIHUAHUA

On 19 January 1984, the facilities of this foundry were contaminated, and the CNSNS personnel started the identification and control of this area. They found scrap metal contaminated with pellets in trucks loaded in El Fenix; they also identified $^{60}$Co as the contaminant.

FIG. 2. Exposure rates (mR/h) in Achisa at 1 m, 25 January 1984.
The contamination was found in the scrap metal yard, in the furnaces, in metallic products, hulls all over the plant, in the dust retention system, and in the slag dump.

All the reinforced bars recovered before or after use in constructions were gathered in a field close to the plant. The slag and hulls were also gathered close to the plant in raw form. An advantage of this was that it consumed little time and money; the major disadvantage was that it produced huge amounts of wastes. Figure 2 shows the radiation levels measured on 25 January 1984, at 1 m from the floor.

The wastes were transported as raw material to the disposal site selected in La Piedrera, located near to Ciudad Juárez (50 km approximately). The total amounts are presented in Table III.

5. HEALTH EFFECTS

After the initial surveys, it was realized that many people had received doses that could endanger their lives. An urgent programme was carried out to identify those who showed medical symptoms and laboratory indications of high doses of radiation. As a result of this programme, calculations made, and chromosomal aberration dosimetry, the doses estimated were: 7 persons received between 3 and 7 Sv; 73 persons received between 0.25 and 3 Sv; 700 persons received between 5 and 250 mSV. All the people recovered, and a follow-up programme was started by the Health Ministry.

6. DOSES DURING THE RECOVERY OPERATIONS

Table II shows the doses received during the recovery operations.

<table>
<thead>
<tr>
<th>Group</th>
<th>Collective dose (man·Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>CNSNS</td>
<td>0.34</td>
</tr>
<tr>
<td>Health Ministry</td>
<td>0.0015</td>
</tr>
<tr>
<td>Contractors</td>
<td>0.403</td>
</tr>
<tr>
<td>Police</td>
<td>0.0172</td>
</tr>
</tbody>
</table>
TABLE III. VOLUME AND ACTIVITY OF THE WASTES BURIED IN LA PIEDRERA

<table>
<thead>
<tr>
<th>Item</th>
<th>Volume (m$^3$)</th>
<th>Activity (TBq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pick-up</td>
<td>7</td>
<td>2.22</td>
</tr>
<tr>
<td>Teletherapy head</td>
<td>0.5</td>
<td>0.18</td>
</tr>
<tr>
<td>Reinforced bars</td>
<td>1040</td>
<td>4.88</td>
</tr>
<tr>
<td>Non-terminated metal</td>
<td>434</td>
<td>2.73</td>
</tr>
<tr>
<td>Table bases</td>
<td>700</td>
<td>0.33</td>
</tr>
<tr>
<td>Scrap metal</td>
<td>2445</td>
<td>0.074</td>
</tr>
<tr>
<td>Drums$^b$</td>
<td>179</td>
<td>5.84</td>
</tr>
<tr>
<td>Soil and slag</td>
<td>16212</td>
<td>0.37</td>
</tr>
<tr>
<td></td>
<td>21017</td>
<td>16.6</td>
</tr>
</tbody>
</table>

$^a$ January 1984
$^b$ Units.

7. WASTE MANAGEMENT

The pellet-carrying wastes generated during the recovery operations were handled in containers with concrete shielding. Wastes with no pellets were handled in metal drums; only the very low concentration of $^{60}$Co in soils and junk was handled as raw material. This comprised most of the wastes. Mexico has a disposal site near Mexico City, but the distance between Ciudad Juárez and this disposal site is 2000 km. Also, the site was not prepared to receive the huge volume of wastes from the accident. For these reasons, it was decided to select and construct a disposal site. Trenches lined with concrete were constructed to receive the wastes in a very isolated area. Table III shows the amounts buried. After two forced changes the site finally selected was La Piedrera. The place is very isolated and dry. Since then an environmental monitoring programme has started.

8. LESSONS LEARNED

It is better to be conservative. It is better to overestimate the resources needed for recovery than to underestimate them. At the beginning, we estimated less person-
nel, time and money than were actually needed to decontaminate areas and this caused delays, confusion and lack of confidence.

The press, and its effect on public opinion can change a good technical decision into a 'not too good' compromised decision. It is desirable to keep the press well informed with good technical information; perhaps the best way is with press bulletins.

During the accident the need of a mobile laboratory became evident, to make more precise measurements of radioactively contaminated objects, soil, metals, etc. This is better than having to rely on portable instruments.

The co-operation received from the US Government was very valuable, particularly the helicopter surveys made in Chihuahua and Ciudad Juárez and of the road between them.

An intervention level of 0.3 $\mu$Sv/h was chosen for the demolition of houses or buildings. Its application was explained in a guide, and it resulted in the demolition of 814 constructions out of 1276 with contaminated rebar.

Interagency co-operation was needed because of the magnitude of the recovery work. The search for constructions made with contaminated rebar was mainly a task assigned to the Health ministry in 17 Mexican states. CNSNS provided technical expertise, but many tasks were performed with the collaboration of other governmental agencies.

We twice had to change the location of the proposed disposal site. When choosing a site for disposal of radioactive wastes, the potential political problems have to be considered. They can be of greater importance than the technical considerations.

BIBLIOGRAPHY

COMISION NACIONAL DE SEGURIDAD NUCLEAR Y SALVAGUARDIAS, Accidente por contaminación con cobalto-60, Rep. CNSNS-IT-001, CNSNS, Mexico City (1985).

THE TECHNICAL HISTORY OF THE THREE MILE ISLAND UNIT 2 CLEANUP: FACTORS, OPTIONS AND DECISIONS

W.C. HOLTON, C.A. NEGIN
Grove Engineering, Inc.,
Rockville, Maryland

R.W. LAMBERT
Electric Power Research Institute,
Palo Alto, California

United States of America

Abstract

THE TECHNICAL HISTORY OF THE THREE MILE ISLAND UNIT 2 CLEANUP: FACTORS, OPTIONS AND DECISIONS.

The Electric Power Research Institute has sponsored a technical history project to ensure that the logic and consequences of decisions made during the Three Mile Island Unit 2 (TMI-2) cleanup are available for future radiological accident recovery operations. The TMI-2 cleanup is examined in terms of: (1) planning and management; (2) plant stabilization following the accident; (3) decontamination and dose reduction; (4) damaged core removal and shipment; (5) radioactive waste management; (6) personnel protection; and (7) the techniques and importance of gathering data. Many technical decisions reflected issues involving funding, public perception, and the regulatory environment. Each decision involved a choice between several strategies; e.g. manual or robotic techniques to defuel the reactor; demineralization, evaporation, or solidification to process radioactive water; and decontamination or dose reduction to support the other cleanup work. The most important technical influence on decision making was the relevant data available when a decision was made. In many cases, limited data or inaccurate assumptions about conditions were serious handicaps to both planning and operations. A central lesson of the TMI-2 cleanup is the importance of proceeding methodically to understand conditions, to develop a simple engineering approach to handle known conditions, and then to repeat this sequence until recovery operations are complete.

1.  INTRODUCTION

Recovering from the accident at Three Mile Island Unit 2 (TMI-2) involved a series of steps that led from plant stabilization and decontamination to removal of the damaged reactor fuel core and finally to a safe and secure storage condition. These technical steps are well documented; however, the rationale that led to each is not. The decision making logic behind the cleanup can provide important lessons
for recovering from other radiological accidents. Recognizing this, the Electric Power Research Institute has sponsored a technical history project to document the significant decisions, factors, and lessons learned during the TMI-2 cleanup [1]. This project is part of the role that the US nuclear power industry has played in the TMI-2 cleanup, beginning immediately after the accident, when senior technical staff were provided, and continuing with the transfer of expertise and technology.

The history is intended to help managers and engineers to understand the kinds of challenges that existed at TMI-2, the questions that had to be asked, the options considered, and the effectiveness of decisions made. This information is often not documented or is difficult to extract. The lessons learned from the many different types of operations performed during the cleanup will also be of value in recovering from less severe nuclear accidents.

The TMI-2 cleanup is examined in terms of: (1) planning and management; (2) plant stabilization following the accident; (3) decontamination and dose reduction; (4) damaged core removal and shipment; (5) waste management; (6) personnel protection; and (7) the techniques and importance of gathering data. Figure 1 provides an overview of the 11 year cleanup and reflects the timing of dominant recovery operations. Figure 2 is a cutaway view of the TMI-2 containment building, showing the reactor vessel and reactor coolant system.

This paper first addresses several significant planning and management issues and then explores by example three technical areas that were essential to the cleanup: (1) selection of the initial defuelling strategy; (2) management of radioactive waste; and (3) the decontamination strategy to support the cleanup.

2. PLANNING AND MANAGEMENT

The most important element in planning cleanup operations was accurate information about conditions. This was often lacking. Limited data or inaccurate projections regarding conditions were serious handicaps to both recovery planning and operations. A great amount of data was gathered from the first day of the accident in 1979, but the systematic tracking, analysing, and organizing of it did not begin until 1982. As a result, false starts were made and overly conservative or optimistic assumptions were often used. The importance of accurate, timely, and accessible data to the cleanup and to recovery from any radiological accident cannot be overemphasized.

In an important management decision, the utility (GPU Nuclear) chose to fight the potential of bankruptcy following the accident by accepting the primary responsibility to conduct the cleanup itself. The Federal Government provided indirect support through a research and development programme and the nuclear industry provided financial and technical support [2]. Thus, the $1 billion cleanup was conducted within the financial and regulatory constraints of an investor owned utility.
FIG. 1. Timing of dominant cleanup operations.
In addition, the decision to work without compromising safety often added technical difficulties, expense, and time. However, the cleanup was carried out with personnel radiation exposures well below the US Nuclear Regulatory Commission's (USNRC) estimates and with an industrial accident rate better than at many operating plants [3].

Experts from external organizations were used because many aspects of the cleanup were beyond the expertise of a utility company. The US Department of Energy (USDOE) had skills and special facilities that did not exist elsewhere; national laboratories, other utilities, service companies, and universities provided valuable resources. Combining these outside experts with the on-site work force was difficult; however, the combination brought much needed technical support, new ideas, and a channel to the worldwide technical community.

For some time after the accident, GPU Nuclear envisioned returning Unit 2 to operation — although public opposition would have been intense. As the extent of damage to the reactor core and the expense of refurbishment became evident, a decision was made to work without regard to the final disposition of the plant. This decision focused available resources on immediate tasks [4].
No one knew how hard the cleanup would be, how long it would take, what conditions existed inside the reactor vessel, or what tools would work. Schedules and plans were quickly outdated. The only reasonable approach became to establish an overall strategy and then to take steps one at a time, collecting data for the next step, performing that step, and then repeating the sequence. Flexibility often required parallel and sometimes redundant approaches until an effective method was found.

Beyond the technical challenges, GPU Nuclear struggled in a difficult regulatory and public environment. USNRC rules did not exist for post-accident conditions and so the utility had continually to demonstrate to the USNRC that changing plant conditions were safe. In doing so, the customary regulatory requirements for an operating plant were slowly eliminated and replaced by more realistic specifications to reflect the stability of the plant and the progress of the cleanup. Public opposition to plans had to be factored into decisions, especially those related to radioactive waste disposal and the long term storage condition of the facility.

3. MAJOR TECHNICAL DECISIONS

The remainder of this paper discusses aspects of decision making in three major technical areas: defuelling, radioactive waste management, and decontamination. To put these in perspective it is necessary to understand the general technical strategy.

Until the actual extent of the damage was understood, the strategy and technical objectives of the cleanup were often overly optimistic. This is illustrated by a 1979 estimate that called for plant restart in 1985. The initial strategic approach to the cleanup, based on a very limited knowledge of reactor core conditions, was:

1. Decontaminate the plant to near normal levels
2. Disassemble the reactor vessel and remove the fuel and
3. Requalify the plant for commercial operation [5].

As the full extent of reactor core damage was discovered, the strategy evolved to its final form:

1. Stabilize plant conditions and gain access to characterize the containment building and reactor vessel
2. Disassemble and defuel the reactor, with supporting dose reduction and decontamination as necessary and
3. Place the plant in a safe, secure monitored storage condition [6].

3.1. Defuelling

The question of how to remove the damaged reactor core lay at the centre of the cleanup. Not only was the damage severe, it was not well understood. Planners
Coating of previously molten material on bypass region interior surfaces

Hole in baffle plate

Ablated incore instrument guide

Upper grid damage

Cavity

Loose core debris

Crust

Previously molten material

Lower plenum debris

Possible region depleted in uranium

FIG. 4. TMI-2 core end state configuration.
continued to hope that the damage beyond the known regions was not as severe as some estimates would have it [7]. In fact, it was worse. For example, most of the unknown region below the upper debris bed was initially anticipated to consist of intact fuel assemblies. This is depicted in Fig. 3, which shows the known conditions when the initial decision on a defuelling strategy was made in 1984. Figure 4 shows the actual conditions in the reactor vessel, as finally determined in 1988.

The key questions facing planners were: Should defuelling be performed primarily with manual or robotic equipment? Should a cautious approach to removing the core debris be invoked in tool design or could higher production (less conservative) techniques be employed? How much water was required for shielding and how should it be processed? How was the entire system to be integrated from debris removal, to canning, to shipping, to final disposition [8]?

This section will discuss the debate over the first question: whether to use manual or robotic equipment to begin defuelling the estimated 135 000 kg of core debris.

3.1.1. Robotic defuelling system

A robotic system proposed in 1983 would have converted the entire core into vacuumable rubble and transferred it directly into canisters in the fuel handling building. The system had three main elements: a remotely operated service arm (ROSA), a shredder, and a debris vacuum/transfer system. ROSA would take the place of human operators in moving the tools and debris in the vessel. It was a 'human-like', computer controlled, programmable, electro servo powered arm [9].

This approach avoided a great deal of the in-containment work associated with preparing for or conducting defuelling activities. It also avoided the most troublesome problems with the manual approach. Water clarity was less of a problem because ROSA could be programmed to work blind; the volume of water used in shielding could be reduced; problems associated with long-handled tools were eliminated; and material handling problems associated with moving shipping canisters inside the containment building did not exist.

One of the most technically challenging elements of the system was the shredder, which had not yet been developed. A shredder had never been used to grind up intact fuel assemblies or core debris material. Considerable developmental engineering would be required, but the delay could be justified by a projected reduction in actual defuelling time and by the extremely low radiation exposures associated with a completely robotic approach. The primary concerns were:

1. The release of radionuclides into the reactor coolant system, which would raise general radiation levels and challenge the water cleanup system; and
2. Pumping the debris outside the containment without first packaging it — a system failure could have very severe radiological consequences.
Many variations of a manual system were proposed. The manual system was a comparatively slow defuelling concept based loosely on normal defuelling practices. Long-handled tools, a vacuum, and a selection of remotely powered tools could be used to move fuel debris into canisters, which would be transferred to the fuel handling building. A flooded refuelling canal above the vessel would provide shielding but would also require long-handled tools of great length, since the personnel work area would be approximately 15 m above the bottom of the core region.

In the spring of 1984, a modified version of this approach (‘dry defuelling’) was selected over the robotic system. It could be implemented quickly with relatively little development work. In addition, the operators and plant managers were familiar with it since it resembled normal defuelling practices and those used in the US Navy.

The concept differed from earlier manual approaches in several ways:

(1) The fuel transfer canal was only flooded at the deep end where the fuel transfer mechanisms were located. Fortunately, provisions existed for a dam to be installed in the refuelling canal. The water in the fuel transfer canal was thus separated from the reactor vessel, greatly simplifying water cleanup. That separation proved especially important later when problems with water clarity developed.

(2) The workers worked on a rotatable, shielded platform directly above the reactor vessel and approximately 8 m above the bottom of the core. Less water above the core meant that the tools did not have to be so long and would be easier to handle.

(3) The tools were manually operated. Fuel handling could begin using simple, long-handled tools and proceed to more complex methods as necessary.

(4) The average exposure for defuelling workers would be approximately 1 Sv/h.

In addition, the dry defuelling concept was compatible with the ROSA and the other robotic tools, if necessary. The arrangement afforded the greatest flexibility to adapt to changing needs.

From this starting point, the evolving knowledge of the reactor core led to the use of various items of defuelling equipment. These ranged from simple vice grips, scoops, and airlift/vacuum equipment to a modified oil drilling rig and state of the art plasma arc cutting technology [10,11]. In retrospect, the approach of selecting a relatively simple, manual defuelling system and then building on it as conditions warranted seems justified. In particular, it is understandable in view of the then projected 10 month schedule for defuelling. It is too easy to second guess the decision in view of the reality that defuelling required four years and encountered many unanticipated challenges.

The manual system did not always work well. The tools designed for it required extensive modifications because they had been designed with limited
<table>
<thead>
<tr>
<th>Need</th>
<th>Obstacle</th>
<th>Selected</th>
<th>Rejected</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gain control of auxiliary building</td>
<td>Water in auxiliary building tanks, sumps, floor</td>
<td>Mixed bed three-stage ion exchange system (EPICOR II)</td>
<td>Two-stage ion exchange system (EPICOR I)</td>
</tr>
<tr>
<td>• Prepare for containment entry</td>
<td>Highly contaminated water in containment basement</td>
<td>High-capacity/low-flow zeolite demineralizer (submerged demineralizer system - SDS)</td>
<td>• Closed-cycle evaporator</td>
</tr>
<tr>
<td>• Capture curies</td>
<td></td>
<td></td>
<td>• Solidification</td>
</tr>
<tr>
<td>• Ensure safety</td>
<td></td>
<td></td>
<td>• EPICOR II alone</td>
</tr>
<tr>
<td>• Reduce radiation fields</td>
<td></td>
<td></td>
<td>• On-site storage</td>
</tr>
<tr>
<td>Establish acceptable working conditions in containment</td>
<td></td>
<td></td>
<td>• Processing w/converted shipping cask</td>
</tr>
<tr>
<td>• Capture curies</td>
<td></td>
<td></td>
<td>• Fluid bed dryers/ calcinators</td>
</tr>
<tr>
<td>• Ensure safety</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintain and control reactor coolant system</td>
<td>Water in reactor coolant system following accident</td>
<td>High-capacity/low-flow zeolite demineralizer (SDS)</td>
<td>Reconfigure EPICOR II</td>
</tr>
<tr>
<td>• Conduct core examination</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>• Prepare to defuel</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>• Capture curies</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintaining acceptable defuelling conditions</td>
<td>Water in reactor coolant system after initial processing</td>
<td>High-flow zeolite demineralizers and filters (defuelling water cleanup system)</td>
<td>Reconfigure SDS</td>
</tr>
<tr>
<td>• Visibility for defuelling</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>• ALARA for defuellers</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Dispose of processed water</td>
<td>Lancaster Agreement of 1980 prevented discharge of water related to accident; public opposition</td>
<td>Open-cycle evaporation of 2 million US gallons of processed water held in storage</td>
<td>• Solidification</td>
</tr>
<tr>
<td>• Ensure safety</td>
<td></td>
<td></td>
<td>• Discharge to river</td>
</tr>
<tr>
<td>• Prevent TMI-2 from becoming a long-term storage facility</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
knowledge of core conditions. However, the untested robotic system would also have faced numerous challenges for which it was not designed. In addition, its decontamination and repair would have been very difficult. Robotic equipment did perform specific and valuable functions during later TMI-2 defuelling operations, but it also experienced problems during use.

3.2. Radioactive waste management

Throughout the cleanup, the project was forced to manage quantities and types of radioactive wastes without precedent in the nuclear power industry. By far the most demanding waste management challenge was the millions of litres of highly contaminated water, along with the related solid wastes. Where was this water to be stored? How was it to be cleaned? How was it to be disposed of?

Most radioactive trash and solid decontamination wastes were handled in ways similar to those used at other nuclear power plants, but on a much larger scale. Approximately 6000 m$^3$ of radioactive waste (excluding fuel debris) were generated at TMI-2. The vast majority — 98% — could be classified as low level radioactive waste that could be commercially buried; the remaining 2% was disposed of by special arrangements with the USDOE. The control of radioactive gases was only an issue until the containment building was vented of 44 000 curies of krypton gas and the first personnel entry was made in 1980 [12].

The contaminated water — the largest challenge — was initially distributed in the containment basement, the reactor coolant system (RCS), the auxiliary building sumps and tanks, over the lower elevation floor of the auxiliary building, and in the RCS support systems. The water and the associated high radiation fields prevented system maintenance, hindered cleanup work, and posed a potential threat to the environment. The existing plant water processing system lacked the capability to process any of this water, which contained $^{137}\text{Cs}$ concentrations initially ranging from 1 to well over 100 mCi/mL. Table I shows a major water processing and disposal challenges of the cleanup, the options considered, and the cleanup method selected.

In addition to the technical challenges, institutional difficulties greatly complicated radioactive waste disposal. The most significant public intervention was a suit filed by the City of Lancaster to prevent the release of TMI-2 water into the Susquehanna River, even if the water met all regulatory criteria. As a result of an out-of-court agreement signed in early 1980, the discharge of 'accident generated' water into the Susquehanna River was not permitted. Over 8 million litres of processed water eventually fell into this category. Thus, the cleanup work was destined to be waterbound — a consequence that, although not prohibitive, ensured that water management at TMI-2 would require operations managers to constantly juggle volumes of water to support or permit other cleanup activities. The water was stored
in several large tanks on the site until GPU Nuclear decided to pursue plans to evaporate it.

Arranging to dispose of solid radioactive wastes created by the accident or generated during the cleanup required extensive negotiations between GPU Nuclear, the USDOE, the USNRC, and state regulated disposal site operators. This was the result of several factors:

1. Until 1982, the federal regulations affecting shipment and disposal of radioactive wastes were not fully defined regarding waste concentrations and forms. It took several more years for the exact methods for complying with these regulations to evolve.

2. Shortly after the accident, the existing commercial low level radioactive waste burial sites were closed to any TMI-2 accident related wastes because of concerns about the volumes that might be produced. One site reopened in late 1979; another did not accept TMI-2 wastes until 1987.

3. In the case of fuel debris and waste that exceeded commercial burial ground criteria for form or content, extraordinary measures were required by the USDOE and the USNRC. Eventually, the USDOE agreed to take the waste for research and development work and temporary storage to ensure that Three Mile Island did not become a de facto long term waste repository [13,14].

In contrast to institutional obstacles, several unusual technical factors made waste management at TMI-2 considerably less difficult than it might have been. These factors had an important influence on the decision making process and existed because TMI-2 had operated commercially for only three months. Thus, a number of structures that were built to support plant outages had not been used and were not contaminated. For waste management, the most prominent examples were:

1. Two spent fuel pools — After the accident, these large, empty volumes were used to house water storage tanks and a water processing system. Later, the canisters containing core debris were stored in one pool before shipment.

2. Steam generator chemical cleaning building — Likewise unused in 1979, it contained two large tanks and was set up for water processing. It provided an excellent location for a radioactive waste water processing system because it was adjacent to the auxiliary building.

3. Limited quantities of fission and activation products (especially $^{60}$Co) — The presence of large quantities of $^{60}$Co would have greatly complicated many elements in the design of water processing systems and would have produced stronger radiation fields throughout the plant.

In addition to using these structures, many new facilities were built. Several others were planned but never constructed. Figure 5 shows the locations of TMI-2 waste management systems and facilities existing at the time of the accident, proposed but not built, and constructed during the cleanup.
3.3. Decontamination

The decontamination strategy at TMI-2 reflected the changing knowledge of conditions, progress in defuelling, and changing cleanup programme strategies. After the first intensive efforts to gain control of major plant areas, decontamination became a support activity to defuelling, and then an activity aimed at establishing stable and secure conditions for long term storage.

Radiation readings in the plant were very high during the early days of the cleanup. In the auxiliary building, the initial radiation readings ranged from 50 mR/h to 5 R/h, with local hot spots up to 125 R/h in some access areas. In the containment building, radiation readings averaged between 200-450 mR/h on the operating floor levels. Decontamination of these areas was primarily a hands-on cleanup — one that required extreme care, preparation, and training [15]. Robotics were used for specific tasks, but more vital to the success of the operation were conventional decontamination methods employed by a labour force with good morale.

Some of the assumptions that were based on 40 years of decontamination experience became outmoded and new approaches were required. Floor coatings were a major source of recontamination; consequently, surface removal was required [16]. The use of chemicals for decontamination was prevented by concerns about how to process and handle the large volumes of chemical waste that would have been generated.

The overall strategy for decontamination evolved from two competing strategies, both of which had the goal of ensuring a working environment that was safe and conformed to the 'as low as reasonably achievable' (ALARA) concept:

1. Decontaminate completely and then defuel — As with a construction project or plant outage, establish a relatively clean plant to support defuelling and possible restart. This approach included the construction of several large decontamination support facilities.

2. Decontaminate/reduce dose as necessary to support defuelling. This approach assumed that limited time and resources were available and stressed the crucial importance of defuelling. It was more in line with the research and development nature of the cleanup, which required emphasis on understanding and defuelling the reactor core.

The actual decontamination operations fell somewhere in between the two above strategies, with the first approach dominating the early years of the cleanup and the second approach the later years.

4. CONCLUSIONS

The post-accident cleanup of TMI-2 provides a lesson on the importance of timely and accurate data to support recovery planning and operations. Without this
information, either overly conservative or overly optimistic assumptions may greatly add to the difficulty of recovery. As a clearer picture of the post-accident situation is obtained, flexibility and the anticipation of unexpected conditions are essential in order to design appropriate recovery systems and strategies. The cleanup of TMI-2 emphasizes the importance of proceeding methodically to understand conditions, to develop a simple engineering approach to handle known conditions, and then to repeat this sequence until recovery operations are complete.

REFERENCES


[14] SNYDER, B.J., COFFMAN, F.E., Memorandum of Understanding between the USNRC and the USDOE Concerning the Removal and Disposition of Solid Nuclear Wastes from the Cleanup of Three Mile Island Unit 2 Nuclear Plant (Original signed July 15, 1981, revision signed March 15, 1982).
THE NUCLEAR EMERGENCY SERVICE COMPANY
IN THE FEDERAL REPUBLIC OF GERMANY

G. BRUDERMÜLLER, W. NEUMANN
Kerntechnische Hilfsdienst GmbH,
Eggenstein-Leopoldshafen,
Federal Republic of Germany

Abstract

THE NUCLEAR EMERGENCY SERVICE COMPANY IN THE FEDERAL REPUBLIC OF GERMANY.

In September 1977 the Kerntechnische Hilfsdienst GmbH (KHG) was founded as a joint private organization of the Federal German electric utility companies, the companies for nuclear fuel fabrication and recycling and the nuclear research centres to render emergency service in the event of accidents in nuclear facilities in the Federal Republic of Germany by providing equipment and specially trained staff. A permanent company staff of 13 persons have given special training and instruction to 140 employees of other firms who normally work as maintenance contractors in nuclear facilities. During the past eight years the company has provided basic equipment and devices, including a communication centre, radiation measuring laboratories, four wheel drive sampling vehicles, equipment for measuring the incorporation of radioactive material, a container for the maintenance and storage of respiration equipment and protective clothing containers for staff, a container equipped with chemicals and devices for decontamination of rooms, three remote controlled tracked vehicles (cable and radio controlled) which are provided with manipulator arms and television cameras and a radio controlled shovel loader. This equipment is installed either within special vehicles or in 20 ft containers which can easily be transported by lorries or by train. There are in fact only very few accidents in nuclear facilities which require the assistance of KHG. In the past KHG has been called for help on an average only once a year, not by operators of nuclear power plants but rather by hospitals, research centres, universities and the chemical industry. Therefore the emphasis is primarily on staff training and instruction. After eleven years of operation, experience has shown that it makes sense to work with a small crew of permanent staff members and to provide for accidents centrally located special highly sophisticated equipment for a large group of users.

In September 1977 the Kerntechnische Hilfsdienst GmbH (KHG) (Nuclear Emergency Service Co.) was founded as a joint private organization of the Federal German electric utility companies, the companies for nuclear fuel fabrication and recycling and the nuclear research centres to provide emergency service in the event of accidents in nuclear facilities of the Federal Republic of Germany by providing equipment and specially trained staff.
The permanent staff of KHG, 13 persons, have given special training and instruction to 140 employees of other companies who normally work as maintenance contractors in nuclear facilities. This particular structure, i.e. a small permanent crew and a large pool of outside staff, requires an emergency call service. KHG assures an around-the-clock telephone service.

During the past 12 years the company has provided the basic equipment and devices for communication, respiration protection, decontamination, radiation measurement and remote controlled operations. The equipment is installed in special cars or 20 ft containers, which can either be carried on trucks equipped with ISO standard interchangeable platforms or by railroad. The fleet of KHG vehicles (Fig. 1) has been composed in such a way that all the equipment can be transported simultaneously to the operational area.

In the following the main equipment of KHG is specified.

A communication vehicle contains radio sets, a radiotelephone, a telexcopier and a private branch exchange telephone system for communication between KHG and the emergency control centre of the plant operator and for directing the action of KHG. Next to the radio centre working space is provided for small headquarters (Fig. 2). Moreover two 20 ft containers are available for the crew, equipped with a rest room, lounge and cabins for changing clothes, etc.

In order to protect the staff against inhalation of radioactive material, storage of respiration protection gear such as compressed air respirators, recirculation equipment and protective suits is provided inside a 20 ft container. Compressed air cylinders can be filled from an on-board high pressure filling station. The low pressure compressors installed on board supply air for respiration to the protective suits (Fig. 3).

In another container various devices and equipment are provided for decontamination of staff, rooms, and tools.

Two shower containers are available for the decontamination of persons (Fig. 4). Specific decontamination can be carried out by means of a hand washbasin and a shower for the head. Two shower cabinets allow whole body decontamination. The shower container is equipped with an integrated electric water heater. The sewage is discharged into mobile tank containers and delivered to a waste disposal plant.

Two four wheel drive vehicles and four 20 ft containers are available for radiation measurement inside and outside the nuclear installations. The vehicles are equipped with special devices for sampling and sample evaluation in the field. If the background permits, aerosol filters and samples can be evaluated for bulk alpha and beta activity. In addition, samples can be evaluated automatically for specific nuclides in a high purity germanium counting station with an attached multichannel analyser. Besides hand held equipment for contamination and dose rate measurement also gear for sampling airborne dust, iodine, soil, plants and water are carried on board (Fig. 5).
A sample preparation container holds installations for recording, storing and preparing samples (Fig. 6). Sample preparation can be carried out under fume hoods. In most cases this stage involves the preparation of sample aliquots or dilutions for subsequent counting evaluation of the samples in the mobile radiation measurement laboratory.

In the mobile laboratory a variety of samples can be examined for bulk alpha-beta activity. Two very high purity germanium counting stations with multichannel analysers and computers connected allow analyses to be carried automatically for specific nuclides (Fig. 7).

For monitoring the radioactivity of persons, a 20 ft container has been equipped as a body counter with a measuring station for analyses of gamma emitting nuclides (Fig. 8). A NaI detector is mounted above a shielding trough. The counting and evaluation steps are computer controlled. The computer program determines the 50 year dose commitment.

Different types of remote controlled vehicles are available for operations inside and outside the facilities in high radiation areas.

The small remote controlled MF 4 system (Fig. 9) is very flexible and is particularly suitable for operating in unknown territory in the presence of strong contamination and high dose rates. It is used for reconnaissance, radiation measurements and for work using special tools. All control, video and audio signals are transmitted by radio. One battery charge allows the manipulator vehicle to be operated for at least two hours.

The cable controlled MF 3 system (Fig. 10) has been designed for use in radioactively contaminated rooms and areas of maximum local dose rate. It is employed for reconnaissance, radiation measurement, for picking up objects up to a maximum weight of 80 kgf\(^1\) and also for working with a number of tools. The manipulator vehicle is supplied from the 220 V mains by a cable 100 m long. The same cable also transmits all control, video, and audio signals. The vehicle can climb normal staircases and pass through normal inside doors.

For operation in contaminated and irradiated areas, both inside and outside buildings, a heavy duty manipulator vehicle of a total weight of about 3 tonnes is currently being built which can be used for measuring and inspection jobs as well as for clearing and rescue services and for decontamination and assembly work (Fig. 11). The vehicle is to have optional radio or cable control capabilities.

For very heavy operation in contaminated and irradiated environments a remote controlled bucket loader (Fig. 12) is available for removing contaminated soil, collecting radioactive material and building shielding walls or excavating pits. All control and video signals are transmitted by radio. The tracked vehicle can be

Text cont. on p. 550

\(^1\) 1 kgf = 9.807 N.
FIG. 1. Vehicle fleet.

FIG. 2. Radio and lead vehicle.
FIG. 3. Respiration protection container.

FIG. 4. Personnel decontamination container.
FIG. 3. Respiration protection container.

FIG. 4. Personnel decontamination container.
FIG. 5. Radiation measurement vehicle.

FIG. 6. Sample preparation container.
FIG. 7. Radiation measurement container.

FIG. 8. Human body counter container.

FIG. 10. Cable controlled manipulator vehicle MF 3.
FIG. 11. Radio controlled manipulator vehicle MF 2.

FIG. 12. Radio controlled bucket loader.
controlled remotely from a control cabinet in the driver's cab of the transport vehicle. The bucket loader is propelled by a diesel engine acting on hydrostatic drives and can run for about ten hours on a tank filling.

Besides the above mentioned equipment there are a lot of special devices such as mobile filter systems, mobile removal systems for radioactive material, power units and shielding material.

In the past KHG was called for help about only once a year and not by operators of nuclear power plants but rather by hospitals, research centres, universities and the chemical industry. In 1981 a $^{60}$Co source of about $10^{13}$ Bq had to be removed following an accident in an irradiation facility for biological material. The piston of the pneumatic conveyor system had been torn off the source. The room could not be entered because of the high dose rate. After the position of the $^{60}$Co source had been detected by radiation measurement the irradiation tube was remotely cut off by one of our MF-3 manipulator vehicles with a grinding wheel and transferred to a shielded cask placed outside. The working time on the spot was 18 hours and the collective dose received by the five members of the recovery staff 0.52 mSv. There were two further accidents in hospitals. In one case the replacement of a $^{60}$Co source in a 'Gammatron' had been done incorrectly. The $^{60}$Co source with an activity of $10^{14}$ Bq had fallen to the floor. At the time the reloading container for the source carrier was closed and located inside the room. To solve the problem, first the source was shielded using the remote controlled MF-3 vehicle to place lead bricks in position in order to lower the dose rate of about 400 Sv/h inside the room. After that, the reloading container was prepared manually for recovery and then the source was placed in the reloading container by the manipulator vehicle. In 1987 a similar recovery operation was necessary at another hospital because of a malfunction during maintenance of a 'Betatron'. A $^{137}$Cs source of about $10^{14}$ Bq slid out of the device and had to be put into the reloading container, also by using our manipulator vehicle.

After the Chernobyl accident the USSR asked KHG for help with remote controlled vehicles. Three different types of vehicles were sent to Moscow, where Soviet engineers were trained by KHG staff to operate the equipment. After training them, the equipment was sent to Chernobyl to pick up radioactive material, for making measurements inside and outside the plant and for video recording.

Fortunately there are only very few accidents in nuclear facilities which require the assistance of KHG. Nevertheless, the KHG staff is occupied in training and instructing the permanent and the outside staff as well as in procuring and maintaining the very specialized equipment needed, most of which cannot be bought on the normal market but is specially designed by or together with KHG. The training courses KHG staff hold for outside staff, of whom about 140 have been trained so far, cover subjects such as radiation measuring programmes, nuclide specification analysis, operations in highly contaminated areas, the required radiation protection measures and the use of remotely controlled manipulator vehicles. Practical training
during these courses is assured by field and laboratory exercises. Besides training of specialists’ staff, practical experience on the site is of equal importance. For this reason it is very important for KHG to organize together with plant operators practical exercises so as to be able to detect possible weak points in organization and equipment available. These exercises provide practical experience, e.g. on the time required for the specialist staff to arrive on site and how radiotelephone communications can be used effectively.

After twelve years of operation, experience has shown that it makes sense to work with a small crew of permanent staff members and to provide for accidents centrally located special highly sophisticated equipment for a large group of users. In the highly unlikely case of a large scale accident the company crew can draw reinforcements from a large pool of trained service personnel.
THE NEW APPROACH OF THE RADIOLOGICAL EMERGENCY RESPONSE TEAM AT THE BRAZILIAN NATIONAL NUCLEAR ENERGY COMMISSION'S INSTITUTE OF RADIATION PROTECTION AND DOSIMETRY AFTER THE GOIÂNIA ACCIDENT*

C.A. NOGUEIRA DE OLIVEIRA, J.C.A. RÉCIO, J. HUNT, I.A. SACHET
Instituto de Radioproteção e Dosimetria,
Comissão Nacional de Energia Nuclear,
Rio de Janeiro,
Brazil

Abstract

The evaluation of the emergency actions taken during the Goiânia accident caused a complete revision of the Brazilian Nuclear Energy Commission's Institute of Radiation Protection and Dosimetry Emergency Response Team. The changes were in both the scope of the emergency responsibilities and in the organization of the emergency team. This new organization permits an emergency response to accidents in nuclear installations such as nuclear reactors or fuel cycle facilities, or accidents involving radiation sources in hospitals, industry, etc. The organization takes into account all the emergency phases, with emphasis on a quick response in the initial phase. Of a total emergency team of one hundred and four people, there are twenty-six members on call twenty-four hours a day.

1. INTRODUCTION

The Brazilian Government has implemented since September 1988 changes in the strategy to respond to radiological emergencies. These changes were based on the experience gained in the past years, when two major accidents (Chernobyl and Goiânia) involved the Government directly or indirectly. Previously, all the emergency actions were mainly focused on a potential nuclear accident at the Brazilian Nuclear Power Plant — Angra 1 — where the task to be implemented would be in a known environment with defined support and accident types, based on simulated exercises.

* Programme supported by the IAEA Technical Co-operation Project BRA/9/029.
However, now, in addition to that, the strategy is oriented to a prompt and efficient response on a national scale to nuclear or radiological accidents. In line with this policy a revision of the Brazilian Nuclear Energy Commission's Institute of Radiation Protection and Dosimetry (CNEN/IRD) Emergency Response Team (ERT) tasks has been implemented.

Changes in the scope of the emergency responsibilities as well as in the organization of the ERT to execute this new strategy have been made. Previously, the principal activity of CNEN/IRD ERT was the off-site monitoring of any nuclear accident which might occur at Angra 1 NPP. Now, the ERT is also responsible for handling any type of radiological emergency that may occur in Brazil, where the accident type, the environment in which it has occurred and the support requirements may be unknown. To achieve this capability during an emergency, it was necessary to develop or improve the ERT's capacity in the following areas:

- To measure and evaluate levels of radionuclides in the environment;
- To propose countermeasures to reduce the radiation dose to the public;
- To assess the dose received by the public and the emergency workers;
- To perform the triage of people, animals, objects and foodstuffs;
- To support the armed forces, fire brigades, civil defence and other organizations in the isolation of areas, evacuation, specialized treatment of victims and other actions made necessary by the accident.

2. ORGANIZATION AND RESPONSIBILITIES

The CNEN/IRD ERT plan takes into account actions from the initial to the recovery phases, with emphasis on a quick response during the initial phase. To accomplish this, the ERT is organized into two co-ordinated bodies under the emergency management, see Fig. 1.

The CNEN/IRD ERT Management is responsible for setting up all the ERT activities within the scope of the IRD, CNEN institutes and other organizations, and to give all needed support to provide a prompt and efficient team response. The management is directly subordinated to the IRD direction and during an emergency, it has to report, via the Field Operations Co-ordination, to the Executive Group of Emergency Co-ordination — GECE.

Field Operations Co-ordination is activated in an accident situation. Its responsibility is to execute the emergency plan and procedures at the site of the accident. This co-ordination has under its responsibility the following field groups:

*Field Monitoring Group:* Its objective is to perform radiometric measurements to permit evaluation of the situation and selection of countermeasures. It has at its disposal three vehicles equipped with various detection systems connected to computers that permit the quick assessment of radionuclide levels in the environment,
and can estimate the gamma radiation dose rates over a large area in a few hours. This group is composed of professionals in radiation measurements, i.e. area monitoring, gamma spectrometry, etc.

**Radiological Evaluation Group:** Its objectives are to assess the measured data coming from the Field Monitoring Group and to propose countermeasures to the Field Operations Co-ordination. This group is composed of specialists in internal and external dosimetry, environmental control and modelling.

**Occupational Control of Field Operations Group:** Its objective is to ensure the execution of all radiation protection procedures for emergency workers under accident conditions. It is composed of specialists in occupational radiation protection.

**Triage Group:** Its objective is to minimize the spread of radioactive materials to unaffected areas through the monitoring and separation of possibly contaminated people, animals, foodstuffs and objects. It is also responsible for decontamination of contaminated persons and specialized medical treatment. It is composed of professionals in radiological monitoring and radiation hygiene.
The Field Operations Co-ordination with its respective groups comprises 104 members who are CNEN/IRD employees, who have responsibilities in the ERT as well as their routine ones in the institute, and who are mainly called upon to act during the initial phase of an accident. Each of these groups is divided into four modules in such a way that each module is on call twenty-four hours a day for the period of one week. In this way, the four modules cover the period of one month, with each of the twenty-six module members being on call for one week every month in the year.

Two criteria were used in the establishment of these four modules: to have in each module at least one specialist in each area necessary to deal with nuclear or radiological accidents, and to avoid having two specialists in the same area on call at the same time.

The Planning and Preparedness Co-ordination acts on a routine basis and is responsible for assuring that the emergency workers are trained and ready in advance of an emergency situation; that the emergency plan and procedures have been written, agreed and tested, and that there is an adequate infrastructure including transport, communication, radiological and medical equipment, available for emergency use at all times.

To accomplish this, it is divided into three sections: logistic support, emergency procedures and training. It has currently a permanent staff of ten professionals and can count on the IRD specialists as consultants for specific technical subjects. This co-ordination is also responsible for training external groups such as fire brigades, civil defence, the armed forces, etc., directly involved in any nuclear or radiological emergency.

Figure 1 shows the CNEN/IRD ERT organization during normal and accident situations.

3. FINAL REMARKS

This new organization permits the response to accidents at nuclear installations such as power or research reactors or fuel cycle facilities, or accidents involving other radiation sources.

All the CNEN/IRD Emergency Response Team personnel have to fulfil basic requirements such as physical and mental fitness; participation in the elaboration of group emergency procedures; training in first aid, communications and nuclear instrumentation, etc., before being considered members. The team members participate in simulation exercises to obtain a better interaction, to test the emergency procedures and to obtain a better interface with the public, the press, civil defence, fire brigade and other bodies. These training procedures make it possible for the ERT to absorb personnel from other organizations, which permits an increase in the
number of module members, within the response strategy established by the ERT management.

To deal with all the phases of a large scale accident, the CNEN/IRD ERT will work together with the Brazilian armed forces, fire brigades, civil defence and other organizations. Training and simulation involving co-operation between the CNEN/IRD ERT and these organizations have already begun, and the lessons learned from these exercises have been incorporated in the ERT planning and procedures.

This new structure will also allow the CNEN/IRD Emergency Response Team to respond to an emergency in a foreign country at the IAEA's request.
INTERVENTION TELECOMMANDEE
EN CAS D'ACCIDENT NUCLEAIRE

J. COUTURE
Compagnie générale des matières nucléaires (COGEMA),
Vélizy

B. NOC
Service de la production thermique,
Electricité de France,
Paris – La Défense

A. KHAIRALLAH
Groupe Intra,
Fontenay-aux-Roses

P. VESSERON
Institut de protection et de sûreté nucléaire,
Commissariat à l'énergie atomique,
Fontenay-aux-Roses

France

Abstract–Résumé

INTERVENTION BY REMOTE CONTROL IN THE EVENT OF A NUCLEAR ACCIDENT.

Discussions in France following the accidents at Three Mile Island and Chernobyl led Electricité de France, the Compagnie générale des matières nucléaires and the Commissariat à l'énergie atomique to establish a joint intervention organization called 'Groupe Intra'. The objective of this organization is to set up, maintain and when necessary use a common stock of remote controlled intervention equipment to deal with serious accidents at any facility belonging to one of the Group's members, and also to offer services to outside French or foreign partners. The Group's equipment is designed to be used essentially in the post-accident evaluation stage and in subsequent phases involving the preparation and implementation of operations to restore normal conditions. One important function is to survey the accident site and the surrounding territory, making measurements that yield information on the situation in general and the radiological situation in particular. These activities involve the use of helicopter-borne radiological mapping equipment and remote controlled vehicles which can inspect both the inside and outside of site buildings. Another function is to contain the contamination of the site and to prepare for possible human intervention. These activities involve the use of civil engineering machines fitted with radiation protection gear or remote controlled equipment. The first generation of this stock of equipment will be operational by the middle of 1990. The second generation, incorporating recent technological developments, should be operational by 1993.
INTERVENTION TELECOMMANDEE EN CAS D'ACCIDENT NUCLEAIRE.

Les réflexions menées en France à la suite des accidents de Three Mile Island et de Tchernobyl ont conduit Electricité de France, la Compagnie générale des matières nucléaires et le Commissariat à l'énergie atomique à créer une structure commune d'intervention, dénommée Groupe Intra. Cette structure a pour mission de réaliser, maintenir et mettre en œuvre un parc commun de moyens téléopérés d'intervention en cas d'accident grave sur une installation de l'un des membres et peut offrir des services à des partenaires extérieurs, français ou étrangers. Les missions retenues pour le parc de moyens du Groupe Intra se situent essentiellement dans la phase d'évaluation de la situation après un accident et dans les phases ultérieures de préparation et de suivi des opérations de rétablissement de la situation normale. Une première catégorie de missions a pour objet la reconnaissance et l'acquisition de mesures, en vue de renseigner sur la situation, notamment radiologique, du site accidenté et de son environnement. Ces missions utilisent un équipement héliporté de cartographie radiologique et des véhicules télécommandés pouvant inspecter soit l'intérieur, soit l'extérieur des bâtiments du site. Une autre catégorie de missions a pour but de contenir la contamination du site et de préparer d'éventuelles interventions humaines. Elles utilisent des engins de génie civil auxquels sont adaptés des équipements de radioprotection ou de télécommande. Une première génération de ce parc sera opérationnelle dès la mi-1990. Une deuxième génération, intégrant les progrès récents de la technique, devrait être, avec sa logistique, opérationnelle dès 1993.

1. INTRODUCTION

Les réflexions menées en France, aussi bien au Commissariat à l'énergie atomique qu'à Electricité de France, dès l'accident de Three Mile Island et surtout depuis celui de Tchernobyl, ont conduit les deux organismes à proposer aux pouvoirs publics d'engager une série d'actions ayant pour objectif de renforcer leur capacité de réponse en cas d'accident grave sur une installation nucléaire française.

C'est dans ce cadre qu'Electricité de France, la Compagnie générale des matières nucléaires et le Commissariat à l'énergie atomique ont décidé de créer, sous la forme d'un groupement d'intérêt économique, une structure commune d'intervention qu'ils pourraient mobiliser aisément, dénommée Groupe Intra.

Créé en juin 1988 pour une durée initiale de dix ans, le Groupe Intra a pour mission principale de réaliser, puis de maintenir et de mettre en œuvre un parc de matériels d'intervention, pour la plupart téléopérés, utilisables en cas d'accident grave sur une installation nucléaire de l'un des membres du groupe. Les statuts du Groupe Intra prévoient, en outre, qu'il peut offrir des services à ses membres ou à des partenaires extérieurs, français ou étrangers.

2. LES MISSIONS RETENUES POUR LE PARC D'INTERVENTION

Globalement, ces missions peuvent être classées en deux grandes catégories:
2.1. Missions de reconnaissance

Les missions de reconnaissance et d’acquisition de mesures ont pour but de renseigner sur l’état des lieux, en particulier sur la situation radiologique du site accidenté et de son environnement, et d’aider à l’évaluation des nécessités et des possibilités d’intervention en vue de limiter les conséquences de l’accident et de permettre le travail des opérateurs: mesures de radioactivité et de température, localisation et identification de sources radioactives éventuelles. À l’intérieur de certains bâtiments, ces mesures peuvent être complétées par des missions d’inspection télévisuelle de l’état de certains composants ou des missions d’intervention légère, compatibles avec la taille des engins.

2.2. Missions de génie civil

Les missions de type travaux de génie civil ont pour but de contenir la propagation de la contamination et/ou de préparer d’autres interventions: creusement de tranchées, de fossés ou de bassins de récupération des eaux de pluie, décagage de sols contaminés, érection de talus de protection, pompage éventuel des eaux risquant de contaminer la nappe phréatique, etc. Ces missions peuvent être assorties de missions annexes d’assistance et de surveillance télévisuelle du chantier.

3. ARCHITECTURE ENVISAGÉE DU PARC

Dans le cadre des missions globales décrites plus haut, des études ont été entreprises pour définir les fonctions et tâches élémentaires à exécuter, l’architecture du parc qui en découle et le système de transmissions nécessaire pour assurer et gérer l’exécution simultanée de plusieurs tâches et les prises d’information correspondantes.

L’état d’avancement de ces études permet de donner une image représentative du parc de moyens du Groupe Intra. Celui-ci devrait comprendre les matériels suivants:
1) Un équipement de reconnaissance héliporté, capable d’effectuer une cartographie générale de l’activité déposée au sol sur le site accidenté et dans son environnement sur plusieurs centaines d’hectares.
2) Des véhicules tout-terrain, radiocommandés, pour la reconnaissance au sol à l’extérieur des bâtiments du site accidenté. Ces engins doivent être capables d’effectuer une cartographie détaillée de l’activité déposée au sol dans le périmètre du site, et éventuellement dans son voisinage proche, de localiser et d’identifier d’éventuelles sources radioactives. Une partie au moins de ces engins devrait être équipée de télémanipulateurs permettant, le cas échéant, de baliser ou de ramasser ces sources,
de même que de réaliser la surveillance d’un chantier de travaux de génie civil et de lui apporter un minimum d’assistance.

3) Des engins télécommandés de reconnaissance à l’intérieur des bâtiments; ces engins doivent être capables, en empruntant les trajets normalement parcourus par l’homme (couloirs, escaliers, etc.) à l’intérieur de différents bâtiments, d’effectuer une cartographie du niveau de radioactivité ambiante, de localiser, d’identifier et si nécessaire de ramasser d’éventuelles sources radioactives, de réaliser des inspections télévisuelles de certains composants tels que pompes, vannes, circuits d’alimentation, tableaux électriques, etc. Ces engins devraient être aussi capables d’effectuer, en télémécanique ou télémanipulation, un certain nombre d’interventions légères compatibles avec leur taille.

4) Différents types d’engins lourds de génie civil. Ces engins seraient mis à disposition du Groupe Intra dans le cadre de contrats ad hoc, à charge pour le Groupe Intra de réaliser les équipements spécifiques nécessaires pour des interventions en milieu hostile. Deux types d’équipement sont étudiés:
   — des équipements de radioprotection des conducteurs, compatibles avec les capacités des engins, pour les travaux dans les zones dont le niveau d’activité est suffisamment bas pour permettre une conduite manuelle;
   — des équipements de télécommande pour les autres cas.

5) A ces moyens d’intervention «de base» seront adjoints des équipements remplissant des fonctions de service complémentaires, telles que:
   — la surveillance de chantier et assistance à d’autre engins, en particulier aux engins de génie civil;
   — la mise en œuvre des engins de reconnaissance intérieure (amenée à pied d’œuvre à l’entrée des bâtiments, alimentation en énergie, relais de transmission);
   — le relais de transmissions entre engins et postes de pilotage pour l’ensemble des engins en action.

6) Bien entendu, cet ensemble de moyens d’intervention devra s’appuyer sur la logistique nécessaire pour son transport, son alimentation en fluides divers et sa maintenance.

4. ECHEANCES

La démarche du Groupe Intra a été de chercher à réduire les délais dans lesquels il pourrait être opérationnel. Cette démarche l’a conduit à envisager la réalisation de son parc en deux générations d’équipements et à se préoccuper dès le départ des problèmes de mise en œuvre (l’opérabilité).

4.1. La première génération

La première génération du parc est constituée de matériels réalisés avec l’état de l’art actuel et pratiquement sans effort de recherche et développement. Les
matériels de cette génération ne sont pas forcément capables de remplir toutes les missions évoquées plus haut. Ils comprennent principalement:

— des prototypes existants des véhicules de reconnaissance à l’intérieur et à l’extérieur des bâtiments (voir figures 1 et 2). Ces prototypes, légèrement améliorés, sont disponibles au Groupe Intra depuis la mi-1989;

— des versions industrielles de ces prototypes, fiabilisés et aux possibilités accrues par rapport aux missions retenues; ces matériels seront livrés à la mi-1990;

— un équipement héliporté de cartographie radiologique, qui sera livré début 1990.
4.2. La deuxième génération

L’objectif visé pour la deuxième génération du parc est qu’elle soit apte à remplir, dans les meilleures conditions d’exploitation, l’ensemble des missions évoquées aux paragraphes 2 et 3. Elle bénéficiera pour cela d’un programme de recherche et développement d’accompagnement et des progrès récents de la technologie sans toutefois présenter de discontinuité majeure de concept par rapport à la première génération. Les principaux problèmes de recherche et développement se situent dans les domaines des transmissions, de la tenue aux radiations des équipements embarqués et de l’assistance par ordinateur au pilotage.

Cette nouvelle génération du parc est actuellement au stade de l’achèvement des travaux de définition de son architecture globale, y compris le système de transmissions et le parc d’engins de travaux publics. Les études de définition détaillée des différents matériels et de leur logistique devraient s’étaler sur les deux ans à venir, et les réalisations s’échelonner jusqu’à fin 1992/début 1993.
4.3. Mesures d’accompagnement

Conscient du fait que l’opérabilité de son parc est conditionnée, non seulement par la disponibilité des matériels et de leur logistique, mais aussi par une organisation et par une formation du personnel d’intervention adaptées aux spécificités de ses missions et de ses matériels, le Groupe Intra a dès le départ accordé une attention toute particulière à ces problèmes, avec l’objectif de rendre la première génération du parc opérationnelle dès la mi-1990.

C’est ainsi qu’en parallèle aux actions liées à la réalisation du parc, évoquées aux paragraphes précédents, ont été lancées les actions suivantes:
— une étude de l’organisation la mieux adaptée aux moyens, aux missions et aux délais d’intervention spécifiés pour le parc; cette étude doit se traduire, dès 1990, par la définition des plans détaillés d’intervention et de leurs différentes procédures: mobilisation, transport, interventions, maintenance, etc.;
— la définition du plan et des moyens de formation, tenant compte des spécificités des engins du parc; le plan de formation sera mis en place au deuxième trimestre 1990, après une session préliminaire de formation, limitée à quelques pilotes, dans le courant de l’été 1989;
— la recherche et la définition des moyens de transport lourd (de référence et de secours) et de maintenance, en période de veille et en cours d’intervention, y compris la décontamination éventuelle des engins.

5. CONCLUSIONS

Par la création du Groupe Intra, Electricité de France, la Compagnie générale des matières nucléaires et le Commissariat à l’énergie atomique se sont dotés d’un outil commun mobilisable en cas d’accident grave sur l’une de leurs installations. Cet outil doit permettre, dans les premières heures suivant d’éventuels rejets radioactifs, d’assurer la reconnaissance indispensable pour évaluer les nécessités et possibilités d’intervention, en vue de limiter les conséquences de l’accident et de permettre le travail des opérateurs. Au-delà de la reconnaissance, le parc du Groupe pourra participer aux phases de préparation, d’assistance et de suivi des opérations de rétablissement de la situation normale.

LE PLAN POST-ACCIDENTEL RELATIF A UN ACCIDENT A CARACTERE RADIOLOGIQUE

M. GENESCO
Direction de la sécurité civile,
Ministère de l'intérieur,
Paris, France

Abstract—Résumé

POST-ACCIDENT PLANNING FOR AN ACCIDENT WITH RADIOLOGICAL CONSEQUENCES.

In France, every basic nuclear facility has an individual emergency plan. These plans are designed to enable the prefects of the départements concerned to adopt the necessary emergency measures to protect the population in the event of an accident which could result in a release of radioactivity beyond the site of the nuclear facility involved. It has now proved essential to supplement these individual emergency plans by post-accident plans dealing specifically with the measures to be taken after a release has occurred, the aim being to re-establish normal life in the area affected by the releases and to assess the possible consequences. These post-accident plans extend both the period and the area covered by the operational provisions of the individual emergency plans, which are applied at the emergency management stage. Decisions regarding the choice of appropriate measures are based on public health, social and economic criteria. It is essential that the criteria governing such decisions be simple enough for the public to understand the reasoning behind them and the logic of their application. It is important to protect the population primarily against the effects of exposure to radionuclides deposited on the ground, in dwelling places, on agricultural land and so on, and against the effects of internal exposure through inhalation of suspended dust or ingestion of contaminated food products. The post-accident plans deal with all these problems.

LE PLAN POST-ACCIDENTEL RELATIF A UN ACCIDENT A CARACTERE RADIOLOGIQUE.

En France, chaque installation nucléaire de base (INB) fait l'objet d'un plan particulier d'intervention (PPI). Ces plans sont conçus de façon à permettre aux préfets des départements concernés de prendre les mesures d'urgence que nécessiterait la protection des populations en cas d'accident sur une installation et de menace de rejets radioactifs à l'extérieur du site nucléaire. Il est apparu indispensable de compléter ces documents par les plans post-accidentnels (PPA) traitant spécifiquement des mesures à prendre après occurrence d'un rejet afin de rétablir la vie normale dans la zone concernée par les effets des rejets, et d'en apprécier les conséquences éventuelles. Cette planification post-accidentelle vise à compléter, dans le temps et dans l'espace, les dispositions opérationnelles des plans particuliers d'intervention qui s'appliquent à la gestion de la phase d'urgence de l'événement. Les décisions relatives aux choix des mesures appropriées sont fondées sur des critères sanitaires, sociaux et économiques. Il est impératif que ces critères de décision soient suffisamment simples pour que le
public comprenne la logique de leur fondement et de leur application. Il importera de protéger la population essentiellement contre les effets d'une exposition provenant des radionucléides déposés sur les sols, dans les habitations, sur les terres agricoles, etc., ainsi que des effets de l'exposition interne par inhalation de poussières en suspension ou ingestion de produits alimentaires contaminés. Les problèmes correspondants font l'objet de dispositions et de réflexions précisées par le PPA.

1. INTRODUCTION

L'industrie nucléaire s'est préoccupée, à toutes les étapes de son développement, de la prévention des risques d'accident tout en poursuivant des recherches sur les effets biologiques des produits mis en œuvre.

On admet aujourd'hui que la probabilité d'occurrence d'un accident impliquant une installation nucléaire située sur le territoire national et conduisant à des rejets radioactifs de grande ampleur est très faible.

Il est néanmoins du devoir des pouvoirs publics de prendre en compte ce type de situation et de veiller à ce que la protection des populations puisse être assurée dans toute hypothèse conceivable.

Des plans particuliers d'intervention (PPI) ont été établis pour faire face aux risques particuliers liés à l'existence et au fonctionnement sur le territoire français d'installations nucléaires et, notamment, des réacteurs à eau sous pression du programme électronucléaire. Ils sont conçus de façon à permettre aux préfets des départements concernés de prendre les mesures d'urgence que nécessiterait la protection des populations en cas d'accident sur une installation et de menace de rejets radioactifs à l'extérieur du site nucléaire.

Après cette phase d'urgence, il est apparu indispensable de compléter ces documents par les plans post-accidentels (PPA); ils traitent spécifiquement des mesures à prendre après occurrence d'un rejet afin de rétablir dans les meilleures conditions la vie normale dans la zone concernée par les effets des rejets et d'en apprécier au plus juste les conséquences éventuelles à moyenne et longue distances pendant la période appropriée.

Tel est l'objectif de cette planification post-accidentelle, qui vise à compléter, dans le temps et dans l'espace, les dispositions opérationnelles des plans particuliers d'intervention qui s'appliquent à la gestion de la phase d'urgence de l'événement.

Le présent mémoire se propose, tout d'abord, de présenter la méthodologie qui a guidé les études des autorités françaises pour concevoir le PPA, puis d'exposer les difficultés rencontrées au cours des travaux d'élaboration de ce plan, et enfin, de décrire les choix fondamentaux qui ont été retenus en matière de gestion des conséquences à long terme d'un accident majeur d'origine nucléaire.
2. METHODOLOGIE D'APPROCHE

2.1. Scénarios accidentels de référence

Le PPA s'applique essentiellement aux conséquences extérieures d’un accident survenant dans une installation nucléaire civile ou militaire et, en particulier, aux sites de production électronucléaire.

Il peut également être mis en application dans l’hypothèse d’un accident majeur impliquant un transport de matières radioactives ou nucléaires.

En cas d’accident survenant hors du territoire national et conduisant à une élévation du niveau de la radioactivité, certaines dispositions opérationnelles du plan doivent également pouvoir être mises en œuvre.

Enfin, tout ou partie du PPA peut s’appliquer à des événements tels que la chute d’un satellite à propulsion nucléaire ayant produit une contamination significative sur une grande étendue.

S’agissant des centrales électronucléaires, les hypothèses accidentelles sur lesquelles sont fondées les dispositions opérationnelles du présent plan (ainsi que celles du PPI) sont les accidents de type S3.

Les études effectuées par les organismes de sûreté ont permis d’établir les conséquences radiologiques susceptibles de découler d’une telle typologie accidentelle de référence.

Il est rappelé que la typologie accidentelle de type S3 est fondée sur:

— un réacteur REP de 900 MWe avec un rejet principal étalé sur 24 heures;
— le modèle IPSN pour le calcul de la dispersion atmosphérique;
— des conditions météorologiques normales.

Les résultats obtenus portent sur:

— l’activité surfacique à la fin du passage du panache (pour les isotopes les plus significatifs: $^{131}$I, $^{134}$Cs, $^{137}$Cs, $^{90}$Sr, $^{106}$Ru);
— les doses externes (doses efficaces) dues aux dépôts reçues en 1 mois et 1 an de séjour;
— les doses reçues par ingestion de produits contaminés;
— la contamination de la chaîne alimentaire.

2.2. Zone d’application du PPA

On rappelle que les PPI retiennent (pour les REP) la possibilité, avec un préavis de 12 à 24 heures, d’évacuer les populations jusqu’à 5 km et de les confiner à domicile jusqu’à 10 km.
Il importe de noter que les dimensions de la planification en phase post-
accidentelle pourront être sensiblement plus étendues dans les mêmes hypothèses de
scénarios accidentels (accidents de type S3).

La raison en est simple. En effet, les dispositifs de détection de radioactivité
sont extrêmement sensibles. Toute augmentation du niveau d'activité dans
l'atmosphère (par rapport au niveau ambiant dû à la radioactivité naturelle) peut être
détectée. Il en résulte qu'un rejet accidentel de radionucléides dans l'atmosphère peut
être perçu très loin du point d'émission, même si à ces distances il n'aura le plus
souvent pas de conséquences sanitaires.

De ce fait, s'il est naturel que l'attention des pouvoirs publics se focalise dans
les premiers instants sur la zone proche (la seule où peuvent être présents des risques
sanitaires à court terme), il est également nécessaire que, rapidement, les autorités
évaluent les niveaux d'activité en zone peu éloignée (au-delà de 10 km) de façon à
rassurer les populations et prendre s'il y a lieu des mesures complémentaires
(prélèvements, suivis médicaux des populations, voire restrictions de circulation ou
de consommation d'aliments).

La dimension de la zone d'action du PPA peut donc largement dépasser celle
des PPI.

2.3. Continuité de l'action des pouvoirs publics pour le passage de la phase
d'urgence à la phase post-accidentelle

Les PPI relatifs aux centrales nucléaires ont été conçus de façon à permettre
au préfet du département de prendre, au cours de la phase initiale, les mesures d'ur­
gence nécessitées par la protection des populations (évacuation, confinement,
administration d'iode stable).

Au cours de la phase post-accidentelle, les structures de commandement et la
répartition des responsabilités — fixées au niveau des PPI et à l'échelon national —
resteront valables jusqu'au retour à la vie normale. Toutefois, la nature des tâches
va évoluer. Aux mesures réflexes vont succéder des actions fondées sur une appré­
ciation précise de la situation.

Le passage progressif et sans hiatus des dispositions du PPI à celles prévues
par le PPA repose essentiellement sur la prise en compte des actions suivantes:

— l'autorité préfectorale (le préfet) responsable de l'ensemble des opérations
conduites dans le cadre du PPI reste chargée de l'application des actions liées à la
présente phase post-accidentelle;
— l'état-major constitué dans le cadre du PPI pour l'aide à la décision du préfet
voit sa composition adaptée en vue de la gestion de la phase post-accidentelle.

Le rôle de cet état-major constitué auprès du préfet est le suivant:

— veiller à ce que le déploiement du dispositif de contrôle et de surveillance
soit dimensionné à la situation;
— interpréter et établir une synthèse des informations recueillies au plan de la radioactivité dans les différents milieux de l'environnement et dans les denrées alimentaires;
— mettre en œuvre les décisions arrêtées, d'une part, par l'autorité locale, et, d'autre part, au niveau de l'échelon central de coordination activé sous l'égide du Ministère de l'intérieur;
— préparer, au profit du préfet (ou de son représentant) les communiqués périodiques destinés à l'information des populations, après consultation de l'organe central d'information;
— préparer les circuits spéciaux d'approvisionnement des populations dans l'hypothèse où des restrictions de consommation de certains produits ont été décidées.

3. LES DIFFICULTES LIÉES À LA GESTION DE LA PHASE POST-ACCIDENTELLE

Si la gestion de la phase d'urgence repose essentiellement sur des actions réflexes qu'il est relativement aisé de planifier à l'avance, il n'en est pas de même pour ce qui concerne la phase post-accidentelle en raison, notamment:
— du manque de retour d'expérience opérationnelle;
— de la multiplicité des acteurs et la complexité des structures;
— des risques de désorganisation durable des conditions de vie socio-économique de la région affectée;
— de la difficulté de créer les conditions de l'information du public sur un sujet sensible et passionnel.

3.1. Retour d'expérience

Seul (ou presque) l'accident survenu à Tchernobyl est de nature à fournir les enseignements susceptibles d'alimenter la réflexion sur les conséquences à long terme d'un accident majeur à caractère radiologique.

Faute d'expériences de référence, la validité des concepts opérationnels du PPA est testée à la faveur d'exercices nationaux qui simulent la survenue d'un accident à conséquences radiologiques impliquant soit une installation civile, soit une installation militaire.

Chaque année, une manœuvre d'intérêt national est ainsi organisée à l'initiative des services du Premier ministre.
3.2. Multiplicité des acteurs

Pendant la période d’urgence, seuls les services de trois ministères (intérieur, industrie, santé) participent, aux côtés du préfet, à la gestion de la crise.

Au fur et à mesure de l’évolution et du développement des événements, des départements ministériels toujours plus nombreux viennent s’intégrer au dispositif, et en particulier:

— l’agriculture,
— l’équipement et les transports,
— le budget et les finances,
— les affaires étrangères, etc.

Chacune de ces autorités possède ses services extérieurs, ses moyens propres, ses modes de fonctionnement. La coordination de tout cet ensemble de structures et des dispositifs est particulièrement complexe à appréhender.

3.3. Désorganisation durable de la vie sociale et économique

L’exemple de Tchernobyl illustre particulièrement l’acuité de ce problème.

3.4. Processus d’information et de communication avec le public

Une politique d’information repose sur la prise en compte des quatre principes suivants:

— la clarté et la qualité,
— la cohérence,
— la crédibilité,
— la régularité.

3.4.1. La clarté et la qualité

Une information précise et de qualité est de nature à dédramatiser un événement. Le souci des responsables de l’information est donc d’acquérir très rapidement les éléments nécessaires à la formulation d’une information claire et accessible.

Dans le cadre de la phase post-accidentelle, trois domaines essentiels sont susceptibles de susciter les plus fortes demandes d’information de la part du public. Il s’agit:

— de l’évaluation des conséquences, notamment aux plans sanitaire et social de l’événement à court et à long terme;
— de l’évaluation des conséquences au niveau de la contamination de la chaîne alimentaire;
— des délais de retour à la vie normale.

3.4.2. La cohérence

Compte tenu de la diversité des sources d’information, une coordination s’impose. Il convient en effet d’assurer la cohérence de l’information dans le cadre d’une stratégie de la communication définie et mise en œuvre par le responsable de l’information.

3.4.3. La crédibilité de l’information

La première démarche des responsables est de vérifier le bien-fondé des renseignements recueillis. Il est évident que toute diffusion d’information non vérifiée ou erronée peut avoir de graves conséquences dans un domaine aussi sensible: accusations relatives à une rétention de l’information, inquiétudes injustifiées susceptibles de mettre en doute aussi bien la sûreté nucléaire que la capacité des pouvoirs publics à dominer la situation et à prendre les mesures nécessaires.

3.4.4. La régularité

La diffusion de l’information doit être rapide et présenter un caractère de continuité et de permanence. Tout hiatus ou vide d’information dans ce processus serait propice aux actions de désinformation.

Des synthèses de la situation doivent être régulièrement diffusées. Ces synthèses portent sur:
— l’évolution constatée de la contamination des échantillons recueillis et analysés,
— l’évaluation, au plan sanitaire, de ces résultats,
— les consignes à observer,
— les délais de levée de ces consignes.

4. LA DYNAMIQUE ET LES CHOIX OPERATIONNELS

La contamination mesurée sera le plus probablement la conséquence de transferts, par voie atmosphérique, des rejets radioactifs issus de l’installation. Toutefois, une propagation partielle de cette contamination peut être relayée par la voie des eaux (eaux de surface, nappes phréatiques affleurant la surface des sols).
En premier lieu, il convient de s'attacher à la délimitation rapide, par tous les moyens disponibles et en particulier par systèmes héliportés, de l’étendue complète de la zone contaminée.

Cette délimitation étant effectuée, les procédures d’indemnisation (notamment au profit des agriculteurs) doivent être entreprises et rendues publiques.

Il importera de protéger la population essentiellement contre les effets d’une exposition provenant des radionucléides déposés sur les sols (habitations, terres agricoles) ainsi que des effets de l’exposition interne par inhalation de poussières en suspension ou ingestion de produits alimentaires contaminés.

Les problèmes correspondants, à savoir:

— les soins ou suivis médicaux des personnes contaminées et des populations voisines du site,
— l’assistance aux populations évacuées,
— le contrôle des zones à accès réglementé,
— le contrôle des produits agricoles,
— les mesures radiologiques,
— la réhabilitation des zones contaminées,
— l’indemnisation des dommages,

font l’objet de dispositions et de réflexions précisées par le PPA.

5. CONCLUSION

Les décisions relatives aux choix des mesures appropriées vont être fondées sur des critères sanitaires, sociaux et économiques. Il est impératif que ces critères de décision soient suffisamment simples pour que la population comprenne la logique de leur fondement et de leur application.

Enfin, les obligations souscrites au titre des conventions conclues avec l’AIEA seraient bien évidemment mises en application dans un tel contexte.
THE RADIOLOGICAL ACCIDENT IN SAN SALVADOR

J.R. CROFT
National Radiological Protection Board,
Cookridge, Leeds,
United Kingdom

P. ZUNIGA-BELLO, A. KENNEKE
Division of Nuclear Safety,
International Atomic Energy Agency,
Vienna

Abstract

THE RADIOLOGICAL ACCIDENT IN SAN SALVADOR.

In February 1989 a radiological accident occurred in San Salvador at an industrial irradiation facility for sterilizing prepackaged medical products. A movable rack holding a 660 TBq (18 kCi) $^{60}$Co source jammed in the exposed position. The operator managed to bypass degraded safety systems and enter the irradiation chamber and, with two helpers, free the rack and lower it manually into the storage pool. The three workers were exposed to very high doses and developed acute radiation syndrome. Initial treatment locally and subsequent more sophisticated treatment in Mexico City was effective in countering the acute effects. However, the legs and feet of two men were so seriously injured that amputation was necessary. Moreover, despite the medical efforts, the most exposed worker died six months after the accident from radiation induced lung damage complicated by a lung injury sustained during treatment. As there are more than 160 industrial irradiation facilities throughout the world, some in countries with little or no infrastructure for radiological protection, an international review was undertaken to document the facts and define lessons for all with safety responsibilities at such facilities. The paper provides a brief summary of the findings of that review.

1. BACKGROUND TO THE ACCIDENT

The irradiation facility at which the accident occurred was built in 1974 and commissioned in 1975. In it the product to be sterilized is loaded into fibreglass boxes and sent on a conveyor system into a shielded irradiation chamber. In the chamber the boxes are pushed and pulled by a series of pistons on two levels around a rectangular source rack that can be raised from a shielding pool. The upper and lower modules of the rack each have an array of 54 pencils, 14 of which contain $^{60}$Co, whilst the rest are inactive.

In an incident in 1975, product boxes had blocked movement of the source rack, the rack was deformed, and some source pencils fell out on the platform. The
installed safety systems and operator training were effective in preventing overexposures and experts from the company that had originally supplied the equipment recovered the pencils. As a result of this and similar incidents at other irradiators, the supplier advised its customers to install a protective metal shroud to prevent the product boxes interfering with the source rack and to check boxes routinely and replace those of marginal quality. However, at this facility the recommended shroud was never installed and by the time of the accident many boxes were in extremely poor condition and held together with tape.

Safety at the facility had, over the years, deteriorated significantly in other respects. This process had been exacerbated by economic and security effects of the civil war in El Salvador which has been going on since 1979. The facility had changed ownership several times, with the first change leading to the operators, who had been formally trained by the supplier, leaving within a year. Subsequently, operators received only informal on-the-job training. During the civil war the company regarded the facility as a high technology one that, if given a high profile, could make a desirable target for bombing. As a result, not only was its presence not advertised, but there was a reluctance to commit anything to writing, even safety and operating procedures. A ‘make do and mend’ approach was prevalent. No preventative maintenance was carried out; key safety systems were not repaired and some were even removed. This led to the adoption of improper entry procedures.

The country had no relevant regulatory control or other form of radiological protection infrastructure. Normally the supplier would expect to visit most plants once every two or three years for source replenishment and at that time would notice any serious deficiencies and instigate corrective actions. The economic and security aspects of the civil war had limited contact since 1977 to the telephone. This lack of regulatory control and lack of contact with radiological protection expertise provided the right environment for a continuous degradation in the safety procedures and systems.

The overall state of affairs can be summarized as being “an accident waiting to happen”. At 02:00 on Sunday 5 February 1989, this potential was realized.

2. THE ACCIDENT

Worker A had joined the facility a year before as a maintenance technician and later also became a shift operator. He was known for his resourcefulness in solving frequent maintenance and operating problems at the facility. Operating alone on the Saturday night shift, he managed to keep the operation going despite a number of power failures and piston problems. At about 02:00 on Sunday morning a fault caused the source rack to be lowered automatically from the irradiation position, triggering the source transit alarm. This kept on ringing, indicating that the rack had not returned to the fully down, shielded position.
FIG. 1. Positions adopted during part of freeing of the source rack (viewed from above).
When he could not reset the alarm, Worker A followed a procedure developed at the facility for use in such situations — a procedure not recommended by the supplier. He applied an overpressure to the hydraulic hoist to force the source rack fully up so that it would then return to the fully down position. When this did not work, he further manipulated the control system and this induced the green source-down light to go on.

The radiation monitor originally installed in the chamber would have indicated the source was still up, but this fixed monitor had been removed years before. To unlock the door in the absence of the fixed monitor, Worker A followed another procedure developed at the facility. He rapidly cycled buttons on the control panel to simulate monitor detection of normal background levels. At about 02:30 the door opened. He waited for a few minutes to allow ozone — and probably in his mind also radiation — to ventilate and then switched off the power supply to the facility and entered.

FIG. 2. Dose rate contours (Gy·min⁻¹) for Worker A during part of his exposure.
Aware that the rack was still in the exposed position but not appreciating the danger, Worker A entered the radiation room without checking the radiation level with the portable monitor. He found, in the row immediately adjacent to the source rack, five boxes jammed into the space of the normal four. Working by flashlight, he removed some boxes and could see the stuck source rack. Unable to free the stuck rack by himself, he left the chamber, turned the power on and went to seek help.

At about 03:00, Worker A returned with Workers B and C from another department of the plant. Assured by Worker A that there was no danger, they entered the chamber and helped him to remove boxes from the upper level adjacent to the source so that the rack could be freed from above. Standing side by side along the upper level, Worker A crouched in front of the rack, with Worker B to his right and Worker C standing further off to the side. (See Figs 1 and 2). They lifted the rack by pulling on the hoist cable and then paid out the cable to lower the rack into the pool. Seeing the Cerenkov blue glow, they hurriedly left the chamber.

Worker A soon began vomiting, having been first exposed an hour before and to more radiation than the others. At 03:30 Workers A and B went to the emergency clinic of a nearby hospital and by then Worker B also was vomiting. After returning to work, Worker C also began to vomit and went to the hospital. They mentioned the radiation source at the plant, but no other symptoms of radiation exposure were then manifest. As has been the case in many previous accidents they were misdiagnosed as having food poisoning, given three-day sick leave certificates and discharged at about 06:00.

2.1. Further exposures

At that same hour Worker D reported for the day shift to find the main door open, the facility shut down, product boxes in disorder, and no sign of Worker A. Worker D straightened up and restarted the facility and, when Worker A did not return for the night shift, ran the operation for a full 24 hours. Although Worker D reported the situation on Monday morning to the maintenance manager, everything seemed in order, so for the rest of the week the facility was operated more or less normally. A notable exception was that on Wednesday 8 February the source rack became stuck again but was released by the overpressure technique.

At noon on Friday 10 February, after QA dosimetry had indicated product dose to be low, the maintenance manager and the QA specialist entered the radiation room. They observed that source pencils were missing from the upper module and, from the Cerenkov glow, that some were lying on the bottom of the pool. The significance of this was not recognized and since the radiation level was normal, operation was continued with longer exposure times.

At 16:00, after another failure, the source rack could not be returned to the storage position. The radiation level was checked with the portable 'beeper' monitor and found to be high, indicating that the source was stuck in the exposed position.
Using their ‘usual’ overpressure method, they evidently managed to lower the source rack, since the source-down light went on and the beep rate went down. However, they were measuring from outside the access door and so did not realise that a noise they heard was due to the remaining source pencils in the upper module being knocked out. While most of them fell into the pool, four of the pencils were left on the conveyor platform. One of them was an active one, containing approximately 23 TBq (620 Ci) of $^{60}\text{Co}$.

Two workers entered the radiation room without further checking the radiation level and, unable to see anything wrong, they asked the maintenance manager and another worker to look. The manager noted that the source rack was indeed in the pool, but that the upper source module was then completely empty of pencils. He left the radiation room to bring the monitor and, pointing it back into the maze, found the dose rate above normal. He closed the access door and had the source rack raised and lowered several times. It moved without difficulty, but the radiation level remained elevated. He concluded something was wrong and ordered the facility closed. Only at this point was the potential for further exposure under control.

None of the workers wore personal dosimeters. The fact that people had been exposed above occupational dose limits was only discovered at a significantly later date because of cytogenetic analysis of blood samples related to the first accident. Four persons were estimated to have received between 90 and 220 mGy in this second exposure.

3. THE RESPONSE TO THE ACCIDENT

The supplier of the equipment was contacted and two experts were sent who determined by means of remote television inspection and a radiation monitor sent into the chamber on the conveyor that there was an active pencil on the upper level of the conveyor system. They drilled a hole through the concrete roof and on 15 February, ten days after Worker A's original entry, they managed to return the active pencil into the pool. Having removed the immediate hazard, and unaware of both the preceding overexposure events, the supplier’s experts, appalled at the state of the safety systems, disabled the operating mechanism and returned to their headquarters. Prior to this they had surveyed the factory to ensure no source pencil had exited the facility and was in the factory. Debris from the drilling made the water too dirty for an inventory of the active source pencils to be done at the time and the pool filtration system, long out of operation, had first to be repaired. Subsequently, in March, a visual check seemed to indicate that all pencils were accounted for but a conclusive inventory was not completed until significantly later.

Meanwhile, on 7 February, Worker A returned to the emergency clinic with not only nausea and vomiting, but also, general erythema and burns on his legs and feet. In the light of his statements about events at the plant, he was hospitalized as
having 'radiation burns' from 'acute exposure to cobalt'. He was placed in reverse isolation to limit the chance of infection. Blood and other tests were initiated, and symptomatic supportive treatment was begun. The treatment initially controlled the situation, but his overall condition began to deteriorate. By 15 February the physicians concluded he should be transferred to a facility with bone marrow transplant capability and on 28 February he was transferred to Hospital Angeles del Pedregal in Mexico City, where an experienced team had been assembled. Worker B was hospitalized on 13 February and transferred to Mexico City on 2 March. Worker C, markedly less ill than Workers A or B, was in and out of hospital several times and finally transferred to Mexico City on 9 March.

Patient A's treatment included a careful nutritional regime and the use of an experimental drug known as GMCSF that may accelerate bone marrow recovery (as in his case a bone marrow transplant was considered more likely to harm than to help). The overall regime led to steady improvement but with ongoing nutritional problems. However, his leg burns were limiting his general recovery and, when gangrene appeared four months after the accident, one leg was amputated above the knee. Although the prognosis as to his continued recovery was guarded, he improved sufficiently in another month and a half to return on 27 July under close medical surveillance to more familiar surroundings in San Salvador. Although he continued to make progress after his return to his country, his other leg was not healing and the likelihood was great that amputation would prove necessary. On 10 August his condition began to deteriorate. On 14 August, after he contracted pneumonia, a catheter placed in a neck artery punctured his lung. After a week in intensive care in critical condition, he died, six and a half months after the accident. Autopsy was not permitted by his family but death was ascribed to residual radiation damage to the lungs complicated by the traumatic perforation.

Effects in Patient B developed more slowly and to a lesser degree, but the treatment regime was similar. Patient B's extremity burns also were severe and necrosis of a toe ultimately led to amputation of a leg. After that he made sufficient progress to return with Patient A to San Salvador under medical surveillance. He continued to improve, but progress was slow owing to the worsening condition of his other leg. After its amputation nearly seven months after the accident, recovery was more rapid and his prognosis is good.

Patient C had less severe symptoms and burns and thus the treatment regime was easier. As he had no other complications, he was returned to San Salvador on 31 March for continued medical supervision. Patient C returned to work with the company on 22 August and later began further rehabilitation treatment to relieve residual chronic effects, particularly in his left foot. The prognosis is good for full recovery.

International involvement began on 24 February when a telex was sent to the IAEA to report "a case of radioactive contamination" and requesting "experts and equipment" and help "to determine ... the effects ... produced." Lacking an emer-
On 28 February the El Salvador radiation protection advisor (the only one working in this field in the country) clarified that medical assistance was needed for three persons in serious condition following exposure to a radioactive source and that there was no contamination. The IAEA staff requested the US Radiation Emergency Assistance Center/Training Site (REAC/TS) in Oak Ridge to send experts to El Salvador to assist in the medical treatment. The mission, delayed because the patients were being transferred, reached Mexico City on 8 March; the team included a physicist to make theoretical dose estimates and a representative of the Pan American Health Organization/World Health Organization (PAHO/WHO), who interviewed the patients and workers at the facility. This process contributed a great deal to the understanding of what had happened, however, in view of the uncertainties in the configurations and lengths of exposure it was concluded that the best estimates of whole body exposure came from cytogenetic tests made on blood samples. These indicated whole body doses of 8.1, 3.7 and 2.9 Gy for patients A, B and C respectively. The doses received were extremely non-uniform and Fig. 3 shows the estimated dose distribution based on the onset and extent of epilation, wet and dry desquamation and early signs of necrotic lesions.
4. LESSONS LEARNED

On the basis of the findings on the San Salvador accident, the IAEA report provided generic recommendations for those having safety responsibilities for such facilities and, of course, many recommendations have a wider applicability. These are briefly summarized below.

(1) *Organizations operating irradiation facilities should ensure that:*

(a) safety systems conform to currently recommended standards  
(b) preventive maintenance is part of the operating plan  
(c) periodic safety audits are done  
(d) operator training in radiation safety is separate from that for production operations and based on up-to-date written procedures  
(e) there are effective arrangements for notifying the authorities about radiation accidents and for initiating action to limit residual hazards.

(2) *The management of the operating organization should, as a minimum, manifest continuing recognition of its primary responsibility for safety by:*

(a) involving itself fully in radiation protection matters  
(b) appointing a qualified radiation safety officer with full authority  
(c) emphasizing to plant personnel the primary importance of safety  
(d) welcoming periodic outside safety review.

(3) *All national authorities should put into place at least the minimum infrastructure needed for radiation safety supervision:*

(a) enabling legislation, a central regulatory authority, and implementing regulations  
(b) an organization adequate to carry out a national inventory of major radiation sources, to register them, and to ensure the essential safety services are provided  
(c) an emergency response plan to ensure speedy notification of accidents, prompt medical treatment, adequate information to the public, and follow-up to determine causes and define corrective action.

(4) *Suppliers of irradiation facilities should:*

(a) maintain regular two-way communication with operating organizations, in all aspects related with radiation safety, and whenever practicable include periodic facility visits with safety audits  
(b) develop and maintain links to national regulatory authorities and international safety organizations, so that they can be alerted before major weaknesses in safety can do harm. (See also recommendation 6.)
(5) The medical community should:

(a) make continued efforts to widen the familiarity of medical practitioners with acute radiation syndrome symptoms and possible accident incidence, so that serious overexposure is more likely to be recognized, thus enabling both appropriate treatment and feedback to prevent any continuing hazard.

(6) International organizations and governments which support the installation of irradiators and other major source facilities should:

(a) make every effort to help responsible officials identify and correct serious safety weaknesses at existing facilities

(b) make every effort to assure that no new facilities are installed unless the operating organizations are competent and adequate regulatory supervision is in place.

5. CONCLUSION

Many of the above recommendations might be described as well known safety aspects of radiological protection, and the accident has simply served to reinforce them. Indeed the accident is a classic example of what can happen, even to a plant with initially good safety features and procedures, when it is operated without the benefits of a supporting radiological protection infrastructure or any regulatory control. It is to be hoped that those countries that do not yet have these features fully in place will be able to benefit from the experience of the San Salvador accident and make appropriate improvements.

An important point to come out of the accident review [1] is that there are more than 160 of these irradiators and an unknown number of other major sources of radiation scattered around the world, some being in countries without a developed radiological protection infrastructure. In many cases the purchase of these facilities has been facilitated by governments and international agencies. Perhaps the most important new recommendations stemming from this accident review is that nations and international agencies facilitating the purchase of these types of sources should jointly investigate with the suppliers possible means of identifying facilities that might be at risk so that efforts can be made to prevent a repetition of the serious degradation of safety found in the San Salvador accident.

REFERENCE

APPLYING LESSONS LEARNED FROM A VARIETY OF EVENTS TO THE MANAGEMENT OF RADIOLOGICAL RECOVERY OPERATIONS IN THE USA

B.H. WEISS, G.G. ZECH
United States Nuclear Regulatory Commission,
Washington, D.C.,
United States of America

Abstract

APPLYING LESSONS LEARNED FROM A VARIETY OF EVENTS TO THE MANAGEMENT OF RADIOLOGICAL RECOVERY OPERATIONS IN THE USA.

There is a great deal to learn from the experience of non-radiological accidents as well as the numerous nuclear power plant exercises and emergencies such as the Chernobyl and Three Mile Island accidents. This paper reviews a number of such events and discusses lessons learned from these and how this knowledge is being incorporated into the US plans for recovery operations in responding to radiological emergencies. Although the initiating events and attendant technologies may differ, the human responses to major emergencies have many similarities. These experiences show that we must continue to learn that continuing vigilance in preparing for emergencies is of utmost importance and that the initial response operations have a great impact on the recovery operations. The paper also discusses the lessons learned with regard to the collection and assessment of environmental data, establishment of consensus recovery standards and advice, communication improvements, international interdependencies and other common concerns.

1. INTRODUCTION

There have recently been several significant insults to the environment, which have required extensive recovery and cleanup efforts. These events are receiving attention all over the world. Government and industrial response organizations should be able to learn a great deal from the non-radiological as well as the radiological incidents in the USA and other countries. The lessons learned from these events that involve recovery operations are having a profound effect on the approach to US response and recovery operations.

At the conclusion of each major incident and exercise, one or more US agencies have considered the lessons learned from the event, particularly how the USA can more effectively respond to a similar event in the future. This paper reviews the major lessons learned from a series of actual incidents and comprehensive exercises to see how the US and other response systems can be improved. In particular, the
paper focuses on the changes that will be necessary in order for the USA to be better prepared to initiate recovery operations in a timely and efficient manner following a major radiological incident. The primary events that were considered for this paper were:

- Three Mile Island accident (March 1979)
- Chernobyl accident (April 1986)
- Sequoyah fuels incident (January 1986)
- COSMOS 1900 reentry preparations (September 1986)
- 1988 Soviet and 1986 Mexican earthquakes
- Major Oil and Hazardous Materials Responses
- Federal Field Exercise 1 (St. Lucie, March 1984)
- Federal Field Exercise 2 (Zion, June 1987)
- Relocation Tabletop Exercise (Beaver Valley, December 1985).

The review of the lessons learned for recovery operations from these events indicated considerable consistency. As a result, there is confidence that making these changes will significantly improve the US response and recovery operations for a radiological incident and possibly for incidents involving other hazardous materials.

2. THE NATURE OF FEDERAL RESPONSE

Perhaps the most important lesson from the Three Mile Island accident and the recent Alaskan oil spill is that all organizations need to maintain continuing vigilance in preparing for potential emergencies and the consequences of such events and not assume that accidents have been designed out of modern technological operations. We must continue to train, plan, and develop systems for responding to emergencies and be prepared to institute effective mitigation and recovery operations in a timely manner. This point was profoundly demonstrated when an aircraft completely lost use of its hydraulic system and had to crashland at a small airport in the USA. The local emergency response units were alerted to the situation and were able to deploy to the scene before the unfortunate crash of the airliner during its aborted landing. These emergency units had taken their responsibilities seriously and had recently conducted an exercise involving a similar situation. This commitment and preparation resulted in the saving of many lives and the avoidance of panic among the surviving passengers.

In the USA, the States have the primary responsibility for protecting public health and safety while the Federal agencies provide the comprehensive support that no single State can afford to maintain. Although States would like to maintain their sovereignty even in an emergency, the scope of the Chernobyl event, the large federal exercises, and the international responses to the Alaskan oil spill and the Mexican and Armenian earthquakes have convinced State officials that they will need extensive Federal support during a major radiological event. No US State can handle
the enormous demands of a severe radiological incident. On the international level, it is becoming clear that there is probably no country that can claim to be independent in responding to and cleaning up major contamination insults to the environment.

Another principle that has been reinforced is the need, as much as possible, to make decisions at the scene of the accident, as soon as possible, and during the recovery phase. It is evident that the crucial technical and human decisions must be made close to the scene of the emergency where the information is best, the communication lines are not stretched and the climate is somewhat less political. Where possible, a single individual with appropriate authority should be in command. In the USA, that will generally be the Governor of the affected State. All other groups should be working towards supporting that decision maker.

In all of the events listed above, distinct phases of an emergency emerged. These phases were all related, but each phase demanded different capabilities, organizations and approaches. An accident is not a static event and the recovery operations need to be tailored accordingly. The Federal Radiological Emergency Response Plan (FRERP) is being modified so that these differences are purposefully considered and the recovery portion of the response will be addressed early in the event by those specific groups and institutions that may not have the expertise for the initial response but will play a major role in the recovery activity. The FRERP will recognize that the primary response in the early phase of a response will be primarily focused on the on-site situation and the immediate concerns. Therefore, a lead Federal agency will be designated to begin immediately to consider the actual and potential recovery concerns.

The events reviewed provide evidence that the Federal plan provides the appropriate operational concept, with some enhancement and modification, to be used in the recovery phase. Although the FRERP will be enhanced by describing the concept of operations for this phase of a radiological response, the plan will not require the same level of procedural details as for the initial response. We have learned that, after the immediate response, each event has unique parameters: developing a detailed recovery response based on past events would be difficult and wasteful. It will be useful though, to have in place basic guidance and an infrastructure that can begin assembling and acting early in the incident so that recovery operations can be initiated as soon as practicable.

Another aspect of the response to the various events was the considerable interest, and involvement of, the international community, whether countries were directly or indirectly involved or not affected at all. This was, obviously, most pronounced during the Chernobyl event, but was noted to some degree in all of the events reviewed. In addition, the IAEA Conventions relating to Early Notification and Radiological Assistance need some consideration in the development of national emergency response plans. The US plan will be expanded to make the Department of State a full partner in the FRERP and focus on this important aspect of the response and recovery phases.
Another aspect which has been seen in the actual events, particularly the earthquakes, is the provision of unsolicited international assistance. Although well intentioned, the provision of unnecessary donations can divert precious resources from their primary mission. There needs to be a better international process for controlling this support to enhance the effectiveness of the overall recovery operations. The implementation of the Convention on Radiological Assistance may be able to assist in limiting this phenomenon in the response to major radiological accidents. The IAEA will be in a position to offer its good offices to co-ordinate assistance to those States requesting support and act as a ‘broker’ between States in responding to requests for radiological assistance.

3. COLLECTION AND ASSESSMENT OF ENVIRONMENTAL DATA

Each event has demonstrated the enormous amount of data that will be collected by a large number of unrelated monitoring and assessment groups. We have learned that having a large quantity of data of unknown quality does not provide the information necessary to make the necessary definitive decisions. The need for credible and assessed data is paramount. The decision makers and the public will demand a huge amount of data in an unrealistic time frame. The Federal Plan recognizes this fact and is developing various approaches for satisfying this demand in a more timely fashion.

In addition, a large effort is now going on to standardize the US monitoring, assessment and advisory support, no matter what kind of radiological situation is encountered. One approach which has been proposed and tried in various events is the establishment of two interagency advisory groups to support States and develop consensus actions and intervention levels. One is a highly technical group which will look at the particular situation as it is unfolding and begin to focus on the activities, tasks, resources, and capabilities that will be required to initiate off-site cleanup recovery and other operations and begin the necessary preparations with State officials. The second, which is discussed in more detail below, will be at a more senior policy level and will focus on developing national consensus recommendations for protecting the health and food of the public. The success of these advisory groups in exercises and actual events have provided planners with the confidence to include these groups in the revised Federal Plan. The US agencies have also been impressed with the need for more standard collection and analytical procedures.

4. ESTABLISHMENT OF CONSENSUS RECOVERY STANDARDS AND ADVICE

The US Environmental Protection Agency has been developing Protective Action Guides (PAGs) for the recovery phase of a major radiological accident.
Drafts of these have been available for various exercises and real events. They have been provided a good test and appropriate changes over the past few years have improved their applicability and usefulness. Such guides will be needed immediately by any group which will begin to consider how to approach the inevitable cleanup of contamination. If such a group has not been pre-identified, experience in these events shows that such an advisory group should be organized quickly and must include regional authorities (States) as well as national participation, which has been shown to be both necessary and invaluable.

In the USA during the response to the Chernobyl accident, the Department of State organized a group which became the model for the Food and Health Effects Advisory Group, which was formalized in the preparations for the possible re-entry of COSMOS 1900. This group represented senior policy makers in Washington whose role was to promote uniform national implementation of protective actions by providing a forum to develop and furnish State and Federal decision makers with recommendations to protect public health in the USA and overseas from radiation exposure, either direct or through ingestion of contaminated food and food products. Although it is recognized that the best information is at the accident site, there is a pragmatic realization that senior politicians will be involved in major decisions. Consequently this group has been well accepted by the response community and, as noted above, will be incorporated into the next revision of the Federal Plan.

The COSMOS 1900 episode also taught us how well the response community can work together. This experience supports the conclusion that the community should not overproceduralize the recovery phase. Too much planning and detailed organization structures will most likely get in the way of the particular scenario.

5. COMMUNICATION IMPROVEMENTS

As a result of the Chernobyl accident, two international conventions have been established for radiological accidents, i.e. early notification and radiological assistance. The IAEA has begun implementing these conventions, but some limited experience indicates that not all countries are familiar with the conventions and fail to utilize them properly. Better and more rapid international communications will be needed to improve the international reporting process and provide timely information to those countries that need information immediately to protect their citizens and determine what actions may be necessary for assuring a safe food supply.

During recent events, US organizations have been able to utilize commercial electronic mail systems to communicate nationally and internationally. This has provided an effective means of communication which should be expanded to other organizations and countries.
6. MEDIA INVOLVEMENT

The events reviewed reinforced the notion that a central news centre where the major responders will come together is absolutely necessary to instil confidence in the public with respect to the institutions that are serving the public and protecting their health and safety. Such an operation must be part of the recovery operations. Experience indicates, however, that it will be difficult, frustrating and extremely important to provide information to the numerous groups that have an interest in the recovery operations.

Experience has also shown that the media may provide a false impression of the impact of the disaster. This may hamper recovery operations because it could create organizational conflicts and require significant resources to respond to falsehoods and rumours. Timely, truthful, and responsive information appears to be the most effective way of satisfying the media representatives.

7. OTHER

Experience in these events indicates that photo and video documentation of the recovery operations is extremely important and useful. Responders have indicated that they have made valuable use of the documentation during recovery operations. Universally, they indicate that additional documentation would have been useful.

8. CONCLUSION

These lessons learned have provided valuable guidance for updating the US Federal Plan for responding to radiological emergencies. We believe that these lessons can be quite useful for other countries and other disciplines should consider using the experience of the radiological community in the design of their response and recovery operations.

BIBLIOGRAPHY


LESSONS LEARNED FROM
THE RADIOLOGICAL ACCIDENT IN GOIÂNIA

J.J. ROZENTAL, C.E. DE ALMEIDA, J.J. LABORNE
Comissão Nacional de Energia Nuclear,
Rio de Janeiro,
Brazil

Abstract

LESSONS LEARNED FROM THE RADIOLOGICAL ACCIDENT IN GOIÂNIA.

The Goiânia accident, in September 1987, presents lessons which cover the pre-accident period, the emergency phase, and the post-accident phase up to the present. Each of those phases, taking into consideration errors, omissions and correct actions taken by people and organizations involved, is thoroughly analysed from technical, social, political and economic points of view. Subsequently the paper deals with the influence of legislation and rules applied; the interfaces between the State, the Province and the National Nuclear Energy Commission CNEN; the main problems and the solutions found during the search for, the identification, tracking and interdiction of contaminated places; the decontamination of people, places and belongings; the different aspects involved in providing information and reports to the government, to the organizations and to the community. Special attention is paid to the general support organization, responsible for co-ordination, managing the logistic activities and the participation of several public and private entities.

1. INTRODUCTION

Radioactive materials are encountered in everyday life and are frequently transported by road, sea and air. Incidents or accidents are therefore unavoidable and, in the event of loss of control, radiation sources may lead to overexposure of workers and members of the public. Table I pictures, in terms of fatalities, accidents occurring in this decade as a result of lack of control and inadequate handling or use of sealed sources.

Some reasons for not achieving control of radiation sources, according to the IAEA, are:

— Lack of an appropriate legal and regulatory framework;
— Lack of an effective radiation protection substructure (notification, registration, licensing, inspection);
— Insufficient training of personnel in the safe handling of radiation sources.
TABLE I. FATALITIES FROM RADIATION ACCIDENTS ARISING FROM SEALED SOURCES IN THE NINETEEN-EIGHTIES

<table>
<thead>
<tr>
<th>Year</th>
<th>Location</th>
<th>Radiation source</th>
<th>Fatalities</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Workers</td>
</tr>
<tr>
<td>1981</td>
<td>Oklahoma, USA</td>
<td>Industrial radiography</td>
<td>1</td>
</tr>
<tr>
<td>1982</td>
<td>Norway</td>
<td>Sterilizer facility</td>
<td>1</td>
</tr>
<tr>
<td>1984</td>
<td>Morocco</td>
<td>Lost Ind. Rad.</td>
<td>8</td>
</tr>
<tr>
<td>1987</td>
<td>Goiânia</td>
<td>Teletherapy device</td>
<td>4</td>
</tr>
</tbody>
</table>

The radiological accident in Goiânia was also a consequence of lack of control. This is surprising, since the management of the facility was composed of radiotherapists and included a physicist specializing in radiation protection.

Notwithstanding all the recommendations contained in publications concerning emergency response planning and preparedness, this radiological accident showed that several adverse vectors not mentioned in the literature were a reality. Thus, not only social, political, economic and technical problems had to be faced but also psychological aspects, discrimination being the most important.

The present paper discusses several objective and subjective aspects related to the actions taken, that were necessary to minimize the exposure of the population to the external radiation and contamination, and the establishment of initial procedures for the recovery operations undertaken by the Comissão Nacional de Energia Nuclear (CNEN) in Goiânia.

Two other points must be mentioned, with their lessons for the future:

— Where radioactive sources are being used or stored on a site, even a temporary site (i.e. for industrial radiography, moisture/density gauges, brachitherapy) the organization should have specific control procedures, especially in the event of an incident.

— Even if the greatest care is taken, accidents are sometimes unavoidable and thus it is absolutely necessary to be well prepared to deal with any that may occur.
2. QUESTIONS AND ATTITUDES OF THE POPULATION

— What happened?
— What is caesium?
— What is radioactivity?
— Can it cause cancer?
— How long will it take to feel the consequences?
— What are the effects on children?
— What are the effects on women?
— Is it infectious?
— What about water and food, can we consume them?
— Is it dangerous where I live?
— Where can I walk?
— Are birds, chickens, pigs and other animals (cats, dogs) in the neighbourhood dangerous?
— How does caesium disseminate?
— Will our life expectancy be reduced?
— "I don’t understand what are you talking about."
— "I won’t get out of here. Do you want me to lose my belongings?"

3. LESSONS LEARNED BY CNEN FROM GOIÂNIA

The following lessons were learned from the Goiânia accident.

(1) A radiological accident due to the break-up of radioactive sources might be aggravated if there is a time lapse between the break-up and its becoming known.
(2) The physical and chemical characteristics of the source are important factors. Records of the sealed sources contain that information. It is suggested that physical and chemical characteristics of sources should be considered in the licensing for manufacturing of such sources, placing emphasis on the consequences of possible accidents and misuse of such sources.
(3) An adequate system of information to prevent public panic is essential. The public should be aware of what radioactivity is, and how radioactive sources are used. In order to deal with the media, a booklet explaining the special terms and units should be available. A special group should be prepared, in cases of emergencies, to present information in schools, churches, parliament, all community associations, and to the media. Personnel working in decontamination procedures and attending victims should be taught how to give information in a language that is understood by the population. Their relationship with the individuals involved in these accidents could be very important; people would try to evaluate the seriousness of the contami-
nation from the manner and attitude of such personnel. The people in the critical area of the accident tested if their households and homes were really free of contamination by inviting the CNEN personnel to accept a cup of water or coffee.

(4) An adequate system of social and psychological support should be provided. The psychological support should cover the individuals involved directly and indirectly in the affected areas and personnel working in the emergency situation. A team of psychologists should be available to provide help, joining the group responsible for quick decisions and planning of action to be taken, evaluating the stress that these decisions would cause to the victims.

(5) The effectiveness of international co-operation depends on the country's infrastructure. The emergency co-ordinated training course of the IAEA should be carried on in underdeveloped and in developing countries and not in developed ones, where all the available facilities exist and work well. In general, these programmes cover emergencies in a well organized organization, with a priori known conditions. In many countries the situation is very different, the instruments are different, the climate is different and the administrative bodies work differently.

(6) A system for air transport of first aid must always be available.

(7) An IAEA record of equipment resources should be available and ready for use. Customs regulations should be adjusted to facilitate import and re-export of complementary material and/or equipment. The IAEA should have a set of equipment ready to be shipped and establish a regional centre for emergency attendance in every continent.

(8) Instrumentation should be adjusted for use in field conditions, e.g. high humidity, high temperature and unstable environmental conditions. Personnel handling these instruments should be trained to obtain a clear indication of dose rate response for a wide range of doses, knowing which are the most suitable instruments for each case and the calibration factors.

(9) Records of specialist human resources available should be kept. Experts from the IAEA, in each area of action, should be contacted if an emergency occurs, and should give support to the local radioprotection teams. These experts should be ready to advise actively on decision making and intervention measures, and to get involved in all the steps of the work that needs to be done. Experience has indicated that the 'better' reports were prepared by experts not directly involved.

(10) A transitory waste repository near the affected areas is considered indispensable. A delay in deciding upon a site, most of the time for political reasons, could be responsible for the dissemination of activity into the environment.

(11) An infrastructure of civil engineering personnel should exist, available to act in decontamination procedures.

(12) When dealing with decision making and the organization of working teams after an accident, the hierarchy should be well defined. The responsibilities for the decision process, from planning to action and evaluation of consequences, should be very clear and each group should know very well its responsibilities. If possible,
teams should be formed with a leader who is the head of the group in normal working conditions.

(13) In general, although a programme of inspection is very important, it is only effective if it is associated with some kind of enforcement system, entailing civil or professional liability.
THE CONTRIBUTION OF THE WORLD HEALTH ORGANIZATION TO INTERNATIONAL CO-OPERATION IN MEDICAL PREPAREDNESS FOR RADIATION EMERGENCIES

I. RIABOUKHINE
World Health Organization,
Geneva

Abstract

THE CONTRIBUTION OF THE WORLD HEALTH ORGANIZATION TO INTERNATIONAL CO-OPERATION IN MEDICAL PREPAREDNESS FOR RADIATION EMERGENCIES.

Ideally, each Member State of WHO should have its national plan of preparedness for radiation emergency medical assistance to affected persons. In fact, only a limited number of countries are able to carry out the wide range of actions on the medical handling of a radiation emergency. Thus, it appears reasonable to establish a global network of a few specialized centres with international responsibilities under the WHO umbrella. In May 1988 the World Health Assembly took the decision to accede to the Convention on Assistance in the case of a Nuclear Accident or Radiological Emergency. Under the provisions of the Convention, WHO can be regarded as one of the appropriate international intergovernmental organizations which may be directly called upon for assistance. For the promotion of radiation emergency medical preparedness and for practical assistance to countries in case of overexposure from any source of radiation, WHO has established 6 collaborating centres: in France, the USA, the USSR, Argentina, Brazil and Australia. These centres serve as focal points for advice and training; assist in the establishment of medical emergency plans for large scale radiation accidents; initiate co-ordinated studies on human radiopathology and radiation epidemiology; and assist in the preparation of relevant documents, guidelines and meetings. In the event of a radiation accident, the collaborating centres could provide a team for on-site medical treatment; a team for rapid external radiation monitoring and/or contamination surveys with appropriate equipment; transportation of patients; facilities and staff for medical examination and treatment; follow-up medical supervision and treatment. The experience and resources of the collaborating centres in France and the USA have already been used on several occasions for international help in radiation emergencies. At present, efforts are under way to increase the number of WHO collaborating centres to about ten and to establish closer ties among them so that an international network of WHO collaborating centres could be set up. As a result some kind of global radiation emergency medical service would be created. Countries which do not have collaborating centres can be involved in the network through their 'liaison institutions', i.e. national points of contact with the appropriate collaborating centre(s), and/or with WHO. A contribution is also expected from 'support institutions', i.e. to assist in solving such national institutions as could be activated or invited to assist in solving particular problems, especially in an emergency situation. The role of WHO is reflected in the Emergency Notification and Assistance Technical Operations Manual (ENATOM) developed by the IAEA as a guide for States party to the Convention.
1. BACKGROUND

The use of nuclear power, industrial and medical applications of radiation and radionuclides from time to time gives rise to radiation accidents. Some of these accidents have entailed overexposure, i.e. irradiation above the limits established for both radiation workers and the public.

Serious radiation injuries are due in most cases to external radiation from X-ray and radionuclide sources. Second in frequency is internal contamination with radionuclides. Reactor accidents involving overexposure are very rare. In general, the risk of serious health hazards from the use of nuclear power, radiation and radionuclides is much less than that from many major activities of humans. Nevertheless radiation accidents, particularly nuclear ones, even though rare, have many medical, administrative, legal, social and psychological implications.

2. NEED FOR INTERNATIONAL CO-OPERATION

Ideally, each Member State of WHO should have its national plan of preparedness for medical assistance to persons affected by radiation. Such a plan should be backed by adequate capability for putting it into effect. In fact, only a limited number of countries are able to carry out the wide range of actions needed for the medical handling of a radiation emergency. Such actions may include health related assessment of the accident, as well as sorting, decontamination, transportation, diagnosis, treatment and follow-up of the exposed persons. The ability of many Member States to cope with radiation accidents could be better strengthened by an international mechanism than by their isolated efforts. The reasons are the following:

(a) The diagnosis and treatment of radiation injuries must be planned for and undertaken in specialized centres having trained personnel and costly sophisticated techniques. Besides, the risk of radiation emergencies is low. Thus, the need to have such a centre on a national level in every country is hardly justified. Instead, it appears reasonable to establish a global network of a few specialized centres with international responsibilities.

(b) Population overexposure can also occur following transboundary radioactive releases.

(c) Scientific data on effects of overexposure at a national level are accumulated slowly due to the low frequency of radiation accidents. Pooling of these data could speed up the development of more effective techniques for diagnosis and treatment of radiation victims.
The Chernobyl accident gave strong support for this approach. The USSR was able to render medical assistance to affected persons at the reactor and around the reactor site by mobilizing its big material, scientific and health service resources. Had such an accident happened in a country which had no specialized institutions and expertise on radiation pathology, the impact of the accident would have been much greater without large scale outside assistance.

3. SPECIFIC RESPONSIBILITY OF WHO IN PROMOTING INTERNATIONAL CO-OPERATION IN RADIATION EMERGENCY PREPAREDNESS AND ASSISTANCE

Medical preparedness and assistance in radiation emergencies should be regarded as part of the overall system for radiation safety and protection. Much work in this field is being done by the IAEA. It plays a central role in the Convention on Assistance in the case of a Nuclear Accident or Radiological Emergency. Valuable information on the effects of overexposure is being accumulated and updated by UNSCEAR. Specific tasks of WHO within the family of UN Organizations are to address those problems which lie within the competence of health authorities and/or are directly relevant to the medical community in Member States. These problems include the preparedness of medical services for handling radiation emergencies and the practices in diagnosis, treatment and follow-up of overexposed persons.

For many years, WHO has been collecting and distributing information on cases of overexposure, and on techniques for its diagnosis and treatment. Meetings have been organized for exchange of information, co-ordination of efforts in this field and elaboration of recommendations for Member States. Publications on this subject have been disseminated among Member States. WHO has been stimulating training of personnel to deal with the medical handling of radiation victims. In carrying out these activities, WHO has been collaborating with IAEA, UNSCEAR, ICRP, ICRU and other international bodies as well as with many national institutions.

In May 1988 the 41st World Health Assembly took the decision to accede to the Assistance Convention. Hence, WHO can be regarded as one of those international intergovernmental organizations which may be directly called upon for assistance under the provisions of the Convention.

For the promotion of radiation emergency medical preparedness and for practical assistance to countries in the event of overexposure from any source of radiation, WHO has established six collaborating centres: in France (International Center for Radiopathology, Paris), in the USA (Centre for Radiation Emergency Assistance, Oak Ridge), in the USSR (Centre for Medical Radiation Pathology, Leningrad), in Australia (Centre for Radiation Protection and Radiation Emergency Medical Assistance, Melbourne), in Argentina (Centre for the Response to Ionizing Radiation Emergencies, Buenos Aires), and in Brazil (Centre for Radiation Protection and Medical Preparedness for Radiation Accidents, Rio de Janeiro).
These centres should serve as focal points for advice and training, assist in the establishment of medical emergency plans for large scale radiation accidents, initiate co-ordinated studies on human radiopathology and radiation epidemiology, and assist in the preparation of relevant documents, guidelines and meetings.

In the case of a radiation accident, the collaborating centres might provide a team for on-site urgent medical treatment, a survey team for rapid external radiation monitoring and/or contamination surveys with appropriate equipment, means for transportation of patients, facilities and staff for medical examination, treatment, and follow-up medical supervision. The experience and resources of the collaborating centres in France and the USA have already been used on several occasions for international help in radiation emergencies.

Efforts are under way at present to increase the number of WHO collaborating centres to about ten and to establish close ties among them so that an international network of WHO collaborating centres with sufficient geographical coverage would be set up. As a result some kind of worldwide radiation emergency medical service would be created. An important step for initiating the network was the first Co-ordination meeting of Existing and Prospective WHO Collaborating Centres on Radiation Emergency Medical Preparedness and Assistance (held in Le Vésinet/Southampton, 1987). The meeting was attended by representatives of three centres existing at that time (in France, the USA and the USSR), and representatives from Australia, Argentina and Brazil. They made the acquaintance of the hosting institute, the Central Service for Protection against Ionizing Radiation (Le Vésinet) which is part of the WHO Collaborating Centre for Radiopathology. A first plan of co-ordinated actions was outlined. The group suggested that countries without collaborating centres could be involved in the network through their ‘liaison institutions’, i.e. national points of contact with the appropriate collaborating centre(s) and/or with WHO, including its Regional Offices. A contribution is also expected from ‘support institutions’, i.e. such national institutions as could be activated or invited to assist in solving particular problems, especially in an emergency situation.

Personal contacts among the participants contributed to the spirit of cooperation which is so important in emergency situations. Such a spirit manifested itself during the radiation accident in Goiânia, Brazil, when American, Argentine, French and Soviet specialists took an active part in providing assistance together with their Brazilian colleagues.

The Second Co-ordination Meeting of Existing and Prospective Collaborating Centres (Oak Ridge, October 1988) continued the work started at the first meeting. Representatives from the Japanese WHO Collaborating Centre on Radiation Effects on Humans and from the All Union Centre for Radiation Medicine in Kiev (USSR) were also present.

The meeting emphasized the necessity to set up the Radiation Emergency Medical Preparedness and Assistance Network of WHO collaborating centres (REMPAN) in the near future. The participants made the acquaintance of the Radia-
tion Emergency Assistance Centre/Training Site which is the WHO Collaborating Centre for Radiation Emergency Assistance. They specified the most important points in developing further collaboration among them. A WHO plan of radiation emergency medical preparedness and assistance was outlined. The REMPAN would be a supporting mechanism for implementation of the plan in the event of an emergency.

The scope and basic machinery of medical assistance to be provided through WHO have been reflected in the Emergency Notification and Assistance Technical Operations Manual (ENATOM) developed by the IAEA as a guide for States party to the Convention.

4. RELEVANT ACTIVITIES IN WHO REGIONAL OFFICES

In addition to WHO’s global activity, similar work has also been pursued at WHO Regional Offices.

The problem of public health preparedness for nuclear accidents has been continuously given consideration by the Regional Office for Europe (EURO). In one of its latest publications on this subject, prepared jointly with WHO headquarters, 'Nuclear Power: Accidental Releases - Practical Guidance for Public Health Action' (WHO Regional Publication, European Series No. 21, Copenhagen, 1987), some aspects of medical assistance to affected persons were tackled. EURO was the WHO focal point in responding to the Chernobyl accident and, after the accident, it has continued its programme on better preparedness in Europe for any future nuclear accidents. In May 1987 in Vienna, EURO convened jointly with the IAEA an Expert Meeting on the methodology of follow-up studies. A meeting organized in November 1987 in Geneva significantly contributed to the improvement of harmonization of European preparedness for a nuclear accident. The problem of medical assistance was directly addressed at the Working Group on Public Health Preparedness for Nuclear Accidents in the Near-Field (Amsterdam, October 1988) held with the support of WHO headquarters.

The WHO Regional Office for the Americas convened with other co-sponsors an international conference on non-military radiation emergencies (Washington, D.C., November 1986).

The Regional Office for the Eastern Mediterranean, in co-operation with headquarters, held an intercountry seminar on radiation protection (Baghdad, October 1986) and supported the First Arab Seminar on Radiation Protection (Baghdad, June 1987). At both seminars, particular attention was given to medical preparedness for radiation emergencies.
5. **A SCENARIO OF MEDICAL ASSISTANCE FROM REMPAN THROUGH WHO**

(1) In the event of actual or suspected overexposure of persons, the authorities of the country affected can request medical assistance from WHO either through the IAEA or directly. The telex is the preferred mode of communication. Messages of all kinds should begin with the code word "EMERCON" repeated twice.

The request should, if possible, contain the following information:

- source of radiation emergency (reactor, radioactive source, X ray device, radioactive fallout, etc.);
- external irradiation only or radioactive contamination;
- number of persons thought to be affected;
- degree of radiation exposure (estimated doses, clinical signs, if any, etc.);
- accompanying injuries, if any (wounds, burns, chemical intoxication, etc.);
- type of assistance needed as best estimated by the country affected (diagnosis, therapy, decontamination, etc.);
- the scope of assistance needed as best estimated by the country affected (advice by correspondence, supply of medicaments and devices, consultants on site, a medical assistance team, transportation of victims to a specialized centre abroad, etc.);
- the possibility of bearing expenses, totally or partially;
- any other information which may help in assessment of the situation.

If the request comes directly to WHO it informs the IAEA about it to receive confirmation of the accident.

(2) In its immediate reply to the requesting country, WHO acknowledges the receipt of the request, gives notification of the type of assistance which will be sought by WHO and describes the collaborating centres which will be approached.

(3) WHO alerts the network of its collaborating centres in radiation emergency medical preparedness and assistance, transmits the message from the affected country and asks for assistance available. The collaborating centres acknowledge receipt of the inquiry from WHO within three hours.

(4) The collaborating centres advise on their availability to assist and specify a way of assistance as soon as possible but not later than within three days. In the meantime, conditions and particularities for providing assistance, including legal and financial matters, could be discussed by telephone or by other rapid means of communication between parties concerned. WHO could organize multinational teams on rendering medical assistance on site.
(5) WHO immediately advises the requesting country about the assistance available and its conditions. As soon as the requesting country decides to accept the offered assistance from a particular centre or centres, it officially notifies WHO and the collaborating centre(s) of its decision. The country also identifies its institution(s) responsible for receiving assistance.

(6) WHO informs all its collaborating centres about the outcome of the request. Thereafter, the requesting country and assisting centre(s) communicate directly, copies being mailed to WHO. WHO might help the parties by inviting experts not only from its collaborating centres but also from other institutions, if necessary.

Additional assistance may be organized from the same centre(s) or other collaborating centres as well as from national institutions supporting REMPAN.

(7) The requesting and assisting parties inform WHO about the termination of assistance.

(8) WHO might help the parties in resolving disputes between them during the implementation of assistance as well as after its termination.

6. CONCLUSIONS

(1) WHO has a particular responsibility for the medical aspects of radiation emergency preparedness and assistance; this responsibility has been enhanced since WHO acceded to the Assistance Convention.

(2) Medical preparedness and assistance are regarded as part of an overall system of radiation safety.

(3) To strengthen its ability to handle radiation emergencies, wherever they may happen, WHO is developing REMPAN - a network of its collaborating centres engaged in radiation emergency medical preparedness and assistance.

(4) REMPAN should be the main instrument for the implementation of the WHO plan for radiation emergency medical assistance in case of a nuclear accident.
THE INFLUENCE OF SEASONALITY ON ACCIDENT CONSEQUENCES AND EMERGENCY RESPONSE PLANNING

C. VIKTORSSON
Nuclear Energy Agency of the OECD,
Paris

G. BOERI
Comitato Nazionale per la ricerca
e per lo sviluppo dell'Energia Nucleare
e delle Energie Alternative,
Direzione Sicurezza Nucleare
e Protezione Sanitaria (ENEA/DISP),
Rome, Italy

Abstract

THE INFLUENCE OF SEASONALITY ON ACCIDENT CONSEQUENCES AND EMERGENCY RESPONSE PLANNING.

The impact of an accidental release of radioactivity to the environment can be strongly influenced by prevailing environmental conditions. Thus, potential variations in accident consequences caused by variable seasonal, meteorological or climatic conditions are of significance to the development and application of protective measures and emergency response plans. The Nuclear Energy Agency of the OECD organized a workshop on these matters in 1988. As a follow-up to this meeting a decision was taken to set up a small group of consultants to prepare some specific guidance addressing these particular aspects of accident consequences and emergency planning. This paper is a progress report of the ongoing work of the group.

1. INTRODUCTION

The extent of the consequences of an accidental release of radioactivity is strongly dependent upon a wide number of parameters. In particular, the characteristics of the source term, seasonal, climatic and meteorological conditions have a substantial influence on the physical factors involved in transport and deposition of airborne contaminants, and on the transfer and accumulation of radionuclides in terrestrial and aquatic ecosystems. These environmental conditions also have a significant influence on living habits and practices, and thus on potential radiological and economic impacts. Moreover, these conditions may affect the features and the impact of countermeasures which are adopted for the protection of the public in the event of an accidental release.
The Nuclear Energy Agency of the OECD organized a workshop in September 1988 to discuss such matters [1]. The workshop provided a review of the influence of such environmental conditions as season, climate and weather on the radiological consequences of an accident, and on the implication of these conditions for the implementation of mitigating measures. Knowledge of these influences on accident consequences exists in the scientific community but, as demonstrated by the response to the Chernobyl accident, does not appear to be fully recognized by decision makers or the public. Clearly, there is a need for transferring the results of scientific studies on seasonal, meteorological and climatic effects into actual emergency response planning processes.

For this reason a small group of experts (see Annex 1) was convened to prepare specific guidelines addressing this particular subject. This paper presents the progress report of this expert group.

2. DISPERSION, ENVIRONMENTAL CONTAMINATION AND EXPOSURE OF MAN

2.1. Source term

The report does not deal with problems related to accident source terms; however, a few remarks will be devoted to the source term since it determines directly the initial conditions for all subsequent transport and exposure processes.

The potential spectra and amounts of radionuclides released from commercial nuclear power reactors in accidental situations have been analysed in the framework of several reactor safety studies. From these, it can be concluded that:

— there are accidents in which mainly noble gases will be released, leading to a short external exposure from the passing cloud;
— in many release scenarios the radioisotopes of iodine and caesium are likely to determine most of the radiation exposure of the public, and this will mostly occur by external irradiation, by the ingestion of contaminated food products and to some extent by the inhalation of contaminated air;
— the less volatile radioisotopes of elements, such as strontium and plutonium, are not expected to be released to the environment in most accident scenarios.

Different source term characteristics have to be expected from radiological accidents other than reactor accidents, e.g. from accidents in reprocessing plants, during transport of radioactive materials and from contamination due to a re-entry of a satellite with radioactive material. The type of environmental contamination which might follow a radiological accident has a direct implication on the strategy for monitoring and countermeasures and therefore, all attempts should be made to determine the release characteristics as accurately as possible by on-site and off-site
measurements, as soon as possible following a release as well as by theoretical considerations.

Release figures may range over many orders of magnitude from insignificant amounts like those of the Three Mile Island accident to significant fractions of the core inventory such as experience from Chernobyl.

The duration of a release may vary from hours to days. This may result in various effects on the environment and the time for taking countermeasures may therefore also be different.

2.2. Atmospheric and aquatic transport

The actual weather conditions at the time of an accident are of prime influence on the transport of released nuclides. The parameters that govern the dispersion during different times of the year in many regions of the world change their values with the season. It is advisable, therefore, to take the full range of these parameters into account in emergency planning and the actual (and not averaged) values in consequent management decisions.

In general, under determined conditions, the radionuclide concentration in air and water, which determines the radiological consequences, can be expected to fall roughly in inverse proportion to the distance from the source. During stable diffusion conditions, however, the lateral spreading of a cloud can be small, which might lead to a rather high concentration in a small affected area at rather long distances, whereas during instable weather conditions the cloud dilutes quickly.

The range over which radionuclides can be transported in the atmosphere depends on their chemical and physical form, as well as on actual weather conditions such as wind fields and precipitation patterns during the transport. In general, larger (>10 μm diameter) aerosols are likely to settle by the gravitational force closer to the source (within a few tens of km) than smaller particles that can be transported over thousands of km. Precipitation can lead to a measurable removal of radionuclides from the atmosphere and may play a very important role in determining the ground contamination at given locations.

2.3. Deposition characteristics

Deposition can occur during dry or wet weather episodes, whereby drastically different consequences may result. During dry periods only a weak contamination of surfaces has to be expected from radionuclides bound to aerosols, whereas elemental gaseous iodine deposits rather efficiently. The deposition velocity is higher for large and rough obstacles such as trees and lower for very smooth surfaces such as streets, roofs, water surfaces, snow cover, etc. The situation of the urban environment is more specific, in that deposition levels will in general be lower than in an agricultural area because of smooth surfaces and more efficient removal processes.
It can therefore be expected that the overall external exposure would be lower in an urban environment than the exposure that can be experienced in an agricultural area.

During wet weather episodes, a higher contamination from aerosol bound radionuclides must be expected than during dry weather. Furthermore, there will be a larger spatial variation of this contamination, because of the generally inhomogeneous spatial pattern of precipitation. Wet deposition by rainfall is influenced by at least two capture processes, namely by 'rainout' when a raindrop is formed in a contaminated cloud and by 'washout' when a raindrop falls through contaminated air masses to ground. The first process normally leads to a 3–10 times higher radioactive contamination of rain water than the second.

Close (within a few km) to a source most of the radionuclides released at a low height may be expected still below a raining cloud, and thus, will be removed mainly through washout processes. At larger distances, however, intimate mixing of the contaminated air masses with a rain cloud must be anticipated, leading also to the more effective rainout processes. Therefore, the concentration of radionuclides in near-ground air need no longer be correlated to their concentration in rain water.

The efficiency of surface contamination by snowfall and in foggy weather conditions is still more uncertain than that by rain and dry deposition. It could be advisable to consider the aerosol deposition by snow as more effective than by rain. Moreover, in foggy conditions the deposition is expected to be more efficient than during dry conditions because of wet surfaces and higher aerosol concentrations in fog droplets than in the ambient air. Important new data in the field of deposition is becoming available after the Chernobyl accident [2].

2.4. Contamination of the environmental media and potential exposure of man

The atmospheric transport of radionuclides and subsequent deposition leads to a general contamination of the environment (vegetation, soil, urban surfaces, etc.) and subsequently also to contamination of food products used for human consumption. The degree and spatial distribution of the initial contamination depend strongly on the source term with its various parameters, the actual weather conditions at the time of release, the status of vegetation in the local agricultural year, etc.; the same is true for the space and time dependent contamination of food chains.

Seasonality has an influence on the physical factors that govern transport, diffusion and deposition of airborne material, or its dispersion, transfer and accumulation in the environment; it can also strongly influence the biosphere and all its relationships that in turn determine the levels of radioactive contamination and its transfer to man. Finally, seasonality and time of the year influence people’s habits and practices (living, agriculture, etc.). Geographical factors (especially latitude, altitude) also play an important role in determining characteristics of the environment and its climate.
An example of this influence is the change of the distribution in space and time of the deposited activity, due mainly to the transport of radionuclides by runoff, which varies with the season. Activity can be transferred to surface and underground waters; the importance of the flow rate of the rivers and of the exchanges between soil and underground waters, which also varies with the season, can affect the resulting water concentrations.

Therefore, no general rule should be drawn concerning the extent of environmental contamination following an accidental release, e.g. from the actual observations after the Three Mile Island or the Chernobyl accidents, because another accident can be expected to be significantly different with respect to source term, weather and time characteristics. When preparing for emergency planning, radiological models for the best estimation of the potential radiological consequences of severe accidents in nuclear facilities should be developed and adopted to local conditions.

Man is radiologically affected by the contamination of the environment through various exposure pathways but could also be affected by the consequences of an accident in the sphere of his social and economic activities, as he could be induced to alter some of his habits, or to renounce certain products or the use of his property, and as a result, commercial patterns could be disrupted.

Among the factors that determine the level of the radiation dose man might receive from a passing cloud of released radioactive material or from deposited material on the ground, there is the length of time an individual spends in the open air as compared to indoors, where lower radiation fields and air concentration values could result in reduced doses, depending strongly on the type of house and its ventilation rate. The time spent indoors and outdoors is determined by the living and working habits and practices of the individual concerned, and these are very likely to be influenced by the external weather conditions, the time of the day, the day of the week and the season of the year. It should also be recognized that the realization that a radioactive contamination has taken place may induce people to alter their habits (i.e. by not using certain facilities or deserting certain resorts, cancelling holiday reservations, etc.), so that, besides the effect on the individuals concerned, there may be an overall economic and social impact. This same applies to commercial activities, especially, but not only, the food industry. Established patterns can be significantly affected, with cancellation of orders, consumers refusing to accept certain products in the light of assumptions on their origins or nature, even when it is certified that no activity is to be found in the product.

The ingestion of contaminated foodstuffs can represent an important exposure pathway following the contamination of the environment. It is also the exposure pathway that is most strongly influenced in its final outcome by the time of the year when the release has occurred, both because of availability of particular products and of feeding patterns of animals. This is particularly true for the first year's dose: this can represent over 50% of release of radioactivity. Figure 1 shows the Chernobyl
consequences as an example of the contribution by the different exposure pathways, to the effective dose equivalent commitment [3].

Diets vary greatly from individual to individual, as well as on a community, regional, religious and national basis. The particular diet of an individual at a given time will, therefore, be a function of these variables, but generally also of the availability of certain foodstuffs, which is influenced by seasonality and trade patterns.

In general, the principal elements of a diet include cereals (in many forms); meat from various types of animals, e.g. beef, pork, lamb, poultry and other farmyard animals; fish; vegetables and fruit; dairy products and drinking water. Particular diets unbalanced as regards specific foodstuffs cannot be excluded for large population groups and these aspects should be considered when planning monitoring programmes and interventions and when evaluating the consequences of an accidental release of radioactivity. These types of diets could also be more strongly influenced by particular conditions, such as weather and seasonal factors.
2.5. Vegetation

The state of vegetation which retains the deposited activity depends on the season. In the case of plants, the initial contamination as well as the translocation of activity to other parts of the plant are both highly dependent on the stage of the plant's development; e.g. the presence or absence of leaves; the surface and the mass of the plant, etc.

The different stages of agricultural plants' development could thus influence the contamination of human's diet by:

— The initial contamination of the plant from dry and wet deposition processes, and
— The subsequent transport, the translocation, of activity to the edible parts of the plant.

Figure 2 shows as an example the development of spring wheat during its growing period from April to August in the south of the Federal Republic of Germany. Until the emergence of ears in the last part of June there is a continuous increase of the leaf area, which then decreases continuously until the harvest due to the leaves dying off.

![FIG. 2. Leaf area index of spring wheat during the growing period and its stages of development [4].](image)
According to the development of the leaf area the amount of activity which is initially deposited on the plants in the south of the Federal Republic of Germany reaches its maximum if the activity release takes place at the end of June, or in the northernmost countries in July to August. The dry deposition rate onto the plants and the interception of wet deposited activity can be approximated by a linear relationship with the leaf area. Outside the vegetation period no direct contamination of the spring wheat plants occurs. Besides spring cereals, however, seasonal effects can also be connected with contamination of wheat and rye sown in autumn, if deposition occurs outside the vegetation period. Contamination of these cereal plants can be either direct or indirect, via snow and its melting waters.

The second relevant mechanism, i.e. the translocation of activity from foliage to grain is illustrated in Figure 3. It shows the percentage of caesium activity which is translocated to the grain depending on the time of the plant’s contamination. In the beginning of the growing season this fraction is rather small. It increases markedly during the period of tillage and reaches a maximum during the period of corn filling. Until harvest there is a decrease due to a decrease in the metabolic activity of the wheat plant (in the last 30 days before harvesting).

The seasonal dependency of the accident’s consequences is different for the various plants and is closely related to the vegetational period and time of harvest.
of each plant species. The vegetational periods, in turn, are strongly dependent on the climatic conditions, especially temperature, insolation and precipitation. Moreover, totally different growing conditions due to the different climates are to be found on an international scale, for example, at any time of the year wheat is harvested somewhere in the world [1]. It is evident that the growing season and the time of harvest will significantly influence the dose to the public via ingestion of contaminated food products and the duration over which countermeasures have to be enforced. This in addition may vary for different food products.

In the long term, vegetation can become contaminated by uptake of radionuclides by the roots from the soil. The principal parameters governing this process are plant species, nature of the soil and its chemical composition and the type of radionuclide.

Agricultural practices, such as ploughing, irrigating, fertilizing, etc., all influence to a considerable extent the mechanisms by which deposited radioactivity is transferred with time to the crops and to man by the food pathway route. Agricultural practices with regard to sowing, harvesting and the types of crops depend on national or regional traditions, many of which are derived from natural constraints or inclinations and are sometimes related to seasonal and climatic characteristics.

2.6. Animals

Many species of animals, especially game and fish, depend for their food on what the season offers. Therefore, it can be expected that the levels of radioactive

![Graph](image-url)
contamination, following a nuclear accident, which will be found in animal tissues and products as an effect of food transfer will be strongly influenced by the season and the climate of a particular area.

Furthermore, many animals, during the first stages of life, show peculiar tendencies and feeding patterns, sometimes completely different from those followed as adult organisms. They can also choose different habitats that could be contaminated to a different degree as compared to the general environment. Normally these life stages are strictly correlated with the season.

In the case of animals, weather conditions have a much less important influence on the direct radiological consequences of an accident, although they can determine the level of contamination of an area. In the case of aquatic life, intense precipitation may lead to high levels of contamination due to radioactivity transported by runoff water. In this case, it is likely that a seasonal effect could be detected related to the level of biological activity in the aquatic environment; in a season when this activity may be particularly lively, e.g. during spring, the radioactive particles captured by aquatic microorganisms may be increased substantially and an increased feeding and filtrating activity by superior organisms may ultimately result in a substantial accumulation of radioactivity in their tissues.

In the case of lakes with a reduced turnover (e.g. due to climatological reasons or to the absence of an outlet) the effects of a contamination can also be found for a rather long time, particularly in carnivorous species. Such an example is shown in Fig. 4.

3. MODELLING AND MONITORING

The choice of appropriate protective measures should be based, among other factors, on the best possible appreciation of the actual situation following the accident.

3.1. The role of predictive models

Models have been developed for predicting the movement of radionuclides in many parts of the environment, including the atmosphere, the soil, water bodies and food chains. In many cases the characteristics of the release and detailed data on the weather conditions are not known with certainty during the early stages of an accident. It is therefore not possible to rely on model predictions alone in order to determine the radiological situation (and, in particular, levels of deposition) and to make
decisions regarding countermeasures except in cases where, for example, the potential hazard is great that a large release will take place, and the model prediction together with judgements on plant conditions indicate that precautionary evacuation is needed.

By developing and applying predictive models, scientists can learn about the possible ranges and types of environmental contamination which could be encountered in different weather conditions and at different times of year. They can also determine which exposure pathways will be important in different situations, and provide advice on how to adapt monitoring programmes and countermeasure decisions for these different weather conditions and seasons.

Another way in which predictive models can be used in emergency planning, is their use as one input for establishing intervention levels. By investigating the potential consequences of accidents in many different weather conditions and seasons, and combining these results with political and practical judgements, ranges of likely intervention levels can be determined in advance, in order to avoid the pressures pertaining at the time of an accident.

Predictive models can also be used as an input to monitoring and countermeasure decisions following an accident. In the short term, model results can be used to help allocate monitoring and countermeasure priorities. For example, a predictive atmospheric dispersion model which included information on rain systems could indicate possible areas of high deposition due to rain, particularly those where the population were most likely to be at risk, either from external exposure or food consumption.

Predictive models can be useful in assessing, in the short term, potential doses to members of the public. For the longer term following an accident, models are very useful in predicting the likely variation of radiation exposure with time, particularly when the release has stopped and deposition levels are fairly well known. If decisions are to be made on implementing long term countermeasures, such as food interdiction or relocation, or on lifting countermeasures, it is important for decision makers to have an assessment of likely future exposures. Predictive models would assist in such situations.

3.2. Monitoring

The aim of a monitoring programme should be to identify the most critical situations and the most sensitive areas so that efforts can be concentrated on areas where radiological problems could appear and where it is necessary to implement the most appropriate protective measures. The efficiency of such a programme depends essentially on timely implementation and the choice of the measurement types and locations.
The monitoring strategy is defined both on the basis of accidental conditions and on the characteristics of the concerned area. The monitoring programme is also time dependent and must be adjusted to the evolution of the situation. However, certain general principles have to be considered, among which are the following:

1. The first objective is to localize the most contaminated area; hence the monitoring programme must be linked to local meteorological data indicating plume movements. When the contamination pattern in general is known, the monitoring has to be focused on the areas where the contamination levels are significant. The most appropriate measurements refer to concentration of radioactive substances in air, dose rate in air and the magnitude and composition of radioactive substances deposited on the ground. The use of mobile equipment, possibly embarked on helicopter or aircraft, and fixed networks are generally most helpful in meeting this objective.

2. The second objective is to assess the exposure of the public. Hence measurements of the contamination of foodstuffs and drinking water, and if necessary of animal feeds, have to be carried out in addition to the above mentioned measurements.

3. The programme has to take into consideration the characteristics of the area, distinguishing between urban areas and rural areas, the latter with their specific activities (production of milk, meat, vegetables, cereals, etc.).

4. The monitoring programme also depends on environmental conditions, especially the season. For instance, if the accident occurs during winter, when the animals are indoors and fed with uncontaminated stored fodder, it will not be necessary from a radiological point of view to establish an extensive programme for the control of milk. On the other hand, if the animals were outdoors, grazing at the time of the accident, milk and dairy produce would constitute a major pathway to man and have to be controlled, in particular if iodine and caesium have been released.

5. Another important point to be considered as regards priorities in the monitoring programme is the timing of the countermeasures. For instance, the transfer pathway arising from the contamination of leafy vegetables by direct deposition, or from the contamination of milk by iodine, is very fast and, if a protective measure is justified, it must be implemented quickly in order to be efficient. On the other hand, contamination of cereals does not normally require rapid interventions.

6. Monitoring should be carried out also to provide a general reassurance to the public, e.g. as a basis for decisions concerning non-affected areas and that countermeasures are not needed in these and other areas.
4. EMERGENCY PLANNING AND COUNTERMEASURES

4.1. General

If a nuclear accident involving radioactive releases occurs, the exposure of the public can be reduced by the implementation of protective measures. The protective measures that could be undertaken are numerous and will depend on the type and severity of the accident and its consequences. The purpose of any measures implemented should be to minimize the detriment or consequences resulting from the accident. The applicability of each protective measure, its efficiency, the associated costs, the social disruption and risks, depend on many factors and among them, the weather conditions and the season of the year.

Some of them such as evacuation, sheltering, and administration of stable iodine, aim at avoiding or reducing direct exposure of the public, especially during the first phase of the accident. Others are related to the contamination of the environment and the transfer of radioactivity through the food chain and their applicability and efficiency are in some cases directly linked to the season. Such countermeasures are decontamination of surfaces and buildings, control on agriculture, animal husbandry and other human practices and control of foodstuffs.

Concerning the contamination of foodstuffs, there are two categories of countermeasures: (1) those avoiding or reducing the contamination of the products themselves, and (2) those avoiding or reducing their transfer to individuals. Both actions aim at reducing internal exposure due to ingestion.

The reduction of the contamination of crops (vegetables, fruit, etc.) is normally relevant only on a small scale, i.e. by washing and peeling. Conversely, in the case of animal production (milk, meat, etc.), cattle can be maintained or returned indoors and fed with stored fodder, protected, at least partially, against contamination by deposition. The efficiency of this protective measure depends on the season and the availability of stored fodder or ensilage. The administration of chemical compounds to animals in order to reduce the uptake of ingested radionuclides might also be considered. These techniques are still in development and their efficiency and usefulness depend on many factors, such as the type of compound administered, the availability and cost of the compounds, and the practicability of administering them.

The reduction of the transfer to individuals of contaminated foodstuffs can be achieved in different ways. The most effective is the ban on consumption of some foodstuffs, which might reduce the intake of radionuclides significantly but raises many social and economic problems, such as the loss of the products and the necessity to replace them. Justification of these protective measures depends largely, in addition to the level of contamination, on the quantities involved.

When the concern is for the contamination from short-lived radionuclides, an important dose reduction can also be obtained by simply delaying the consumption of the contaminated food.
Foodstuffs that cannot be consumed directly can be processed to decrease the contamination levels; for instance, transforming of milk into cheese or butter can reduce the level of contamination significantly. The feasibility of such actions as well as of alternative uses of foodstuffs such as for animal feeding needs to be checked carefully and should be considered in the emergency response planning.

Finally, it must be remembered that techniques which are normally used when preparing dishes, such as mechanical actions (peeling, cutting off, etc.), washing and cooking also result in a reduction of the contamination levels. Their efficiency can be weak or important according to the nature of the contamination and the type of product. Studies are under way in many countries to find out useful methods and techniques in this field.

Decontamination of surfaces and buildings has often been invoked in the past as a countermeasure for reducing external exposure, inhalation and contamination risks. It has been applied to limited portions of buildings and surfaces, and extensively in the USSR after the Chernobyl accident.

In northern countries removal of snow can, in some cases, be used as a measure to reduce external dose rate and contamination of soil. Experiments on the efficiency of this method have been carried out [5].

Finally, some agricultural techniques can be employed for reducing the availability and the transfer of contaminants deposited onto the soil. Among these are: (a) the decortication of the upper layer of soil, if it is feasible and the transfer of contamination cannot be avoided in another way; (b) the ploughing of soil in order to reduce external radiation and to dilute the contamination and perhaps make it less available for resuspension, runoff and uptake by plants and animals; and (c) the use of fertilizers and other chemical agents that tend to compete with radionuclides in the uptake by plants and by animals.

4.2. Applicability of countermeasures

Weather and seasonal conditions can influence the applicability of countermeasures, for example, the nature of the deposition that has occurred will influence what decontamination processes are likely to be effective. Radionuclides deposited on a wall in very heavy rainfall for example, are unlikely to be removed subsequently by hosing. Equally, different agricultural activities at different times of the year will influence whether interdiction of various food types would be effective in saving dose. For example, if the grain had been harvested and stored in a relatively airtight building just prior to the accident, there would be no need to ban grain consumption. However, it might be necessary to consider whether or when the next crop should be planted. Again, there would be no need to consider immediate banning of milk or meat if the animals were consuming uncontaminated stored food, although consideration would be given to how long, beyond normal practice, the animals should be kept off their usual pastures. From Fig. 5 it is evident that in many instances
Beginning of renunciation (days after deposition)

Duration of renunciation (days): 60, 30, 14, 7

Dose reduction (%)

FIG. 5. Effect of renunciation of feeding contaminated grass to dairy cows on the reduction of dose via milk consumption for deposition of $^{137}$Cs on 1 September [1].

countermeasures, if implemented in time, can save doses even when applied only for a shorter period of time.

Some seasonal implications however may be less self evident. For example, in a tourist area, the resident population may live in solidly constructed houses, whilst many tourists stay in caravans or tents. In this case, sheltering during times when few tourists were present could be an effective countermeasure, whilst during the tourist season it would not. Similarly, the problems of evacuating a large tourist population, particularly in remote areas with poor road and rail networks, could be significant, and, if sufficient solidly constructed buildings were available, could result in higher exposures than sheltering. It is important that such considerations are built into the emergency plan.

4.3. Levels of intervention

The second way in which the season could affect the implementation of countermeasures is by influencing the considerations for deriving the values of the intervention levels, both for implementing and for lifting them. The intervention criteria reflect a balancing of all the detrimental effects of not taking the countermeasure against all the detrimental effects of taking it. Often the judgements made in obtaining the balance are not explicit, but they usually include factors such as direct health risk both from the radiation exposure and from the countermeasure itself, economic
cost directly and indirectly attributable to the countermeasure and of treating any induced health injuries, social factors such as the likely reaction of the population, disruption, etc., and also implications for international relations and trade.

Since seasonal effects can strongly influence the ease with which certain countermeasures can be carried out it may well be that the balance point between the adverse effects of taking and not taking the decision to evacuate is judged to be at a higher level of dose or radiation risk at different times. For example, if an area is a tourist area at certain times of the year, the number of people affected by a decision to evacuate that area may easily vary between a few hundred and a few thousands, or tens of thousands. Clearly these extremes place very different pressures on the available resources. In particular, there may be severe problems in communicating advice clearly to everyone (including language difficulties), and in enabling everyone to leave the area quickly. The physical risks and social upheaval caused by the evacuation may well be higher when many tourists are present, and certainly the resources required to implement the evacuation would be significantly larger.

It is therefore important that decision makers are aware of the fact that intervention levels are set by balancing the detrimental effects of taking and not taking the countermeasure. They do not represent a transition from safe to unsafe levels, and so flexibility in the level actually used is both possible and necessary.

Finally, in order to ensure that those with responsibility for radiological protection following an accident are properly trained to carry out their role in all seasons and weather conditions, it is important that emergency exercises are planned and carried out regularly, and that the exercise scenarios incorporate a range of seasonal considerations and realistic weather patterns.

5. CONCLUSION

The ongoing review by the Expert Group shows that such factors as weather and seasonality strongly influence the consequences of a radiological accident. Atmospheric transport of released material and environmental transfer processes of the contamination can vary by orders of magnitude at specific locations and the full range of these variations should be taken into account at the planning and the intervention stages.

Furthermore, weather and seasonal factors can influence the applicability of planned countermeasures and therefore could require different intervention levels in different situations.
MEMBERS OF THE EXPERT GROUP

G. Boeri  
ENEA, Italy

A.A. Cigna  
(Chairman)  
ENEA, Italy

R. Coulon  
Commissariat à l’énergie atomique (CEA), France

M.E. Morrey  
National Radiological Protection Board,  
United Kingdom

H.G. Paretzke  
Gesellschaft für Strahlen- und Umweltforschung  
(GSF), Federal Republic of Germany

C. Viktorsson  
Nuclear Energy Agency of the OECD, Paris

REFERENCES


In the Panel which concluded the symposium, chairmen of the individual sessions summed up and commented on major points. The following summarizes their statements and comments.

Session I: RECOVERY OPERATIONS FROM ACCIDENTS WITH RADIOACTIVE MATERIALS

Session I was dedicated to recovery operations from accidents with radioactive materials not related to nuclear power plants. The medical and scientific applications of ionizing radiation require, as do the energy applications, a prudent regulatory framework which should be thoroughly implemented. The strict implementation of radiation protection standards in the medical and industrial field is just as important as in any other field. It is very necessary that the competent authorities in radiation protection avoid the unnecessary proliferation of radioactive sources and installations and also establish complete inventories of such sources and installations.

The best possible level of protection can be obtained only if, on the one hand, clear regulations exist with a sound scientific basis which are applied strictly and, on the other hand, if international co-operation efforts are maintained and where necessary extended. A tremendous gap exists between the technical, scientific and regulatory community and the media. It is necessary to make a major effort to bridge this gap.
Enhanced training of intervention personnel remains an important priority for the competent authorities and the industry. Close co-operation is always required between radioecologists, experts in dosimetry and radiobiologists (i.e. an interface between routine monitoring and research). Information for the public about any potential hazards of ionizing radiation should be an essential component of good radiation protection policies and practices. Training and teaching covering radiation and radiation protection should be an integral part of educational programmes even at primary and secondary school.

Session II: ON-SITE RECOVERY OPERATIONS (NUCLEAR FACILITIES)

This was perhaps the first time that the subject of on-site post accident management at both Chernobyl and Three Mile Island (TMI 2) has been presented in a public meeting by managers of day-to-day activities during these cleanups, thus representing the root of the experiences at the two sites. There are order of magnitude differences between the consequences of the two accidents, but some points of similarity in the management approach developed can be extracted: three years after the Chernobyl accident, from a management viewpoint, the situation there is not so different from that of Three Mile Island when it was three years into its ten year cleanup. The situation has been stabilized both as regards thermal and recriticality potential. The units are separated from their sister units. Today Chernobyl is gathering data and evaluating how to proceed from this intermediate status, a situation that is very similar to the status of TMI 2 in 1982–1983. Stability has been accomplished with a minimum dose to workers for achieving this isolation. At every new phase of the programme there have been surprises and obstacles which required new thinking and a development of innovative techniques. The decision was taken at TMI 2 to maintain and monitor the isolation of Unit 2, after fuel has been removed, for an extended period of time, allowing the residual radiation to decay until both units can be decommissioned. Chernobyl 4 is now facing a similar decision making process.

Both these accidents have provided innovative techniques and devices in methods of radiation measurement and personnel protection. These should be considered for development. Many of these techniques can be applied in situations of lesser magnitude. However, extreme caution was urged when discussing the development of regulations. It must be re-emphasized that the major lesson at TMI 2, repeated many times throughout this symposium, was that on-site cleanup activities are not standard in any way. They require innovative thinking at each step. For this reason one must be very careful regarding prescriptive pre-specified regulations or safety guides that may be perceived as regulatory in nature. The clear challenge today is how to capture the useful technical experience to make sure that it is available for future use.
Part 2 of this session made a contribution to another problem, the question of organizing, planning and supply for a recovery operation both on-site and off-site. Planning and organizing the reaction to an emergency has now become a primary focus of interest.

Papers dealt with basic planning and implementation of measures on cleanup and on decontamination; how services in various countries do this work on a very large scale, seeking to mitigate consequences; and what is done on a State level.

The on-site recovery work outlined in one paper (IAEA-SM-316/45) underlined a very interesting problem that was met. The question of planning, organizing and supplying the emergency services is being addressed in many countries and is rather acute. Chernobyl as one of the biggest accidents has led to much technological development. But it is concluded that not everything possible in that area has been done: questions of planning, financing and setting up supply routes and, particularly, establishing complex remote handling systems have not yet been answered satisfactorily. Chernobyl suggested that one has to use a lot of very complex robotic equipment. Today the approach is changing slightly. It is considered that the robots used in Chernobyl are not necessary for today's emergency services because what one is looking at is the setting up of complex systems which allow one quickly to monitor the radiation situation, to do reconnaissance work and carry out the necessary technical operations. All this was very well outlined in two papers (IAEA-SM-316/36 and 50) stating that the methods must be selected in line with the specific conditions that a country encounters in its recovery work.

As regards the dose received during recovery, radiation exposure is a very serious matter regardless of where and how it is received and deserves serious attention. Aspects to be studied and correlated from the experience of all countries include documentation on the organization of emergency services currently in force, manuals on the organization and monitoring of the radiation situation, the creation of new equipment and the supply to emergency teams.

**Session III: OFF-SITE RECOVERY OPERATIONS (NUCLEAR FACILITIES)**

Session III comprised several interesting subjects. One paper (IAEA-SM-316/42) described ultraviolet luminescence measurements for the detection of highly active contamination spots, new fast radiochemical and scintillation methods for the detection of plutonium and strontium and decontamination of heavily contaminated forest areas. More than 400 m³ of trees were entombed with a reduction factor of gamma radiation levels by 35-40, in some cases up to 1000.

Two papers (IAEA-SM-316/4 and 48) described the utilization of dose assessment models for estimation of population doses after a nuclear accident where radioactivity is released to the environment. Combining results from models with
radiological measurements can give emergency response managers valuable information on future dose distributions in the exposed populations and they can thus estimate the effects of different countermeasures for reducing these doses. In the future one will probably see an increasing use of such models.

One special focus was the huge amount of information from the Chernobyl, the TMI 2 and the Goiânia accidents. Although not reported in greater detail during the symposium, a lot of quantitative data regarding the off-site recovery operations must be available, such as data on decontamination efficiency for both urban and rural areas, and monetary and manpower resources devoted to these large scale recovery operations. It was considered highly desirable that some sort of reference document be prepared in which all these detailed data are collected and evaluated.

A critical question was posed: "Can dose limits for introduction of dose reducing measures after a nuclear accident not in many cases result in more harm than good and therefore be unjustified on pure radiation protection grounds?" The very important message that came out from this symposium is that one should be very careful in elaborating radiation protection recommendations. Justification for actions should be pursued first and then, once the action is justified, optimization of the levels at which one takes the action. If that approach is not recommended, the answer to the provocative question posed above unfortunately will be: "Yes, one can produce more harm than good."

Session III also included a paper (AIEA-SM-316/60) relating to some rather interesting and surprising findings about the current status of the remnants of the Chernobyl reactor core, a reminder that events happen which are unforeseeable in the light of current knowledge.

Another paper (IAEA-SM-316/56) gave a fascinating description of the human side of the recovery operations at Chernobyl; i.e. how the decisions were made and actions taken quite quickly to protect public health and safety although there was little pre-planning for many of these actions and the decisions that were necessary were unique. It was concluded that no major errors were committed in doing what was necessary for the public.

Two papers (IAEA-SM-316/6 and 19) discussed in detailed technical terms the impact of the Chernobyl accident in other countries and again showed the significant impact such accidents can have on neighbouring countries. Two further papers (IAEA-SM-316/25 and 26) demonstrated the serious and concerted planning that is going on in France so as to be ready to evaluate the impacts and protect the public after any possible radiological accident. The audience was impressed by the vehicles that have been developed for monitoring members of the public.

It might be useful to formalize the provision of assistance so as to speed up its availability in the case of an accident. However, what is the balance between pre-planning and acquiring specialized equipment on the one hand and depending on normal capabilities and the energy of professionals and volunteers without pre-planning on the other? Obviously, some pre-planning is needed. Many of the pre-planning
tasks described in this symposium are not costly. Others, such as the impressive rail car for monitoring the public and the aerial monitoring capabilities being developed are very costly and few countries can afford such a capability. What might a reasonable national programme cost? How much should be pre-planned in the sense of not wasting a lot of effort and cost? A philosophy needs to be developed directed towards a reasonable recovery planning programme.

No paper in Session III addressed the problem of warning and rapid assessment. There are IAEA Conventions on Early Notification of a Nuclear Accident and on Assistance in the Case of a Nuclear Accident or Radiological Emergency, but at what level should they be triggered? It requires real time monitoring devices linked to a computer network and a computer model. More development is needed in this area.

Concerning cleanup, two papers addressed the urban environment problem, a strategy paper (IAEA-SM-316/44) and an experimental paper (IAEA-SM-316/29). These described only part of the efforts that are going on in these areas and, therefore, they were illustrative rather than definitive. A further paper (IAEA-SM-316/28) dealt with part of the agricultural contamination problem. But, with a minor exception, there was not any real discussion of priorities. There was only a single paper on cost effectiveness (IAEA-SM-316/48) though there are many areas where the cost effectiveness problem needs to be addressed. In the event of a major accident there simply will not be the resources to carry out decontamination to the extent that was done after the Goiânia accident. To decontaminate down to background levels may be considered to be a luxury.

There was relatively little discussion of logistics. How does one get hold of all these mechanical devices needed in an accident? Where does one get the supplies of clothing? There were many examples of somewhat unexpected needs following an accident. It did not appear that much attention has yet been given to the logistics of actually supplying those in an accident. It was reiterated that the effectiveness of any in-country, bilateral or multilateral co-operation in the event of an emergency depends on the country’s infrastructure.

One can do some planning in advance for the very early stage, but for the long term one has to take the decisions on a case by case basis, justifying them and optimizing the level of intervention.

There were two papers (IAEA-SM-316/37 and 12) on intervention levels. One gave a very clear exposition of the rationale for the levels in the USSR and the levels that were actually applied after the Chernobyl accident. The other presented an analysis of the interventional level picture as far as the rest of the world was concerned and an estimate of the overall radiological consequences of the Chernobyl accident.

There are intervention levels which are predicated by radiological considerations and there are intervention levels which are predicated by political considerations. It must be made absolutely clear in which category any particular intervention
level lies. Radiological ones should be based on ICRP concepts but “one should get unanimity about the actual values of the levels” and then state “that is the level which is required to assure radiological safety”. Any further reduction in those levels should be quite clearly put at the door of the politicians. One should neither attempt to enter that field or to disguise the fact that a lot of these intervention levels are political and they are based on public perception of risk rather than real risk.

It was very clear from the Goiânia event that a medical intervention plan is needed. Many countries are rather backward in that area. Public information and media plans are also needed. It is too late once an accident has taken place; it is not sufficient just to have a pressroom with 20 telephones or whatever. One has to start years in advance of the accident, one has to establish relationships with the press and television in particular. One should know the names and make contact with the news editors on major television channels. They like to have lists of people they can contact when there is an event of importance. Everything should be done to help them, remembering that after a major accident it is much better to help reporters provide the kind of copy that one would like to see than to leave them to use their imagination.

Session IV: LESSONS LEARNED FROM RADIOLOGICAL ACCIDENTS

Concern about radioactive wastes, especially organic wastes, was evident, especially as regards the post-accident phase. This problem does not seem to have been mastered fully on a world level yet — and this is surprising. Another element which needs to be looked into more in depth relates to the post-accident strategy as regards the dose level for returning to normal conditions.

Concerning the operational level, a great amount of hardware and software is already well established but getting it to the spot in a post-accident situation or even during the accident itself if something which is not a matter of course in every country. Indeed, no country can claim that it has perfect control of all the logistics.

One has to understand the human element, the human dimension of the post-accident situation. “We have to know what is the state of mind of the victims, the workers, the public, what their mental thought processes are during the actual accident, so that we can take care of this, deal with it and take it into account in whatever measures are adopted.”

Work is needed on matters which do not seem to have been completely settled yet: how one can reuse the soils, how one has to deal with livestock and so on, but one should also take into account the restoring work to revitalize and rehabilitate contaminated zones; one has to take into account economic, financial and social factors which are always important, and this is perhaps where one should introduce economists and agronomists in the nuclear sphere.
One should attempt, for example under the auspices of the IAEA, to produce something like a manual, or something more ambitious than a manual, which would be available in each State to provide the necessary information for those who have to intervene very quickly.

Co-operation is something which should be specially developed, removing the nuclear field from the ghetto into which public opinion would like to put it.

In Part 2 of the session, one paper (IAEA-SM-316/35) outlined the objectives, elements and problems of the post-accident planning process being developed and one (IAEA-SM-312/31) described some of the new equipment, mainly robotic, which is being developed to help in the implementation of such programmes. Another paper (IAEA-SM-316/43) described an interesting and cost effective mobile emergency response service with high capability to assist in an emergency, the key feature of this service being that it has a small core of permanent staff plus a large pool of outside experts who can be called on very quickly to provide rapid deployment of equipment and staff.

Other papers (IAEA-SM-316/22 and 14) illustrated the importance and the value of good documentation after an accident. The Goiânia accident showed that the country learned from the accident and has since developed a much improved approach to radiological emergency response planning and implementation.

Paper IAEA-SM-316/54 showed once again the sad stories which can result and certainly will happen again until every country has a good regulatory regime and a licensing body which can keep track of radioactive sources. The existence of many hundreds or thousands of radiation therapy units around the world suggests that such a tragedy will most likely occur again.

Where does one go from here? The first decision the Member States must make is whether they are going to address the problem of a large accident or whether they are going to assume that it will not happen. Certainly in some countries the problem is being addressed directly. Other countries are backing away from it and either not addressing it or only addressing certain areas.

Some of the topics which need to be developed to give the recovery teams the tools to assess and to carry out remedial actions and to assist Member States to make it easier to do such planning are as follows: first, more work is required to develop the computer programs which can be used to do analysis on the various factors which influence the recovery criteria for re-entry so that the recovery team can decide what is the best strategy. The programme should include all the factors which are affected by the selected level of the recovery criteria. It is certain that the more stringent the recovery criterion is, the lower will be the health risk but the higher will be the decontamination costs and other property related costs. The re-entry criteria will be set on the basis of what the country can afford: this will mainly be an economic decision.

It is practically impossible to review all the technical aspects of recovery operations, to look at the research going on and to look at the planning which is being
done. The audience interested in detailed radiation protection papers differs from that interested in detailed technical papers. Certainly there is a wide scope for much more detailed discussion on technical subjects, for example, waste volume reduction. Management methods are available for the safe disposal of large volumes of waste. The overall operational plans which are available in some countries need to be compared. Possibly a technical handbook needs to be prepared which might include some of the basic elements that are needed by people who are doing cleanup in order that they may do it safely, efficiently and at reasonable cost. It is of limited value to have detailed and sophisticated radiation protection criteria and guidance if the cleanup is not done properly and if the technology is not there.

One should not forget the accident at the 40 MW NRX reactor in 1952 where a large core meltdown occurred. Important features of this accident were the methods that were used to rehabilitate the reactor and bring it back into operation in 1956, and what happened to the wastes. One million cubic metres of highly contaminated water were dumped into a clump of sand. This has migrated since 1952, and has been modelled and examined. There is a great deal to learn from this on the retention of radionuclides even in such things as sand.

In part 3 of the session was very encouraging to see that OECD/NEA has attacked a problem which is very important with respect to seasonal, meteorological and climatic conditions which can affect the development or application of protective measures and emergency response plans (IAEA-SM-316/49). The OECD/NEA report should be published very soon. The importance of better intervention criteria was also pointed out.

The World Health Organization (WHO) is developing a network of its collaborating centres engaged in radiation emergency medical preparedness and assistance (IAEA-SM-316/49). This is good news, and should be pointed out to the media.

Another paper (IAEA-SM-316/8) pointed out the need for credible data, and that a single individual with appropriate authority should be in command in the event of an emergency. Some kind of consensus actions regarding intervention levels should be developed. There is still need for more standard collection and analytical procedures. Guides in that direction could be very useful, especially with respect to the cleanup of contamination.

One should remember that the public does not understand what radioactivity means. An international effort towards an adequate system of information to prevent public panic is essential. Serious accidents have happened and from these accidents one can learn. This information should be available to the public also in a form that it can really understand — not only the technical facts but also actual experiences.
**Special Session: THE ACCIDENT IN THE SOUTHERN URALS**

The special session was fully devoted to reports of Soviet authors on the accidental releases of large activities of fission products following an explosion of one of several high radioactive waste storage tanks of a plutonium extraction plant in the Southern Urals. This was the very first presentation to those who were personally involved in handling the accident consequences. They presented details of the release itself, dissemination, deposition, the evaluation of the radiation situation, the estimate of dose that characterized health studies as well as information on approaches elaborated for the reuse of contaminated land. Many inputs were provided for future programmes, such as data on transfer of radionuclides in different soils, and different reuse of land for the production of agricultural items. Further information on the mechanisms and conditions of the accident itself, on the meteorological conditions leading to the creation of a very narrow radioactive path and on the psychological reactions of the population at the site would be desirable. The event occurred 32 years ago, i.e. in the early phase of the history of emergency response planning. However, the information on the varying conditions for taking decisions on countermeasures and performing them should not only be of historical interest. Very important was the fact that there was no case of early radiation health effects amongst the population and that up to now a very thorough follow-up has not revealed any indication of elevated frequency of those health effects which could be connected with the exposure of the public due to this accident.

The Kyshtym event again underlined the importance of public information. Only transparency can assure a progressive improvement of emergency planning and create public confidence in measures planned to be implemented in the event of an accident.

The symposium programme did not include any paper on the Windscale accident in the United Kingdom in 1957 but the audience was informed that a comprehensive historical report on this accident was to be published in March 1990.
CHAIRMEN OF SESSIONS

Session I  R. CIBRIAN  Commission of the European Communities
           C.A. NEGIN  United States of America
           V.I. ZABRODIN  Union of Soviet Socialist Republics

Session II  P. HEDEMANN JENSEN  Denmark
            B.H. WEISS  IAEA
            A.E. J. EGGLETON  United Kingdom

Special Session  E. KUNZ  Czechoslovakia

Session IV  Y. MOURÈS  France
           M.A. FERADAY  Canada
           S. CHAKRABORTY  Switzerland

Panel  L.B. SZTANYIK  Hungary

SECRETARIAT OF THE SYMPOSIUM

Scientific Secretaries:  F.N. FLAKUS  Division of Nuclear Safety,
                       F. WENSLAWSKI  IAEA, Vienna
                       E. ASCULAI

Conference Organizer:  H. PROSSER  Division of External Relations,
                       IAEA, Vienna

Proceedings Editor:  R. PENISTON-BIRD  Division of Publications,
                     IAEA, Vienna

French Editor:  J.-N. AQUISTAPACE

Spanish Editor:  L. HERRERO

Russian Editor:  O. MELNIK
LIST OF PARTICIPANTS

Agatiello, O.E.  
Central Nuclear Atucha,  
Casilla Correo 20, 806 Lima, Buenos Aires, Argentina

Al-Hasawi, A.A.M.  
Civil Defence,  
Ministry of the Interior,  
P.O. Box 6504, Riyadh-11452, Saudi Arabia

Ammann, R.  
Section Matériel et technique d'engagement,  
Corps suisse d'aide en cas de catastrophe,  
Eigerstrasse 71, CH-3003 Berne, Switzerland

Andersson, K.G.  
Health Physics Dept,  
Risø National Laboratory,  
P.O. Box 49, DK-4000 Roskilde, Denmark

Arapis, G.  
Instituto PRYMA,  
Centro de Investigaciones Energéticas,  
Medioambientales y Tecnológicas,  
Avenida Complutense 22, E-28040 Madrid, Spain

Bacher, P.  
(UNIPEDE)  
UNIPEDE,  
39, avenue de Friedland,  
F-75008 Paris, France

Baggenstos, M.  
Division principale pour les installation nucléaires,  
HSK,  
CH-5303 Würenlingen, Switzerland

Barber, P.Y.  
Centre d'études nucléaires,  
60-64, avenue de la Division Leclerc, B.P. 6,  
F-92265 Fontenay-aux-Roses, France

Becker, K.  
(ISO)  
ISO/TC 85 Nuclear Energy,  
DIN,  
Burggrafenstrasse 6, Postfach 1107,  
D-1000 Berlin 30

Belovodskij, L.F.  
State Committee on the Utilization of Atomic Energy,  
Staromonetnyj pereulok 26, Moscow Zh 180, Union of Soviet Socialist Republics

637
LIST OF PARTICIPANTS

Bertini, A.  
Divisone della Produzione e Trasmissione,  
ENEL,  
Via G.B. Martini 3, I-00196 Rome, Italy

Blaser, V.  
Section Matériel et technique d’engagement,  
Corps suisse d’aide en cas de catastrophe,  
Eigerstrasse 71, CH-3003 Berne, Switzerland

Borovoj, A.A.I.V.  
Kurchatov Institute of Atomic Energy,  
ul. Kurchatova 46, 123182 Moscow D-182,  
Union of Soviet Socialist Republics

Borras, C.  
(PAHO/WHO)  
Pan American Health Organization,  
525 23rd St. N.W.,  
Washington, DC 20007, United States of America

Breznik, B.  
Nuclear Power Plant Krsko,  
Vrbina 12, YU-68270 Krsko, Yugoslavia

Brudermüller, G.  
Kerntechnische Hilfsdienst GmbH,  
Am Schröcker Tor 1, D-7514  
Eggenstein-Leopoldshafen,  
Federal Republic of Germany

Buldakov, L.A.  
Institute of Biophysics,  
Ministry of Health,  
ul. Zhivopisnaya 46, 123182 Moscow,  
Union of Soviet Socialist Republics

Bull, A.O.  
University of Oslo,  
Blindern, P.O.Box 1060, N-0316 Oslo 3, Norway

Burton, O.  
Unité de radioécologie,  
Avenue de la Faculté 8, B-5800 Gembloux,  
Belgium

Carboneras, P.  
PTO de Seguridad y Licenciamiento,  
Paseo de la Castellana 135, E-28046 Madrid, Spain

Carfi, N.  
CISE spa,  
Via Reggio Emilia 39,  
I-20090 Segrate Milan, Italy

Carvalho, A.B.  
Comissäo Nacional de Energia Nuclear,  
Rua General Severiano 90, Botafogo,  
22294 Rio de Janeiro, RJ, Brazil
LIST OF PARTICIPANTS

Chakraborty, S. Swiss Federal Nuclear Safety Inspectorate, CH-5303 Würenlingen, Switzerland

Champ, D.R. Chalk River Nuclear Laboratories, Chalk River, Ontario, K0J 1J0, Canada

Cheng, K.M. Radiation Health Unit, Department of Health, Sunning Plaza, 4/F Hysan Avenue, Causeway Bay, Hong Kong

Chéreau, A. Centre d'études du Ripault, B.P. 16, F-37260 Monts, France

Chevalier, C. Service général de médecine du travail, 30, avenue de Wagram, F-75382 Paris Cedex 08, France

Cibrian, R. Direction générale de l'environnement, de la sécurité nucléaire et de la protection civile, Bâtiment Jean Monnet, Plateau du Kirchberg, L-2920 Luxembourg, Luxembourg

(CEC) Collée, R.F. Université de Liège, Rue Ponson 19, B-4500 Liège, Belgium

Cortella, J. Centre de Valduc, F-21120 Is-sur-Tille, France

Croft, J.R. National Radiological Protection Board, Northern Centre, Hospital Lane, Cookridge, Leeds LS16 6RW, United Kingdom

Damjanovich, I. Government Nuclear Emergency Commission, P.O. Box 25, H-1885 Budapest, Hungary

de Ville de Goyet, C. Pan American Health Organization, 525 23rd St. N.W., Washington, DC 20037, United States of America

(PAHO/WHO) De Waal, S.W.P. P.P.O. Box 69301, Bryanston 2021, South Africa

de Witt, H. Brenk Systemplanung, Heinrichsallee 38, D-5100 Aachen, Federal Republic of Germany
LIST OF PARTICIPANTS

Delaire, L. Compagnie générale des matières nucléaires, B.P. 1515, 41, rue Barthélemy Thimonier, F-87020 Limoges, France

Dickerson, M.H. Lawrence Livermore National Laboratory, P.O. Box 808, L-262, 7000 East Avenue, Livermore, CA 94550, United States of America

Dinnie, K.S. Nuclear Studies and Safety Dept, Ontario Hydro, 700 University Avenue, Toronto, Ontario M5G 1X6, Canada

Dodig, D. Department of Nuclear Medicine, University Hospital Rebro, Kispaticeva 12, YU-41000 Zagreb, Yugoslavia

Dolgin, N.N. Office of the Director of Civil Defence Organization for the Protection of the Population, ul. Vatutina 1, 121357 Moscow, Union of Soviet Socialist Republics

Eggleton, A.E.J. Harwell Laboratory, United Kingdom Atomic Energy Authority, Building 551, Didcot, Oxfordshire OX11 0RA, United Kingdom

Fache, P. Centre d'études nucléaires de Cadarache, F-13118 Saint-Paul-lez-Durance, France

Feraday, M.A. Apartment 207, 2045 Lakeshore Boulevard W., Toronto, Ontario, M8V 2Z6, Canada

Foster, K.T. Lawrence Livermore National Laboratory, P.O. Box 808, L-262, 7000 East Avenue, Livermore, CA 94550, United States of America

Fowels, G.A. Ontario Hydro, 700 University Avenue, Toronto, Ontario M5G 1X6, Canada

Garcia-Bermejo, R. División Protección Radiológica y Medio Ambiente, Piquer 7, E-28033 Madrid, Spain

Genesco, M. Ministère de l'Intérieur, 1, place Beauvau, F-75800 Paris, France
LIST OF PARTICIPANTS

Ginot, P. 
Centre d'études nucléaires de Fontenay-aux-Roses, 
B.P. 6, F-92265 Fontenay-aux-Roses, France

Gonen, Y.G. 
Swiss Federal Nuclear Safety Inspectorate, 
CH-5303 Würenlingen, Switzerland

Gritti, R. 
(CEC) 
Centre Commun de Recherche, 
Ispra, Varese, Italy

Gros, R. 
Centre d'études de Bruyères-le-Châtel, 
B.P. 12, F-91680 Bruyères-le-Châtel, France

Guimbail, H.-C. 
Direction générale, 
Electricité de France, 
32, rue de Monceau, F-75008 Paris, France

Gustavsson, J.S.V. 
Swedish State Power Board, 
S-162 887 Vällingby, Sweden

Heureux, M. 
Ministère de l'Intérieur, 
Protection civile, 
Avenue des Arts 27, B-1040 Brussels, Belgium

Hildebrand, J.E. 
GPU Nuclear Corporation, 
One Upper Pond Road, 
Parsippany, NJ 07054, United States of America

Holton, W.C. 
43 Brunswick Road, Troy, NY 12180, 
United States of America

Honegger, P.J. 
Bundesamt für Gesundheitswesen, 
Nationale Alarmzentrale, 
Ackermannstrasse 26, CH-8044 Zurich, Switzerland

Hultqvist, G. 
Swedish State Power Board, 
S-742 00 Osthammer, Sweden

Ikezawa, Y. 
JAERI, 
Tokai Research Establishment, 
Tokai-mura, Naka-gun, 
Ibaraki-ken 319-11, Japan

Jalal, M.M.S. 
Directorate of Civil Defence, 
Riyadh-11174, Saudi Arabia
<table>
<thead>
<tr>
<th>Name</th>
<th>Organization and Address</th>
</tr>
</thead>
<tbody>
<tr>
<td>Jonkman, R.</td>
<td>Ministry of Internal Affairs, P.O. Box 20011, NL-2500 EA The Hague, Netherlands</td>
</tr>
<tr>
<td>Kalman, C.J.</td>
<td>Ministry of Defence, Room 1837, Empress State Building, Lillie Road, London SW6 1TR, United Kingdom</td>
</tr>
<tr>
<td>Karacson, P.</td>
<td>Ämt der Niederösterreichischen Landesregierung, Herrengasse 9-13, A-1014 Vienna, Austria</td>
</tr>
<tr>
<td>Kaspar, D.</td>
<td>Ministerium für Umwelt, Postfach 10 34 39, Kernerplatz 9, D-7000 Stuttgart 1, Federal Republic of Germany</td>
</tr>
<tr>
<td>Katchalovskij, E.V.</td>
<td>Council of Ministers of the Ukrainian SSR, ul. Kirova 12, Kiev, Ukrainian Soviet Socialist Republic</td>
</tr>
<tr>
<td>Kayser, P.</td>
<td>Division de la radioprotection, Ministère de la Santé, Luxembourg</td>
</tr>
<tr>
<td>Kenneke, A.</td>
<td>Division of Nuclear Safety, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria</td>
</tr>
<tr>
<td>Khairallah, A.</td>
<td>Centre d'études nucléaires de Fontenay-aux-Roses, B.P. 6, F-92265 Fontenay-aux-Roses, France</td>
</tr>
<tr>
<td>Khan, F.A.</td>
<td>Nuclear Safety and Radiation Protection Division, Bangladesh Atomic Energy Commission, P.O. Box 158, Ramna, Dhaka-1000, Bangladesh</td>
</tr>
<tr>
<td>Kholina, Yu.B.</td>
<td>State Committee on the Utilization of Atomic Energy, Staromonetnyj pereulok 26, Moscow Zh 180, Union of Soviet Socialist Republics</td>
</tr>
<tr>
<td>King, M.A.</td>
<td>Safety and Reliability Directorate, Wigshaw Lane, Culcheth, Warrington, Cheshire WA3 4NE, United Kingdom</td>
</tr>
</tbody>
</table>
LIST OF PARTICIPANTS

Klug, N.P.  
US Department of Energy,  
Mail Stop NE-42,  
Washington, DC 20545, United States of America

Kochetkov, O.A.  
Biophysics Institute,  
Ministry of Health,  
ul. Zhivopisnaya 46, 123182 Moscow,  
Union of Soviet Socialist Republics

Korun, M.A.  
Jožef Štefan Institute,  
Jamova 39,  
P.O. Box 100, YU-61000 Ljubljana, Yugoslavia

Kowalczyk, J.  
213, Rio Brano,  
Los Alamos, NM 87544, United States of America

Krajewski, P.  
Department of Radiation Hygiene,  
Central Laboratory for Radiological Protection,  
ul. Konwalinowa 7, PL-03-194 Warsaw, Poland

Kunz, E.  
Institute of Hygiene and Epidemiology,  
Srobárova 48, CS-100 42 Prague 10,  
Czechoslovakia

L’Homme, A.  
Centre d’études nucléaires de Fontenay-aux-Roses,  
B.P. 6, F-92265 Fontenay-aux-Roses, France

Lambotte, J.M.  
Service de protection contre les radiations ionisantes,  
Ministère de la santé publique et de l’environnement,  
Cité administrative de l’Etat,  
Quartier Vésale N2-324, B-1010 Brussels, Belgium

Larsen, B.I.  
Directorate of Health,  
P.O. Box 8128 DEP, N-0032 Oslo 1, Norway

Lee, J.H.  
S-Cubed,  
Division of Maxwell Laboratories,  
2501 Yale Boulevard SE, Suite 300,  
Albuquerque, NM 87109, United States of America

Legrand, B.  
Centre d’études nucléaires de Cadarache,  
F-13118 Saint-Paul-lez-Durance, France
LIST OF PARTICIPANTS

Litai, D. Israel Atomic Energy Commission, P.O. Box 7061, Tel Aviv 61070, Israel

Lossik, A. Embassy of the USSR, Erzherzog Karl-Strasse 182, A-1220 Vienna, Austria

Manesse, D. Centre d’études nucléaires de Fontenay-aux-Roses, B.P. 6, F-92265 Fontenay-aux-Roses, France

Marchant, C.P. Royal Naval College, Greenwich, London SE10 9NN, United Kingdom

Marcus, F.R. Nordic Liaison Committee for Atomic Energy NKA, P.O. Box 49, DK-4000 Roskilde, Denmark

Martinčič, R. Jožef Štefan Institute, Jamova 39, P.O. Box 100, YU-61000 Ljubljana, Yugoslavia

Mattsson, C.-G.H. Ringhals Nuclear Power Plant, Swedish State Power Board, S-430 22 Vaeröebacka, Sweden


Méthivier, H. Centre d’études nucléaires de Fontenay-aux-Roses, B.P. 6, F-92265 Fontenay-aux-Roses, France

Miaw, S.T.W. Centro de Desenvolvimento da Tecnologia Nuclear, Comissão Nacional de Energia Nuclear, Cidade Universitária-Pampulha, Caixa Postal 1941, 30000 Belo Horizonte, Minas Gerais, Brazil


Mollah, A.S. Institute of Nuclear Science and Technology, Atomic Energy Research Establishment, P.O. Box 3787, Dhaka-1000, Bangladesh
LIST OF PARTICIPANTS

Moroni, J.P. Service central de protection contre les rayonnements ionisants, Ministère de la Santé, B.P. 35, F-78110 Le Vésinet, France

Morrey, M.E. National Radiological Protection Board, Chilton, Didcot, Oxfordshire OX11 0RQ, United Kingdom

Mourès, Y. Secrétariat général du Comité interministériel de la sécurité nucléaire, 54, rue de Varenne, F-75007 Paris, France

Murase, M. Energy Research Laboratory, Hitachi Ltd, 1168 Moriyama-cho, Hitachi-shi, Ibaraki-ken 316, Japan

Negin, C.A. Grove Engineering, Inc., 15215 Shady Grove Road, Suite 202, Rockville, MD 20850, United States of America

Neumann, W. Kerntechnische Hilfsdienst GmbH, Am Schröcker Tor 1, D-7514 Eggenstein-Leopoldshafen, Federal Republic of Germany

Nierlich, J. Centre d’études nucléaires de Fontenay-aux-Roses, B.P. 6, F-92265 Fontenay-aux-Roses, France

Nikipelov, B.V. State Committee on the Utilization of Atomic Energy, Staromonetnyj pereulok 26, Moscow Zh 180, Union of Soviet Socialist Republics

Noc, B. Service de la production thermique, Electricité de France, Cedex 57, F-92060 Paris—La Défense, France

Nogueira de Oliveira, C.A. Instituto de Radioproteção e Dosimetria, Comissão Nacional de Energia Nuclear, Avenida das Américas, km 11.5, Barra da Tijuca, CEP 22602, Caixa Postal 37750, Rio de Janeiro, RJ, Brazil
<table>
<thead>
<tr>
<th>Participant</th>
<th>Organization</th>
<th>Address</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nowell, R.L.</td>
<td>Chalk River Nuclear Laboratories, Chalk River, Ontario K0J 1P0, Canada</td>
<td></td>
</tr>
<tr>
<td>O'Gabhlain, M.</td>
<td>Civil Defence School, Phoenix Park, IR-Dublin 8, Ireland</td>
<td></td>
</tr>
<tr>
<td>O'Reilly, C.</td>
<td>27 Millview Lawns, O'Falihive, Dublin, Ireland</td>
<td></td>
</tr>
<tr>
<td>Olasinde, T.A.</td>
<td>University Clinic for Radiotherapy and Radiobiology, Vienna University, Alser Strasse 4, A-1090 Vienna, Austria</td>
<td></td>
</tr>
<tr>
<td>Palacios, E.</td>
<td>Comisión Nacional de Energía Atómica, Avenida Libertador 8250, 1429 Buenos Aires, Argentina</td>
<td></td>
</tr>
<tr>
<td>Panfilov, A.P.</td>
<td>State Committee on the Utilization of Atomic Energy, Staromonetnyj pereulok 26, Moscow Zh-180, Union of Soviet Socialist Republics</td>
<td></td>
</tr>
<tr>
<td>Paretzke, H.G.</td>
<td>GSF Institut für Strahlenschutz, Ingolstädter Landstrasse 1, D-8042 Neuherberg, Federal Republic of Germany</td>
<td></td>
</tr>
<tr>
<td>Perminov, S.P.</td>
<td>State Committee on the Utilization of Atomic Energy, Staromonetnyj pereulok 26, Moscow Zh-180, Union of Soviet Socialist Republics</td>
<td></td>
</tr>
<tr>
<td>Piéra, G.-J.</td>
<td>Secrétariat général du Comité interministériel de la sécurité nucléaire, 54, rue de Varenne, F-75007 Paris, France</td>
<td></td>
</tr>
<tr>
<td>Popović, S.</td>
<td>Department of Nuclear Medicine, University Hospital Rebro, Kispaticeva 12, YU-41000 Zagreb, Yugoslavia</td>
<td></td>
</tr>
<tr>
<td>Pucelj, B.</td>
<td>Jožef Štefan Institute, Jamova 39, P.O. Box 100, YU-61000 Ljubljana, Yugoslavia</td>
<td></td>
</tr>
</tbody>
</table>
LIST OF PARTICIPANTS

Quevedo García, J.R.  
Centro de Protección e Higiene  
de las Radiaciones de la Secretaría  
Ejecutiva para Asuntos Nucleares,  
Calle 18A No. 4110 et 41 y 43,  
Miramar, C. Havana, Cuba

Rebiffé, J.  
Cabinet du Haut-Commissaire,  
Commissariat à l’énergie atomique,  
31-33, rue de la Fédération,  
F-75752 Paris Cedex 15, France

Riaboukhine, I. (WHO)  
World Health Organization,  
CH-1211 Geneva 27, Switzerland

Richardson, M.R.  
British Nuclear Fuels Ltd (BNFL),  
Sellafield, Seascale, Cumbria CA20 1PG,  
United Kingdom

Robinson, E.R.  
Department of Nuclear Science and Technology,  
Royal Naval College,  
Greenwich, London SE10 9NN, United Kingdom

Romanov, G.N.  
State Committee on the Utilization  
of Atomic Energy,  
Staromonetnyj pereulok 26, Moscow Zh-180,  
Union of Soviet Socialist Republics

Ronkainen, S.T.  
Miscellaneous Services Security,  
Helsinki University Central Hospital,  
Stenbäckink. 9, SF-00290 Helsinki, Finland

Rozental, J.J.  
Comissão Nacional de Energia Nuclear,  
Rua General Severiano 90, Botafogo,  
22294 Rio de Janeiro, RJ, Brazil

Ryder, H.P.  
Civil Defence and Emergency Planning Agency,  
Vordingbørggade 18, DK-2100 Copenhagen,  
Denmark

Salava, J.  
Ministry of Health and Social Affairs,  
Tr. W. Piecka, 120 37 Prague 10, Czechoslovakia

Saleh, F.M.  
Iraqi Atomic Energy Commission,  
P.O. Box 765, Tuwaitha, Baghdad, Iraq
LIST OF PARTICIPANTS

Schaffhauser, M.  Secrétariat général du Comité interministériel de la sécurité nucléaire, 54, rue de Varenne, F-75007 Paris, France

Schnadt, H.  TÜV Rheinland, Institute for Accident Research and Ergonomics, Postfach 101750, D-5000 Cologne, Federal Republic of Germany

Schober, H.-Ch.  Niedersächsisches Umweltministerium, Archivstrasse 2, Postfach 4107, D-3000 Hannover 1, Federal Republic of Germany

Seaborn, R.M.  Sellafield Works B403/5, Seascale, Cumbria CA20 IPG, United Kingdom

Secchi, S.  Via Arnobo 14, Rome, Italy

Séguy, J.  9, allée Cuvier, F-94420 Le Plessis-Treise, France

Severo, A.J. da Conceição  Laboratório Nacional de Engenharia e Tecnologia Industrial, Departamento de Proteção e Segurança Radiológica, E.N. 10, P-2685 Sacavém, Portugal

Shramchenko, A.D.  Scientific and Technical Committee on Civil Defence of the USSR, ul. Vatutina 1, 121357 Moscow, Union of Soviet Socialist Republics

Sigurbjöernsson, B.  Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture, International Atomic Energy Agency, P.O. Box 100, A-1400 Vienna, Austria

Sinkko, K.T.S.  Finnish Centre for Radiation and Nuclear Safety, P.O. Box 268, SF-00101 Helsinki, Finland

Sinnaeve, J.  Direction générale de la science, de la recherche et du développement, Rue de la Loi 200, B-1049 Brussels, Belgium

Soyberk, Ö.A.  Çekmece Nuclear Research and Training Centre, Airport P.O. Box 1, Istanbul, Turkey
Spoonley, B. Nuclear Installations Inspectorate, 
St. Peter’s House, Stanley Precinct, Bootle, 
Merseyside, United Kingdom

Stillman, D.B. Los Alamos National Laboratory, 
P.O. Box 503, Los Alamos, NM 87545, 
United States of America

Strassner, G. Abteilung SOE, 
Industrieanlagen-Betriebsgesellschaft mbH, 
Einsteinstrasse 20, D-8012 Ottobrunn, 
Federal Republic of Germany

Strauss, H. Staatliches Amt für Atomsicherheit und Strahlenschutz, 
Waldowallee 117, DDR-1157 Berlin, 
German Democratic Republic

Šujak, P. Ministry of Health and Social Affairs, 
Ul. Ceskoslovenskey Armady 6, 
CS-813 05 Bratislava, Czechoslovakia

Sulgin, Yu.I. Scientific and Technical Committee on Civil Defence of the USSR, 
ul. Vatutina 1, 121357 Moscow, 
Union of Soviet Socialist Republics

Susanna, A. ENEA-DISP, 
Via Vitaliano Brancati 48, 
I-00144 Rome, Italy

Sztanyik, L.B. Frédéric Joliot-Curie National Research Institute of Radiobiology and Radiohygiene, 
Pentz K. u.5, H-1221 Budapest, Hungary

Tiemessen, G. Ministry of Internal Affairs, 
P.O. Box 20011, NL-2500 EA The Hague, 
Netherlands

Travers, W.D. US Nuclear Regulatory Commission, 
Washington, DC 20555, United States of America

Tschurlovits, M. Atominstitut der Österreichischen Universitäten, 
Schüttelstrasse 115, A-1020 Vienna, Austria
Van Beek, P.W.  
Regional Fire Brigade,  
van de Speigelstraat 100,  
NL-4381 VC Vlissingen, Netherlands

Van Lith, D.  
National Institute of Public  
Health and Environmental Protection,  
P.O. Box 1, NL-3720 BA Bilthoven, Netherlands

Vesseron, P.  
Centre d'études nucléaires de Fontenay-aux-Roses,  
B.P. 6, F-92265 Fontenay-aux-Roses, France

Viktorsson, C.  
(OECD/NEA)  
Nuclear Energy Agency of the OECD,  
38, boulevard Suchet, F-75016 Paris, France

Vinhas, L.A.  
Instituto de Pesquisas Energeticas e Nucleares,  
Comissão Nacional de Energia Nuclear,  
Caixa Postal 11049, 01000 São Paulo, SP, Brazil

Vinther, F.H.  
Risø National Laboratory,  
P.O. Box 49, DK-4000 Roskilde, Denmark

Vishnevskij, I.N.  
Nuclear Research Institute of Kiev,  
Academy of the Sciences  
of Ukrainian Soviet Socialist Republic, Kiev,  
Ukrainian Soviet Socialist Republic

Vladar, M.  
Research Institute of Preventive Medicine,  
Limbova 14, CS-833 01 Bratislava, Czechoslovakia

Vlatkovic, M.  
Department of Nuclear Medicine,  
University Hospital Rebro,  
Kispaticeva 12, YU-41000 Zagreb, Yugoslavia

Walford, J.G.  
United Kingdom Atomic Energy Authority,  
Dounreay, Thurso, Caithness KW14 7TZ,  
Scotland, United Kingdom

Wallin, M.W.  
Swedish National Institute of Radiation Protection,  
P.O.Box 60204, S-10401 Stockholm, Sweden

Walmod-Larsen, O.  
Risø National Laboratory,  
P.O. Box 49, DK-4000 Roskilde, Denmark

Weiss, B.H.  
Office for Analysis and Evaluation  
of Operational Data,  
Washington, DC 20555, United States of America
LIST OF PARTICIPANTS

Williams, B.G.  Nuclear Fuel Cycle Department,
National Power Division,
Central Electricity Generating Board,
Sudbury House, 15 Newgate Street,
London ECIA 7AU, United Kingdom

Williams, B.J.  Nuclear Installations Inspectorate,
St.Peter's House, Stanley Precinct, Bootle,
Merseyside, United Kingdom

Yamane, N.  Toshiba Nuclear Research Laboratory,
4-1 Ukishima-cho, Kawasaki-ku,
Kawasaki City 210, Japan

Zabrodin, V.I.  Ministry of Atomic Energy,
Kitajskij pr. 7, 103074 Moscow K-74,
Union of Soviet Socialist Republics

Zuniga-Bello, P.  Division of Nuclear Safety,
International Atomic Energy Agency,
P.O. Box 100, A-1400 Vienna, Austria
AUTHOR INDEX

Numerals refer to the first page of a paper by the author concerned

Agatiello, O.E.: 167
Ali, M.A.T.: 57
Andersson, K.G.: 217
Andreev, I.I.: 105
Avetisov, G.M.: 305
Awal, K.O.: 57
Barabanova, A.V.: 305
Belovodskij, L.F.: 87, 105, 125, 135
Belyaev, I.A.: 125, 135
Belyaev, S.T.: 185, 229
Boeri, G.: 607
Bolotov, Yu.A.: 105
Borovoj, A.A.: 185, 229
Bourgeois, C.: 347
Brenk, H.D.: 355
Brown, R.M.: 23
Brudermüller, G.: 541
Bruno, H.: 301
Buldakov, L.A.: 305, 373, 419
Buzulukov, Yu.P.: 185
Camus, H.: 507
Carvalho, A.B.: 463
Chakraborty, S.: 441
Champ, D.R.: 23
Chastel, R.: 347
Cooper, E.L.: 23
Cornett, R.J.: 23
Cortella, J.: 347
Couture, J.: 559
Croft, J.R.: 575
de Almeida, C.E.: 3, 593
de Witt, H.: 355
Demin, S.N.: 419
Dickerson, M.H.: 203
Dobrynin, Yu.L.: 185
Dolgin, N.N.: 185
Fache, P.: 495, 507
Fedorov, E.A.: 433
Feraday, M.A.: 283
Foster, K.T.: 203
Franz, W.A.: 479
Frenkler, K.: 355
Gaevoj, V.K.: 105
Gagarinskij, A.Yu.: 185, 229
Gauthier, D.: 507
Genesco, M.: 567
Goldammer, W.: 355
Gonen, Y.G.: 441
Gonzalez, A.J.: 313
Grishmanovskij, V.I.: 87, 105
Gros, R.: 365
Guzella, M.F.R.: 49
Hamoniaux, M.: 507
Hildebrand, J.E.: 65
Hille, R.: 355
Holton, W.C.: 525
Hultqvist, G.: 145
Hunt, J.: 553
Jacobs, H.: 355
Kachalovskij, E.V.: 235
Karim, S.M.F.: 57
Kenneke, A.: 575
Khairallah, A.: 559
Khan, F.A.: 57
Kholina, Yu.B.: 373, 433
Klug, N.P.: 479
Kochetkov, O.A.: 151, 305
Koshurnikova, N.A.: 419
AUTHOR INDEX

Kostyuchenko, V.A.: 419
Krajewski, P.: 257
Krestinina, L.Yu.: 419
Kryuchkov, V.P.: 151
Kunst, J.J.: 301
Kyt’kov, V.A.: 151
L’Homme, A.: 495
Laborne, J.J.: 593
Lambert, R.W.: 525
Lapa, L.G.: 151
Le Drean, C.: 365
Lebedev, L.A.: 125, 135
Legrand, B.: 495, 507
Mannan, M.A.: 57
McGoff, D.J.: 479
Mendonça, A.H.: 3
Miaw, S.T.W.: 49
Mikheenko, E.I.: 373
Mikheenko, S.G.: 125, 135
Molina, G.: 517
Mourès, Y.: 249
Negin, C.A.: 525
Neumann, W.: 541
Nikipelov, B.V.: 373
Noc, B.: 559
Nogueira de Oliveira, C.A.: 553
Osanov, D.P.: 151, 305
Owen, D.E.: 479
Palacios, E.: 301
Panfilov, A.P.: 87, 105
Parmentier, N.: 495
Popov, V.I.: 151
Rab Molla, M.A.: 57
Récio, J.C.A.: 553
Reis, L.C.A.: 49
Riaboukhine, I.: 599
Romanov, G.N.: 373, 405, 433
Rosenberg, A.: 347
Roux, L.: 365
Rozental, J.J.: 3, 593
Sachett, I.A.: 553
Santos, P.O.: 49
Saurov, M.M.: 419
Shramchenko, A.D.: 185
Shvedov, V.L.: 419
Silva, E.M.P.: 49
Soyberk, Ö.A.: 273
Spirin, D.A.: 373
Stroganov, A.A.: 125, 135
Sul’din, Yu.I.: 185
Tello, C.C.O.: 49
Ternovskij, I.A.: 419, 433
Teverovskij, E.N.: 433
Tokarskaya, Z.B.: 419
Travers, W.D.: 79
Vesseron, P.: 559
Viktorsson, C.: 607
Vinhas, L.A.: 39
Weiss, B.H.: 585
Zech, G.G.: 585
Zuniga-Bello, P.: 575
## TRANSLITERATION INDEX

<table>
<thead>
<tr>
<th>Russian Name</th>
<th>Transliteration</th>
</tr>
</thead>
<tbody>
<tr>
<td>Андреев, И.И.</td>
<td>Andreev, I.I.</td>
</tr>
<tr>
<td>Беляев, И.А.</td>
<td>Belyaev, I.A.</td>
</tr>
<tr>
<td>Беляев, С.Т.</td>
<td>Balyaev, S.T.</td>
</tr>
<tr>
<td>Беловодский, Л.Ф.</td>
<td>Belovodskij, L.F.</td>
</tr>
<tr>
<td>Боровой, А.А.</td>
<td>Borovoj, A.A.</td>
</tr>
<tr>
<td>Болотов, Ю.А.</td>
<td>Bolotov, Yu.A.</td>
</tr>
<tr>
<td>Булулуков, Ю.П.</td>
<td>Buzulukov, Yu.P.</td>
</tr>
<tr>
<td>Бuldakov, Л.А.</td>
<td>Buldakov, L.A.</td>
</tr>
<tr>
<td>Гаевой, В.К.</td>
<td>Gaevoj, V.K.</td>
</tr>
<tr>
<td>Гагаринский, А.Ю.</td>
<td>Gagarinskij, A.Yu.</td>
</tr>
<tr>
<td>Гришмановский, В.И.</td>
<td>Grishmanovskij, V.I.</td>
</tr>
<tr>
<td>Добрынин, Ю.Л.</td>
<td>Dobrynin, Yu.L.</td>
</tr>
<tr>
<td>Долгин, Н.Н.</td>
<td>Dolgin, N.N.</td>
</tr>
<tr>
<td>Качаловский, Е.В.</td>
<td>Kachalovskij, E.V.</td>
</tr>
<tr>
<td>Кочетков, О.А.</td>
<td>Kochetkov, O.A.</td>
</tr>
<tr>
<td>Крючков, В.П.</td>
<td>Kryuchkov, V.P.</td>
</tr>
<tr>
<td>Кутьков, В.А.</td>
<td>Kut'kov, V.A.</td>
</tr>
<tr>
<td>Лапа, Л.Г.</td>
<td>Lapa, L.G.</td>
</tr>
<tr>
<td>Лебедев, Л.А.</td>
<td>Lebedev, L.A.</td>
</tr>
<tr>
<td>Мikerin, Е.И.</td>
<td>Mikerin, E.I.</td>
</tr>
<tr>
<td>Михеенко, С.Г.</td>
<td>Mikheenko, S.G.</td>
</tr>
<tr>
<td>Никипелов, Б.В.</td>
<td>Nikipelov, B.V.</td>
</tr>
<tr>
<td>Осанов, Д.П.</td>
<td>Osanov, D.P.</td>
</tr>
<tr>
<td>Панфилов, А.П.</td>
<td>Panfilov, A.P.</td>
</tr>
<tr>
<td>Попов, В.И.</td>
<td>Popov, V.I.</td>
</tr>
<tr>
<td>Романов, Г.Н.</td>
<td>Romanov, G.N.</td>
</tr>
<tr>
<td>Спирин, Д.А.</td>
<td>Spirin, D.A.</td>
</tr>
<tr>
<td>Строганов, А.А.</td>
<td>Strogonov, A.A.</td>
</tr>
<tr>
<td>Сул'гин, Ю.И.</td>
<td>Sul'gin, Yu.I.</td>
</tr>
<tr>
<td>Холина, Ю.Б.</td>
<td>Kholina, Yu.B.</td>
</tr>
<tr>
<td>Шрамченко, А.Д.</td>
<td>Shramchenko, A.D.</td>
</tr>
<tr>
<td>IAEA-SM-316/</td>
<td>Page</td>
</tr>
<tr>
<td>-------------</td>
<td>------</td>
</tr>
<tr>
<td>1</td>
<td>479</td>
</tr>
<tr>
<td>3</td>
<td>57</td>
</tr>
<tr>
<td>4</td>
<td>203</td>
</tr>
<tr>
<td>6</td>
<td>273</td>
</tr>
<tr>
<td>7</td>
<td>79</td>
</tr>
<tr>
<td>8</td>
<td>585</td>
</tr>
<tr>
<td>9</td>
<td>23</td>
</tr>
<tr>
<td>10</td>
<td>3</td>
</tr>
<tr>
<td>12</td>
<td>283</td>
</tr>
<tr>
<td>13</td>
<td>39</td>
</tr>
<tr>
<td>14</td>
<td>553</td>
</tr>
<tr>
<td>16</td>
<td>49</td>
</tr>
<tr>
<td>17</td>
<td>593</td>
</tr>
<tr>
<td>18</td>
<td>463</td>
</tr>
<tr>
<td>19</td>
<td>257</td>
</tr>
<tr>
<td>22</td>
<td>525</td>
</tr>
<tr>
<td>23</td>
<td>65</td>
</tr>
<tr>
<td>24</td>
<td>441</td>
</tr>
<tr>
<td>25</td>
<td>249</td>
</tr>
<tr>
<td>28</td>
<td>347</td>
</tr>
<tr>
<td>29</td>
<td>365</td>
</tr>
<tr>
<td>31</td>
<td>559</td>
</tr>
<tr>
<td>32</td>
<td>495</td>
</tr>
<tr>
<td>33</td>
<td>507</td>
</tr>
<tr>
<td>34</td>
<td>607</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>IAEA-SM-316/</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>35</td>
<td>567</td>
</tr>
<tr>
<td>36</td>
<td>151</td>
</tr>
<tr>
<td>37</td>
<td>305</td>
</tr>
<tr>
<td>38</td>
<td>105</td>
</tr>
<tr>
<td>39</td>
<td>125</td>
</tr>
<tr>
<td>40</td>
<td>87</td>
</tr>
<tr>
<td>41</td>
<td>135</td>
</tr>
<tr>
<td>42</td>
<td>185</td>
</tr>
<tr>
<td>43</td>
<td>541</td>
</tr>
<tr>
<td>44</td>
<td>355</td>
</tr>
<tr>
<td>45</td>
<td>167</td>
</tr>
<tr>
<td>46</td>
<td>301</td>
</tr>
<tr>
<td>48</td>
<td>217</td>
</tr>
<tr>
<td>49</td>
<td>599</td>
</tr>
<tr>
<td>50</td>
<td>145</td>
</tr>
<tr>
<td>53</td>
<td>517</td>
</tr>
<tr>
<td>54</td>
<td>575</td>
</tr>
<tr>
<td>55</td>
<td>373</td>
</tr>
<tr>
<td>55-1</td>
<td>405</td>
</tr>
<tr>
<td>55-2</td>
<td>419</td>
</tr>
<tr>
<td>55-3</td>
<td>433</td>
</tr>
<tr>
<td>56</td>
<td>235</td>
</tr>
<tr>
<td>57</td>
<td>313</td>
</tr>
<tr>
<td>60</td>
<td>229</td>
</tr>
</tbody>
</table>

657
An exclusive sales agent for IAEA publications, to whom all orders and inquiries should be addressed, has been appointed in the following country:

UNITED STATES OF AMERICA  UNIPUB, 4611-F Assembly Drive, Lanham, MD 20706-4391

In the following countries IAEA publications may be purchased from the sales agents or booksellers listed or through major local booksellers. Payment can be made in local currency or with UNESCO coupons.

ARGENTINA  Comisión Nacional de Energía Atómica, Avenida del Libertador 8250, RA-1429 Buenos Aires
AUSTRALIA  Hunter Publications, 58 A Gipps Street, Collingwood, Victoria 3066
BELGIUM  Service Courrier UNESCO, 202, Avenue du Roi, B-1060 Brussels
CHILE  Comisión Chilena de Energía Nuclear, Venta de Publicaciones, Amunategui 95, Casilla 188-D, Santiago
CHINA  IAEA Publications in Chinese: China Nuclear Energy Industry Corporation, Translation Section, P.O. Box 2103, Beijing
         IAEA Publications other than in Chinese: China National Publications Import & Export Corporation, Deutsche Abteilung, P.O. Box 88, Beijing
CZECHOSLOVAKIA  S.N.T.L., Mikulandska 4, CS-11686 Prague 1
         Alfa, Publishers, Hurbanovo námestie 3, CS-815 89 Bratislava
FRANCE  Office International de Documentation et Librairie, 48, rue Gay-Lussac, F-75240 Paris Cedex 05
HUNGARY  Kultura, Hungarian Foreign Trading Company, P.O. Box 143, H-1389 Budapest 62
INDIA  Oxford Book and Stationery Co., 17, Park Street, Calcutta-700 016
         Oxford Book and Stationery Co., Scindia House, New Delhi-110 001
ISRAEL  Heiliger & Co. Ltd., 23 Keren Hayesod Street, Jerusalem 94188
ITALY  Libreria Scientifica, Dott. Lucio de Biasio "aiiou", Via Meravigli 18, I-20123 Milan
JAPAN  Maruzen Company, Ltd., P.O. Box 5050, 100-31 Tokyo International
PAKISTAN  Mirza Book Agency, 65, Shahrah Quaid-e-Azam, P.O. Box 729, Lahore 3
POLAND  Ars Polona-Ruch, Centrala Handlu Zagranicznego, Krakowskie Przedmiescie 7, PL-00-068 Warsaw
ROMANIA  Ilexim, P.O. Box 136-137, Bucharest
SOUTH AFRICA  Van Schaik Bookstore (Pty) Ltd, P.O. Box 724, Pretoria 0001
SPAIN  Díaz de Santos, Lagasca 95, E-28006 Madrid
         Díaz de Santos, Balmes 417, E-08022 Barcelona
SWEDEN  AB Fritzés Ungl., Hovbokhandel, Fredsgatan 2, P.O. Box 16356, S-103 27 Stockholm
UNITED KINGDOM  Her Majesty’s Stationery Office, Publications Centre, Agency Section, 51 Nine Elms Lane, London SW8 5DR
USSR  Mezhdunarodnaya Kniga, Smolenskaya-Sennaya 32-34, Moscow G-200
YUGOSLAVIA  Jugoslovenska Knjiga, Terazije 27, P.O. Box 36, YU-11001 Belgrade

Orders from countries where sales agents have not yet been appointed and requests for information should be addressed directly to:

Division of Publications
International Atomic Energy Agency
Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria