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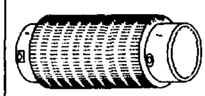
SPENT FUEL MANAGEMENT NEWSLETTER

INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA



No.1

March 1990

CONTENTS

FOREWORD

This Newsletter has been prepared in accordance with the recommendations of the International Regular Advisory Group on Spent Fuel Management and the Agency's programme (GC XXXII/837, Table 76, item 14).

The main purpose of the Newsletter is to provide Member States with new information about the state-of-the-art in one of the most important parts of the nuclear fuel cycle - Spent Fuel Management.

The contents of this publication consists of two parts:

(1) IAEA Secretariat contribution - work and programme of the Nuclear Materials and Fuel Cycle Technology Section of the Division of Nuclear Fuel Cycle and Waste Management, recent and planned meetings and publications, Technical Co-operation projects, Co-ordinated Research programmes, etc.

(2) Country reports - national programmes on spent fuel management: current and planned storage and reprocessing capacities, spent fuel arisings, safety, transportation, storage, treatment of spent fuel, some aspects of uranium and plutonium recycling, etc.

The IAEA expects to publish the Newsletter once every two years between the publications of the Regular Advisory Group on Spent Fuel Management.

The members of IAEA Secretariat are always ready to provide further information on any aspect of the Agency's activities. We hope that the Spent Fuel Management Newsletter will become a forum for the exchange of information and provide ideas on all aspects of this important subject.

The IAEA is grateful to all the authors who contributed to this Newsletter. F. Sokolov of the Division of Nuclear Fuel Cycle and Waste Management was the IAEA Technical Officer responsible for the preparation of this publication.

INTRODUCTION

At present, two main spent fuel management options are being followed: reprocessing and direct disposal of spent fuel. One group of countries such as France, Japan, UK and the USSR has chosen to reprocess of spent fuel and to introduce the commercial utilization of MOX fuels, initially in thermal reactors and later possibly in fast breeder reactors. Such countries as the Federal Republic of Germany, Belgium, Italy, Japan, the Netherlands and Switzerland have contracts for reprocessing their fuel outside of their countries. Bulgaria, Czechoslovakia, GDR, Hungary and Finland have contracts to return spent fuel to the country of manufacture (USSR). China and Argentina are in an advanced stage of R&D reprocessing programmes with the aim of completing a prototype facility starting in several years. Another group of countries, such as Canada, Spain, Sweden and the USA are following the long-term storage of spent fuel as a necessary step before direct disposal into deep geological formations.

At the present time much attention is given to the subject of spent fuel storage in practically all countries with nuclear programme whether they are reprocessing spent fuel or not. The spent fuel is stored at the reactor (AR), away from reactor (AFR) and in reprocessing plants, awaiting either chemical processing or final disposal, depending on the fuel cycle concept chosen by the individual countries.

At present the largest part of spent fuel is stored in wet condition predominates. The alternative, the dry storage is regarded for two potential applications: (i) as an interim option to wet storage and (ii) as a method of final spent fuel disposal. Four dry storage concepts have reached a mature status: (i) vaults, (ii) silos or concrete canisters, (iii) spent fuel transport and storage casks, (iv) sub-surface dry caisson or dry wells.

Spent fuel management is currently very important for countries with operating nuclear power plants as well as for countries which are going to develop nuclear programmes.

Projections for future spent fuel management are heavily influenced by the delay or lack of political decisions in a number of countries, and their impact on the timely development of technical solutions for the selected spent fuel strategy.

Studies made by the NEA and IAEA* have projected that annual spent fuel arisings will reach about 10 000 t HM in 2000 and between 10 000 and 15 000 t HM in 2005. In the same years, cumulative arisings will reach about 165 000 t and between 210 000 t and 240 000 t HM, respectively.

These estimates provide an order of magnitude of spent fuel arisings which must be stored, prior to either reprocessing or final disposal. Recognizing that it will take many years to develop large scale repositories for spent fuel (and high level wastes from reprocessing plants, or major breeder reactor programmes), storage will be the primary method for management of spent fuel until well into the next century.

Consequently nearly all nuclear countries will increase their existing at-reactor pool capacities by implementing compact racks, double-tierce, rod consolidation and by increasing the dimensions of existing

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* Source: NEA/IAEA, Nuclear Energy and its Fuel Cycle, Prospects to 2025, OECD, Paris (1987)

pools. Investigation of spent fuel behaviour in storage pools regarding higher burnups, storage of consolidated fuel rods and longer storage time will be continued.

To provide the necessary additional away-from-reactor (AFR) storage capabilities some countries have already developed the capacities needed (Bulgaria, Finland, Germany,FR, Sweden, USSR etc.) while others are in advanced planning stages (Spain, USA, etc.); a third group of countries such as the Republic of Korea and Mexico are in the early stage of spent fuel storage planning.

R & D on laboratory and industrial scale to design new away-from-reactor dry storage facilities - Federal Republic of Germany (bunker store - passively cooled vault), Republic of Korea, Netherlands (vault), UK (vault for AGR fuel) and USA (NUHOMS) will be carried out.

Taking into account the currently existing storage capabilities, those under construction and in the planning stage as well as the improvements to increase existing storage capacities there is reasonable assurance of sufficient storage capabilities for spent fuel arising in the future.

The reprocessing industry, already established or will be established in some countries, could increase further if strategies including widescale introduction of breeders and/or thermal recycle predominate.

At the end of 1989, about 5% of discharged LWR fuel were reprocessed. The quantity of spent fuel to be reprocessed will be increased in the near future as well as reuse of extracted fissile and fertile materials in thermal and fast reactors. It should be taken into account that intensive developing of reprocessing in even several countries will have an influence on many aspects of the nuclear fuel cycle and require international co-operation. Provision of the back-end of the nuclear fuel cycle facilities with reliable, corrosion-resistant and unexpensive construction materials as well as development and improvement of remote technologies will remain important tasks in the nuclear community for the near future.

In general, the design, technological, economic and material problems of safe spent fuel storage will remain one of the important tasks of the nuclear community, especially dry storage technology, rod consolidation and other advanced concepts.

IAEA NEWSBRIEF

In the first issue of the Spent Fuel Management Newsletter it may be of interest to our readers to learn something of who we are and what we do here in the IAEA. Figure 1 shows the organization of the IAEA.

Since 1957, the International Atomic Energy Agency in Vienna, Austria, has served as the world's central intergovernmental forum for scientific and technical cooperation in the field of peaceful nuclear energy. It has a unique place within the international community and the United Nations system, as governments have endowed it with special responsibilities integral to strengthening the world's economic development and well-being.

The IAEA's fundamental mission is to assist countries in their peaceful uses of the atom in agriculture, health, energy, and other fields, and to apply its safeguards on nuclear materials and facilities provided

ORGANIZATIONAL CHART

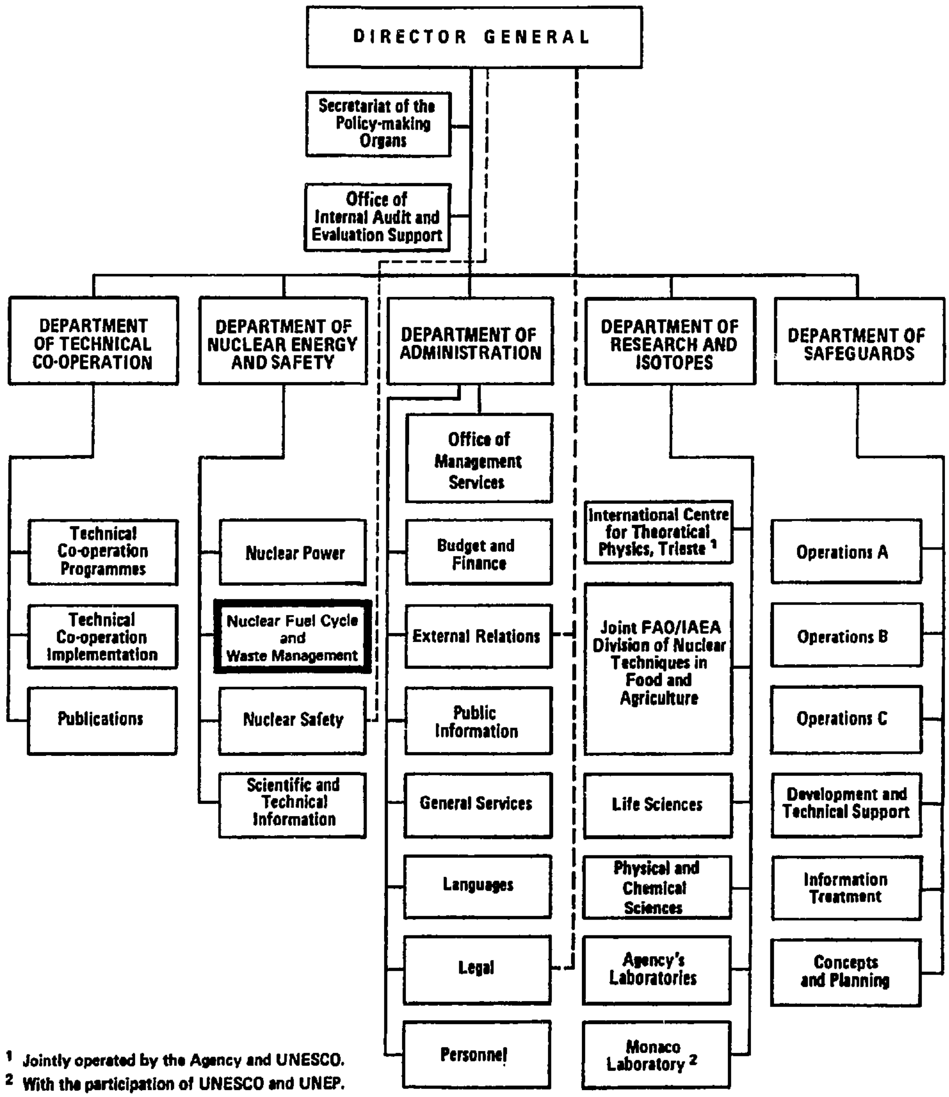

Fig. 1 Organization of the International Atomic Energy Agency

through its programmes or upon request from Member States, so as to verify that they are not used for military purposes. This distinctive verification role is unprecedented in international relations.

At present 113 countries in the world are members of the IAEA. Their interests are represented through the Agency's two principal policy-making organs - the General Conference, which consists of all Member States, and the Board of Governors. Their decisions shape the organization's resources, programmes, and priorities.

The spent fuel management, technology and safety subprogramme is one part of the Nuclear Materials and Fuel Cycle Technology Section activity. The Division of Nuclear Fuel Cycle and Waste Management consists of two Sections: (i) Nuclear Materials and Fuel Cycle Technology and (ii) Waste Management.

HISTORY OF IAEA

Activities in Spent Fuel Management

Experience with the handling and storing of spent nuclear fuel now exceeds forty years. In 1976 the International Atomic Energy Agency (IAEA) identified interim storage of spent fuel as an important independent step in the nuclear fuel cycle. Before this time spent fuel storage was seen by most nuclear power countries as well as by the IAEA as a problem of waste management. The increased importance placed on spent fuel storage was caused by many factors; political, economic and technical and posed difficulties for many IAEA Member States.

The initial IAEA study pointing out the importance of spent fuel storage was a working group review called the Regional Fuel Cycle Centre Study (RFCC). It was found in the RFCC study that while the front-end of the fuel cycle was well developed and was providing fuel for the various power reactors, the back-end of the fuel cycle was not yet fully developed for many countries. Because of various economic, political, and technical considerations, a final decision on the ultimate disposition of spent nuclear fuel is being investigated by many nuclear power countries.

Various studies sponsored or hosted by the IAEA have shown that this is a worldwide problem. The RFCC study estimated in 1977 that a large amount of fuel would need to be stored by year 1985; the International Nuclear Fuel Cycle Evaluation Study (INFCE) stated in 1980 that significant spent fuel arisings will be produced by the year 2000; and most recently the OECD/NEA-IAEA 1987 study on Nuclear Power and its Fuel Cycle stated that by 2020 major amounts of fuel will still remain in storage.

The IAEA recognizes the importance of spent fuel storage to many Member States and has been involved in related activities and studies. The first assistance was given in 1976 when the IAEA and the United States Nuclear Regulatory Commission (USNRC) sponsored a training session on compact storage racks. The RFCC study was completed in 1977. During this study consideration was given to the combining of various fuel cycle activities at a central location. The first step would be to consider a spent fuel storage facility as a location which could eventually be used to complete recycle activities, including reprocessing, mixed oxide fuel production and perhaps later also waste disposal. One of the most significant contributions of this study was the identification of the need for international cooperation and coordination of effort. It also made the fact clear that very little information on the storage of spent fuel was available.

It also became apparent that to a large extent spent fuel storage was being managed by individual utilities within the countries rather than on a national level. For example, as a reactor pool became full, the utility would re-rack with compacted racks. In some countries utilities did not have the freedom to move the fuel to other storage areas but were under contractual obligation to restrict the movement of spent fuel.

At first, the majority of the water cooled nuclear plants were constructed with normal spent fuel storage racks capable of storing perhaps two-thirds to one and two-thirds cores. It was expected that after an initial cooling time of two to three years, the spent fuel would be moved to a reprocessing plant. When the original programmes to reprocess were delayed, the plan was modified to allow the removal of the original fuel racks and their replacement with "compact" racks with smaller centre-to-centre distances. More precise calculations of the coefficient of criticality (K^{eff}) used for licensing permitted more fuel assemblies to be stored in the same space. It then became apparent that maximum storage could be achieved by the addition of racks containing neutron poisons. These changes have been made now at most water reactors. Therefore, additional storage must come from new means, either by the "compaction" of the fuel pins from several assemblies into the same amount of space required by one discharged assembly or by the construction of additional storage facilities. Each change of existing storage facilities proved expensive and time consuming and involved some safety risk caused by more frequent movement of the fuel in the pool itself.

One of the primary goals of the IAEA is to identify methods of increasing storage capacity so that utilities that are still building facilities might be able to avoid the expense and time of intermediate steps and go directly to an optimum storage technique if required.

In 1978, a special Consultants Group was convened by the IAEA to identify what involvement the Agency should have and what the status of spent fuel storage was throughout the world. The results of the meeting indicated that coordination on an international basis would be beneficial and, specifically, that alternatives for solving the problem were needed.

In 1979-80 the IAEA hosted the INFCE study. Many beneficial recommendations came from the study. Among the conclusions in the Working Group 6, Spent Fuel Storage, were the following:

- Experience with wet storage of water reactor spent fuel exists for periods up to 20 years with low burnup fuel. No significant difficulties are expected in projecting spent fuel behaviour in wet storage to longer storage times and higher burnups. Nevertheless, observation and investigation should be continued to evaluate the behaviour of high burnup spent fuel assemblies during prolonged storage periods and to confirm the present experience.

- Wet storage of water reactor spent fuel, including the use of compact racks, should be regarded as a proven technology. The use of compact racks should be encouraged in existing and future reactors to increase storage capacity and to minimize the risk of deficiencies in capacity.

- Some concepts of dry spent fuel storage are in operation in various countries. Further investigation are necessary to decide whether those concepts can supplement wet storage beyond that needed after initial cooling. Dry storage is an alternative to extended interim storage.

The Coordinated Research Project on the behaviour of spent fuel assemblies during extended storage (BEFAST) was started in 1981 as a follow on study to the original 1978 program. It's goal is to study the behaviour of spent fuel and storage facility components during extended storage of spent fuel. BEFAST II initiated work during 1987 with an increase of work scope to include dry storage monitoring.

The results of two World Surveys of Spent Fuel Storage Experience for Wet and Dry Storage were published by the IAEA in 1982 and 1988 respectively. Both of these studies concluded that the storage of spent nuclear fuel over extended periods of time continues to be a safe practice and that while the vast majority of fuel is being stored in water pools the trend for long term storage is towards dry storage.

In 1979 the IAEA initiated a study on international spent fuel management. The objectives of this study were to examine the potential for international cooperation in the management of spent fuel and to assist the IAEA in defining what role it might play in solving problems created by the growing accumulations. The results published in 1982 covered technical, economic, and institutional aspects of spent fuel management.

Coordinated meetings on the topic of spent fuel management were held in April 1982, with ANS/ENS in Brussels; in May 1982, with OECD and the Junta de Energia Nuclear of Spain in Madrid; in September 1983, with JEN in Madrid; in 1987 in Vienna a joint IAEA/NEA Symposium on the Back-End of the Nuclear Fuel Cycle: Strategies and Options; Conference of Nuclear Power Performance and Safety and others.

RECENT ACTIVITIES

The Nuclear Materials and Fuel Cycle Technology Section activities in the field of spent fuel management are concentrated on the improved technology with emphasis on safety and reliability as well as environmental and economic points of view.

During 1989 a number of international meetings were organized and publications were prepared:

- "Guidebook on Spent Fuel Storage" (2nd edition). Same as the first edition, this document is expected to be widely used in Member States for critical comparison of various existing approaches and for justification of concrete national solutions.

- "Safe Spent Fuel Storage and Possible Ways to avoid Fuel Damage". This report is the result of international investigations which started in 1986 and covers some considerations on how to avoid the occurrence of an accident and how to mitigate it if it had occurred.

- "Management of Severly Damaged Nuclear Fuel and Related Waste". An Advisory Group finalized this report and gave recommendations on further Agency activities in this area, including management training, technology catalogue, technical handbook, directory of experts, safeguards requirements for damaged fuel, etc.

- "Management of Spent Fuel from Research and Prototype Power Reactors and Residues from Post-Irradiation Examination of Fuel" reviewed the state-of-the-art in the field of management of the above mentioned fuel and developed recommendations for the future activities in this field.

- A Technical Committee Meeting on "Methods for Expanding Spent Fuel Storage Facilities" with participation of specialists from 14 countries analysed the state-of-the-art in this field and defined areas of concentration for future coordinated efforts in the optimization of rod consolidation, high density racks, transport/storage casks, silos and vaults. The proceedings of the meeting was submitted for publication.

- A Technical Committee Meeting on "Decontamination of Transport Casks and Spent Fuel Storage Facilities" with participation of specialists from 11 countries discussed in depth one of the important components of spent fuel storage technology, aiming at increasing radiological safety. The proceedings of the meeting was submitted for publication.

- A Consultancy for the elaboration of a "Multilingual Glossary of Terms related to Spent Fuel Storage" was held. This Glossary will help to improve the level of understanding between the specialists from various countries.

- An Advisory Group Meeting was convened to finalize the report on "Methodology on Economics of Spent Fuel Storage". The reason to start this project was the realization that all comparisons are too often improperly presented regarding the relative economics of different spent fuel storage options because an appropriate methodology has not been used. This error is often done by attempting to compare assessments of different spent fuel management strategies undertaken by different nations. The present book, which is planned to be submitted for publication in early 1990, has been written to inform professionals involved in the development and implementation of policy decisions and it should act as an aide mémoire for experience nuclear engineers.

- The Technical Committee Meeting on "Improvements of Structural Materials Resistance to Chemical Degradation and Irradiation in the Back-End of the Nuclear Fuel Cycle" was held in connection with the programme to increase the capacity for the prediction of the long-term materials behaviour in aggressive and radiation environment (typical for facilities in the back-end of the nuclear fuel cycle). In addition to papers presented, three panel discussions have been held on specific problems of corrosion and materials resistance in reprocessing, spent fuel storage and waste management. The meeting evaluated the status of present knowledge on the subject and clarified some areas of future investigations.

- A specific project on behaviour of structural materials under irradiation with emphasis on heterogeneous processes was started in November and is planned to be performed as corresponding CRP in order to fulfill the presently existing gap in the mechanistic understanding of irradiation effects.

- A status report on the "Feasibility of the Separation and Utilization of Palladium, Rhodium and Ruthenium from High Level Nuclear Waste" was issued and the new study on recovery and utilization of Cs and Sr was started with participation of experts from six countries. It is planned that the joint study will result in producing a comprehensive report providing a basis for proper considerations of the options for Cs and Sr recovery and for the formulation of appropriate strategies. The future document may be of interest to policy makers in the nuclear fuel cycle and in waste management, as well as to those interested in the processing and applications of radiocaesium and radiostrontium.

- The Coordinated Research Programme (CRP) on Behaviour of Spent Fuel and Storage Facilities Components During Long-Term Storage (BEFAST-II) is in progress. The programme started in 1987 and date of completion is 1991. 12 countries with 16 agreements participate in this CRP. The main task of the programme is to review the status of the national activities in the area of extended spent fuel storage and summarise the results achieved during BEFAST-II programme activity. The final meeting is planned for spring 1991.

The Technical Cooperation Project MEX 9/035/1 on Spent Fuel Management between the Mexican Government and the IAEA is in progress. This project will assist to choose methods for increasing the temporary storage capacity for spent fuel at the site of the Laguna Verde nuclear power plant

REGULAR ADVISORY GROUP ON SPENT FUEL MANAGEMENT

The Regular Advisory Group on Spent Fuel Management (RAG SFM) was established in accordance with the recommendations of the Expert Group on International Spent Fuel Management in 1982. It has met during 1984, 1986 and 1988.

The RAG SFM is a regular advisory group within the framework of the International Atomic Energy Agency (IAEA).

The Advisory Group consists of nominated experts from the countries with considerable experience and/or requirements in such aspects of the back-end of fuel cycle as storage, safety, transportation, treatment of spent fuel and uranium and plutonium recycling. The members of the SAG are from the following countries:

Argentina, Belgium, Canada, Czechoslovakia, France, Germany,F.R., Japan, Sweden, Switzerland, UK, USA and USSR.

Scope

The RAG SFM activities cover the following main topics:

a) Analysis and summary of spent fuel arisings and storage capacities;

b) Interface between spent fuel storage and transportation activities;

c) Spent fuel storage process and technology and related safety issues;

d) Treatment of spent fuel and some aspects of uranium and plutonium recycling.

Objective of activity

To provide technical advice to the Secretariat regarding the Agency's programme in the back-end of the nuclear fuel cycle;

To serve as a means of exchanging information on the current status and progress of national programmes;

To discuss the Agency's publications in this field and to advise the Agency on proposals for reviews, publications, guidelines and studies;

To assist in the coordination of international activities in the field of the back-end of the nuclear fuel cycle.

Methods of work

The Advisory Group meets every 2 years in order to:
- review and comment on the present activities of the Agency in the area of the back-end of the nuclear fuel cycle;

- discuss the participants' presentations on the national current situation and future plans;

- define the most important directions of national efforts and international cooperation in the area of the back-end of the nuclear fuel cycle;

- review the IAEA activities completed since the last Advisory Group meeting;

- prepare recommendations for future IAEA meetings and other activities in the field of the back-end of the nuclear fuel cycle.

The work between RAG meetings is carried out and coordinated by the Scientific Secretary in cooperation with the Chairman on the basis of the RAG SFM meetings' recommendations. The Scientific Secretary keeps the advisors of the group promptly informed of any important discussions and changes.

After each RAG meeting the TECDOC "Spent Fuel Management: Current Status and Prospect" is published. The document includes:

a) Summary of Agency's spent fuel management programme,
   - publications
   - meetings
   - other Agency related activities (i.e. waste handling, transportation, etc.)

b) Country status reports

c) Summary and recommendations.

IAEA INTERNATIONAL MEETINGS 1990
OF INTEREST TO SPENT FUEL MANAGEMENT SPECIALISTS

(i)	Advisory Group on Spent Fuel Management: Current Status and Prospects Secretary: Mr. F. Sokolov	19 - 22 March Vienna
(ii)	Advisory Group on Environmental Effects of Nuclear Fuel Cycle Facilities and Public Opinion Secretary: Mr. J. Rojas de Diego	2 - 6 April Vienna
(iii)	Advisory Group on Design, Construction Maintenance, Operation and Licensing for Safe Long-Term Storage of Spent Fuel Secretary: Mr. F. Takáts	18 - 22 June Vienna
(iv)	IAEA/NEA Seminar on Spent Fuel Storage - Safety, Engineering and Environmental Aspects Secretary: Mr. F. Sokolov	8 - 12 October Vienna
(v)	Advisory Group on Safe Spent Fuel Management after Nuclear Accident Secretary: Mr. F. Takáts	Oct./November Vienna

IAEA PUBLICATIONS ON SPENT FUEL MANAGEMENT

IAEA-112	Plutonium Utilization	1966
IAEA-115	Reprocessing of Highly Irradiated Fuels	1970
IAEA-143	Plutonium Recycling in Thermal Power Reactors	1972
IAEA-175	Plutonium Utilization in Thermal Power Reactors	1975
IAEA-217	International Inventory of Training Facilities in Nuclear Power and its Fuel Cycle	1979
STI/DOC/10/218	Storage of Water Reactor Fuel in Water Survey of World Experience	1982
STI/DOC/10/240	Guidebook on Spent Fuel Storage	1984
IAEA-TECDOC-333	Status of the Treatment of Irradiated LWR Fuel	1985
IAEA-TECDOC-345	IAEA Spent Fuel Glossary	1985
IAEA-TECDOC-359	Status of Spent Fuel Dry Storage Concepts: Concerns, Issues and Developments	1985
IAEA-TECDOC-360	Status of LWR Rod Consolidation for Storage Purposes: Concerns, Issues and Trends	1985
IAEA-TECDOC-345/R	IAEA Spent Fuel Glossary - Russian Edition	1986
IAEA-TECDOC-408	The Nuclear Fuel Cycle Information System/ An International Directory of Nuclear Fuel Cycle Facilities	1987
IAEA-TECDOC-414	Behaviour of Spent Fuel Assemblies during Extended Storage	1987
IAEA-TECDOC-418	Long-Term Wet Spent Nuclear Fuel Storage	1987
IAEA-TECDOC-419	Spent Fuel Management: Current Status and Prospects of the IAEA Programme	1987
STI/PUB/738	Back-End of the Nuclear Fuel Cycle: Strategies and Options (Proc.Int.Symp.)	1987
IAEA-TECDOC-421	Materials Reliability in the Back-end of the Nuclear Fuel Cycle	1987
STI/PUB/761	Nuclear Power Performance and Safety (Proc.Int.Conf.)	1988
IAEA-TECDOC-345/S	Glosario del OIEA sobre Terminos de Almacenamiento de combustible gastado	1988
IAEA-TECDOC-461	Spent Fuel Surveillance and Monitoring Methods	1988

IAEA-TECDOC-345/F	Glossaire de l'AIEA sur le Stockage du Combustible Irradié	1988
STI/PUB/751	Nuclear Power and Fuel Cycle: Status and Trends 1988	1988
STI/DOC/10/290	Survey of Experience with Dry Storage of Spent Nuclear Fuel and Update of Wet Storage Experience	1988
IAEA-TECDOC-487	Spent Fuel Management: Current Status and Prospects	1988
STI/PUB/749	The Nuclear Fuel Cycle Information System. A Directory of Nuclear Fuel Cycle Facilities	1988
IAEA-TECDOC-513	Management of Spent Fuel from Research and Prototype Power Reactors and Residues from Post-Irradiation Examination of Fuel	1989
STI/DOC/10/305	Nuclear Fuel Cycle in the 1990s and Beyond the Century: Some Trends and Foreseeable Problems	1989
STI/DOC/10/308	Feasibility of the Separation and Utilization of Palladium, Rhodium and Ruthenium from High-Level Waste	1989
STI/PUB/838	Nuclear Power and Fuel Cycle: Status and Trends 1989	1989
(Internal Report)	Advanced Structural Materials for the Nuclear Fuel Cycle Facilities	1989
(Internal Report)	Airborne Gamma Spectrometric Measurements of the Fall-Out over Sweden after the Nuclear Reactor Accident at Chernobyl, USSR	1989

Joint IAEA/VTT Publication

VTT Symposium 63	IAEA Technical Committee Meeting on Methods Used in Design of Spent Fuel Storage Facilities (published by VTT Espoo, Finland)	1985
CONF-860417 UC-85	Third International Spent Fuel Storage Technology Symposium/Workshop Volume 1 of 2 Proceedings (published by NTIS, US Dep. of Commerce)	1986

Coming Back to the "Old" Ideas

During the Atom for Peace initiatives of the late 1950s one of the popular and attractive topic for discussion was the concept of utilizing the by-products from spent nuclear fuel processing. This idea was "renewed" in the 1960s and 1970s, but up to now nothing has been introduced in practice*).

Considering by-products as a valuable element for a variety of processing industry and assuming that removal of selected fission products could, in principle, simplify requirements for the waste storage and disposal, one should not close one's eyes to the fact that the current policy in the back-end of the nuclear fuel cycle can make these theoretically prospective ideas absolutely impracticable. Thus, today the nuclear community has two real alternatives:

- to agree apriori with the practical unfeasibility of the concept and to adopt a technology in accordance with already accepted strategies;
- to undertake one more attempt to analyse the situation in depth, taking into account all aspects of the problem and assuming, if necessary, that the existing approaches can be varied. It seems that after 5 to 7 years it will be too late to raise this question.

Although for the present one cannot generally assert that the "nuclear world" has started the new stage of investigation on the goal-oriented separation of HLW with a subsequent transmutation of long-lived TRU and utilization of selected elements for various purposes, some steps in this direction are taking place. Among them, one should mention the integrated programme "OMEGA" in Japan, corresponding an international project performed under the OECD/NEA "umbrella" and a few IAEA studies.

The IAEA limited activity is mainly intended to provide a basis for further consideration of the options for the recovery of non-fissile by-products, and for the formulation of appropriate strategies. In the framework of this project one analytical report entitled "Feasibility of Separation and Utilization of Ru, Rd and Pd from High-Level Wastes" (TRS NO. 308) was issued in December 1989, and a study on the separation and utilization of Cs and Sr is planned to be finalized in late 1990.

The readers of TRS No. 308 (policy makers in the nuclear fuel cycle and in waste management as well as processors and consumers of platinum group metals - PGMs) will find in the document an analytical review of the market potential of PGMs in the light of the present and future economic prospects; an assessment of the importance of the nuclear by-products in the overall market; a short description of the separation techniques available at present and an evaluation of their merits for PGM recovery from high activity streams. The report prepared by a group of experts from Belgium, China, Germany, F.R., UK and USSR also contains general conclusions and recommendations some of which are stated below:

"Ruthenium, rhodium and palladium are all useful and strategically important. In the short and medium terms, the demand for rhodium is likely to continue to increase because of the growing requirement for motor vehicle exhaust catalysts. The demand for the other PGMs is also likely to increase steadily.

*)One exception: in the early days of the nuclear industry, Caesium and Strontium were separated from defense reprocessing waste at Hanford (USA).

In spite of strategic uncertainties, supply and demand are likely to remain in balance in the medium term. However, in the long term, pressure on supplied as reserves become depleted may lead to increased prices and provide an incentive to look for alternatives. In these circumstances, an alternative source of supply, such as high level nuclear wastes, might become an attractive source of PGMs.

If the worldwide development of nuclear power proceeds as currently envisaged, it is likely that the PGMs (Ru, Rh, Pd) in spent fuel and fission product wastes will constitute a substantial potential resource, perhaps comparable in size to known natural reserves by about 2025. It is important to consider carefully the options for utilizing this potential resource, within the context of the nuclear fuel cycle.

If the PGMs are to be available for use, they will have to be separated before vitrification. Since the PGMs have similar chemical properties, it my be advisable to develop a technique for separation from HLW which will isolate the PGM fraction from the rest of the HLW. The PGM fraction could then be stored together with the insoluble residues for a 30-50 year period. After this period, residual activity would be low and separation of the PGMs for utilization could be evaluated against the technical and economic criteria prevailing at that time. The proposed techniques for the initial separation have to be compatible with the current reprocessing operation and must not create additional waste streams.

Though many techniques have been tested on the laboratory scale none of them have been demonstrated under realistic conditions. Some techniques, e.g. combined reduction and precipitation, show potential for further development.

The radioactivity of the PGM fraction is a strong impediment to its use as a chemical feedstock. In particular, the high radioactivity of ruthenium complicates any PGM recovery processes undertaken within 30-50 years after discharge from the reactor unless a method is designed that can separate almost all ruthenium from the other PGMs. Beyond that period ruthenium activity will have substantially decayed.

The technology of rhodium separation has not yet been developed up to the point where its use in the nuclear fuel cycle context can be guaranteed. Further developments have to be undertaken in order to provide better separation methods. However, after 30-50 years'cooling time and in the absence of ruthenium, the rhodium radioactivity will have decreased to acceptable levels, allowing chemical processing and unrestricted use.

Palladium can be extracted from the PGM mixture by several methods, but it will keep its intrinsic β-activity unless isotopic separation of the long lived ^{107}Pd is performed. No method for such a separation is yet available on an industrial scale.

It is possible to consider using such palladium (or other very low active PGM materials) in applications where its activity is unimportant. But it would then be essential to segregate this active metal from natural material during recycle operations so as to avoid contamination of the whole inventory. This segregation would be difficult to control in many applications.

Since PGMs of nuclear origin are produced by various reprocessing plants international collaboration may be appropriate in order to define the criteria with which the concentrates have to comply for further refining in a specialized facility.

The use of PGMs of nuclear origin requires a careful analysis of the market situation for products which may be slightly contaminated with radioactive nuclides, in order to identify uses which might be compatible with such materials."

A new study, being conducted by experts from Belgium, Germany, F.R., Japan, UK, USA and USSR, examines the present and future market demand for Cs and Sr, the technological feasibility of their separation, the possible impacts on high level waste management resulting from their separation, economic aspects, safety, environmental and public acceptability aspects of their utilization. This project is in progress but open for voluntary contributions from Member States in respect to related considerations, ideas, published materials, comments, etc.

It is expected that these joint efforts will help clarifying the potential of high-level waste partitioning, and will also contribute to the provision of guarantees that civilization and environment are at all times reliably protected from the influence of "atomic rubbish" and that by any chance, mankind does not miss the opportunity to use the unique energetic and material resources generated in nuclear fuel.

A.F. Nechaev
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Cycle and Waste Management

COUNTRY REPORTS

Argentina

1. INTRODUCTION

Argentina has ruled out any option involving the disposal of spent fuel elements without first reprocessing them, not only in view of the significant energy potential of the plutonium they contain (i.e. increased energy reserves of nuclear origin), but also in the light of ecological considerations.

Although for Argentina the problems arising from the storage of high-level radioactive wastes will only become significant around the turn of the century, it was nevertheless decided to tackle in advance the technological aspects of their disposal, so as to ensure that the problem will not be passed on to future generations. This decision was based not only on technical evaluations, but also on ethical considerations, it being felt that the problem should be solved by the generations currently enjoying the benefits of nuclear energy, rather than by those of the future.

2. THE REPROCESSING PROGRAMME

During the period 1968-69, Argentina reprocessed fuel elements from an Argonauta (RA-1) reactor at a laboratory scale plant (PR-1), using a simplified Purex process. The plutonium from 11 kg of 20% enriched uranium that had been irradiated in the Argonauta reactor was recovered. In that connection, it is interesting to note that fuel elements have since been fabricated for the RA-6 pilot reactor, utilizing the uranium recovered at that time.

In 1979 work started on construction of the Radiochemical Processes Laboratory (LPR), Argentina's pilot reprocessing facility.

The reprocessing programme has two main objectives: first, to purify a sufficient quantity of plutonium so as to enable a demonstration programme of mixed oxide recycling to be carried out at the Atucha-I nuclear power plant; and second, to gain sufficient experience in all aspects of reprocessing technology so as to be able subsequently to operate an industrial plant.

The pilot reprocessing plant is currently at the final phase of construction. It is designed to process spent fuel elements from power reactors and will have a daily capacity of 20 kg of uranium. The fuel will be dissolved using the chop and leach process, followed by a three-cycle Purex process. The Pu will be separated during the second cycle by means of a reducing agent, prior to its final purification using anionic resins. The conversion to mixed oxides will be carried out in another part of the plant. We also envisage a liquid waste pre-processing stage, using evaporators in series.

The civil construction work, processing cells and services are now more than 80% complete. The design for the spent fuel element transportation cask has entered the detailed engineering stage. The installations for the pre-processing of the radioactive wastes are at the advanced basic engineering stage. The fluid dynamics tests were commenced in 1987. During 1989, the facility should be partially completed, at which point an initial hot campaign will be mounted as one of the phases involved in startup. Budgetary problems have delayed the startup of the facility.

The facility will be licensed for the following programme:

- To reprocess spent fuel elements from the Atucha-I nuclear power plant with a burnup of approximately 6000 MWd/t and with a decay of over ten years;

- To operate the plant at its planned capacity for a period equivalent to five years.

The plutonium that is purified at the LPR will be made into an oxide, mixed with uranium oxide and turned into pellets, after which it will be sheathed in a separate section within the plant building. The fuel elements will be assembled in the nuclear fuel plant operated by CONUAR S.A.

3. WASTE DISPOSAL

There is currently international consensus on the view that the disposal of high-level radioactive wastes, conditioned in solid form and located in suitable deep geological formations, is a solution which, for present and future generations, will involve risks that are no greater than those normally accepted for daily life.

The radioactive wastes will be incorporated in a borosilicate-type vitreous matrix within a stainless steel recipient. This recipient will then be encased in a lead wall some 10 cm thick with an external metal protective sheet, in order to ensure that the wastes remain isolated for a period of approximately 1000 years. The design of the containers will also comply with the requirements of the IAEA's Regulations for the Safe Transport of Radioactive Materials.

The four nuclear power plants that are planned up to the end of this century represent an installed electrical capacity of 2.3 GW. The wastes arising from reprocessing of the fuel consumed in the course of 30 years will require approximately 2000 containers of approximately 0.60 m diameter and

1.60 m in height. The storage facility for these 2000 containers has a surface area of less than 1 km^2.

Preliminary evaluations showed that the disposal of radioactive wastes in crystalline rocks at a depth of 500 m or more would sufficiently reduce the overall radiological impact. For this reason, and bearing in mind the geological characteristics of the country, it was decided to dispose of the wastes in stable granitic formations located at a depth of 500 m, away from seismic zones and with low hydraulic conductivity.

The first stage in the siting studies was to examine all of the known outcrops of granitic rocks in Argentina. This led to the identification of 198 possible granitic outcrops.

The second stage was to draw up a shortlist from among the formations identified; this resulted in the choice of seven granitic outcrops located in the provinces of Chubut and Río Negro, in the south of the country.

The third stage, involving the a topographical survey of the granitic outcrop shortlisted led to identification of the La Esperanza and Chasicó massifs, in the province of Río Negro, and those of Sierra de Calcatapul and Sierra del Medio, in the province of Chubut, as the most appropriate in which to continue with detailed studies.

The studies commenced with the granitic outcrop of Sierra del Medio, in accordance with the following plan:

(a) Photographic interpretation;

(b) Statistical alignment analysis;

(c) Geological and geophysical inspection of the rock massif;

(d) Intermediate perforations to a depth of 200 m;

(e) Regional geomorphological and hydrogeological analysis; and

(f) Small-diameter deep perforations to a depth of 800 m.

In the near future we plan to commence detailed geological and hydrogeological studies on the site selected in order to determine the hydraulic conductivity of the massif and to conduct various chemical and physico-chemical measurements in the deepest perforations. We are also performing geological studies of the volcanic masses located in the Gastre depression, which surrounds the Sierra del Medio. The aim of these studies is to determine the past and future influence of volcanic eruptions on the stability of the Sierra. The studies carried out to date have enabled us to establish that the outflows have not resulted in deterioration of the Sierra and that the ages of the outflows, determined using the Ar-K technique, are greater than $0.8 + 0.1 \cdot 10^6$ years.

The results obtained so far confirm the suitability of this site. However, should subsequent results disqualify it, we will undertake a study on one of the other shortlisted granitic bodies.

4. PERIOD OF CONSTRUCTION AND COST OF THE REPOSITORY

The time necessary for construction of the repository, including the definitive planning stage, is in the order of eight to ten years. The period of operation is expected to be 50 years, with sealing to take a further 3 years. The repository should become ready to receive containers during the period 2010 to 2015.

The cost analysis which forms part of the feasibility study has not yet been completed. At the present stage of the study, however, it is estimated that the cost will be approximately US $350 million, which represents some 1.5 to 2% of the cost of a nuclear-generated kWh.

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Belgium

1. NUCLEAR POWER PLANTS

Belgium had 7 nuclear power plants in operation at the end of 1988, all PWR's with a total capacity of 5415 MWe. The Doel 1 and 2 (785 MWe total) and Tihange 1 (870 MWe) power plants have been brought on line in 1974-1975. In 1982, two new 900 MWe units, Doel 3 and Tihange 2 were connected to the grid, in June and October respectively ; Doel 4 and Tihange 3 (both of 980 MWe) followed in April and June 1985.

Belgium and France are partners, on a 50/50 basis, in the existing Chooz power plant in France and in the Tihange 1 plant in Belgium.

2. REPROCESSING AND SPENT FUEL MANAGEMENT (Appendix 1)

2.1. Reprocessing

Four reprocessing contracts have been signed with COGEMA, a 100% subsidiary of the French Commissariat à l'Energie Atomique.

The first two signed in October 1976 pertain to fuel from the first thirds of the cores of Doel 1 and 2 and of Tihange 1 (about 40 tU). All the fuel covered by these contracts has been reprocessed in May 1980 and in December 1981.

A third contract for about 100 tU has ensured reprocessing, in COGEMA's UP2 plant in La Hague of most of the fuel unloaded between 1977 and 1979. The last fuel covered by this contract has been reprocessed in May 1985.

The uranium recovered so far within the framework of the first three contracts has been sent to enrichment plants, whereas the plutonium has been used in priority in fast breeder reactors at Creys Malville and Kalkar.

A fourth contract covers the reprocessing, from 1989 to 1998, of 464 tU of Belgian fuel in the new COGEMA's UP3-A reprocessing plant at La Hague.

The conditioned waste resulting from the reprocessing of Belgian fuel, except the part covered by the 1976 contracts, shall be sent back to Belgium from 1993 onwards.

2.2. Spent fuel management

The spent fuel storage capacity at reactor has been progressively increased since 1979 and now equals 1350 tU. Expressed in terms of annual unloadings, the evolution of the spent fuel storage capacities is the following : 2 years in 1977, 9 years in 1988 and 12 years around 1994 (high burnup fuel).

In order to avoid economic penalties in case of disturbance in the spent fuel management (necessity of a whole core unloading, delay in transport, ...), a storage capacity equivalent to one whole core plus about 0.4 core will normally be kept free in each nuclear unit.

Therefore, either a new increase of storage capacity will be needed around 1995, or a new reprocessing contract will have to be signed before that date (see Appendix 2). The choice between these two solutions will be widely influenced by the commercial policies of the reprocessors and the other companies dealing with the fuel recovered after reprocessing.

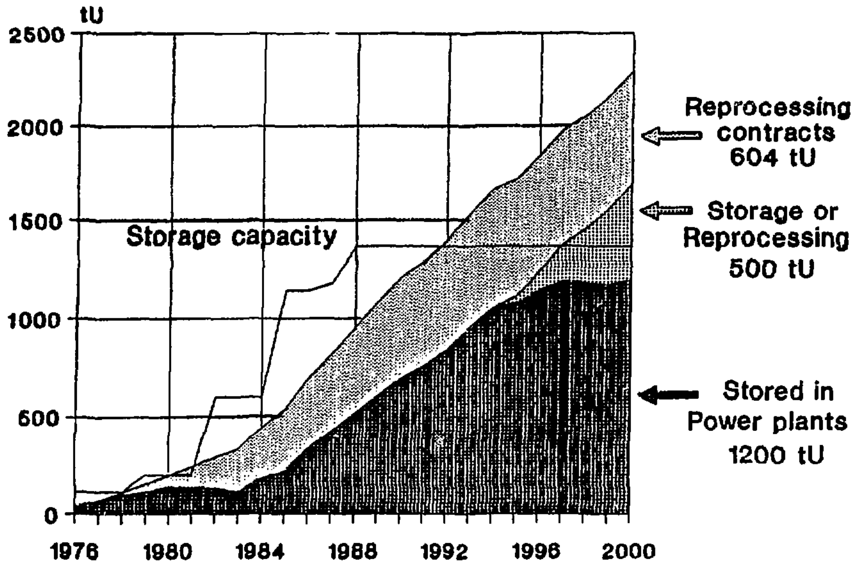
APPENDIX 1

BELGIUM
SPENT FUEL - SITUATION END 1988

REPROCESSING CONTRACTS WITH COGEMA	
UP2 76	40 tU
UP2 77/79	100 tU
UP3A 80/89	464 tU
TOTAL	604 tU
QUANTITIES OF FUEL (Situation end 1988)	
REPROCESSED (UP2)	140 tU
STORED IN LA HAGUE - UP3A 80/89	289 tU
STORED IN THE REACTOR POOLS	516 tU
TOTAL (discharged)	945 tU
FUEL STORAGE CAPACITY AT THE POWER PLANT	1366 tU
ADDITIONAL CAPACITY OF ONE WHOLE CORE FOR EACH PLANT	ABOUT 420 tU
QUANTITIES OF FUEL DISCHARGED	
BURNUP - 33,000 MWd/tU	150 tU/YEAR
BURNUP - 45,000 MWd/tU	110 tU/YEAR
PLUTONIUM CONTENT OF THE IRRADIATED FUEL (BURNUP 45,000 MWd/tU)	
Pu TOTAL	10 Kg/tU
Pu FISSILE	6.5 Kg/tU

APPENDIX 2

Irradiated fuel management



Received from Permanent Mission of Belgium

Brazil

Angra - I

Present capacity of the Spent Fuel Pool : 363 fuel elements.

Planned capacity of the Spent Fuel Pool : approximately 900 fuel elements, depending on the results of feasibility studies being undertaken on the utilization of super compact racks.

Forecast of Spent Fuel Storage per annum : 40 fuel elements.

Angra - II

Capacity of the Spent Fuel Pool project : 768 fuel elements.

Planned capacity of SFP : there is no increase foreseen at present.

Forecast of Spent Fuel Storage per annum : 64 fuel elements.

Received from Permanent Mission of Brazil

Bulgaria

INTRODUCTION

1. Nuclear facilities in Bulgaria

The use of atomic energy for peaceful purposes in Bulgaria started in 1961 with putting into operation of the IRT-1000 research reactor near Sofia. After its reconstruction its power has been increased to 2000 kw and it is now known as IRT-2000.

The very first nuclear power reactor WWER-440 type was commissioned in 1974. Nowadays there are four WWER-440 and one WWER-1000 reactors in operation all of them at the NPP Kozloduy. Start-up of the next WWER-1000 at that NPP is forthcoming. Another four WWER-1000 units are under construction at our second NPP - Belené.

2. Storage capacities

The spent fuel of our facilities is stored in water pools at the reactors. The spent fuel pond of IRT-2000 has a capacity of 54 F.As. It could be doubled if necessary by inserting aluminium tubes into the rack cells and putting two assemblies in them. This method has been tested and its applicability demonstrated. The pools of WWER-440 have a capacity of 360 F.As.if the fuel is stowed in one rack. In special cases, with the permission of the state authority, it also could be approximately doubled by using the second /upper/ rack. For WWER-1000 the designed capacity is 410 F.As. stowed in one rack. The First Away From Reactor Storage located at the NPP Kozloduy site is expected to be in operation by the end of 1989. Its design capacity is 4900 F.As. - WWER-440 type. Due to the commissioning of WWER-1000 reactors at the NPP Kozloduy its design has been slightly changed in order to receive and store this type of F.As. as well. Things being so the total number of assemblies of both types stowed in it will be less than 4900. Erection of another AFRS is foreseen in next years.

SPENT FUEL ARISING

There are 34 assemblies in the IRT-2000 spent fuel pond stored-up during its 27 years of operation. They are planned to be stored there for a long term or to be transferred to some of the

AFRS. No transfer of spent fuel from that facility took place in the past.

Concerning WWER-440 the annual spent fuel arising is 115 F.As. per unit or 460 totally. 164 assemblies per year from the two WWER-1000 are expected to be included in the balance during the next 2-3 years and another 330 from our second NPP in the coming years.

TRANSPORTATIONS

According to the agreements the spent fuel is to be sent back to the supplier after a certain period of interim storage - at present 5 years. For the time being only WWER-440 type fuel assemblies are transferred back. Since 1979 21 shipments have been performed and about 3000 F.As. have been returned back to the supplier. All the shipments have been carried out by using TK-7 containers with 30 F.As. capacity. The containers are loaded in the reactor hall, positioned vertically on the special autotrailer and transported to the port of the NPP Kozloduy at Danube. There they are re-loaded on barge and shipped to the supplier. The future shipments of WWER-1000 type fuel assemblies are planned to be carried out in the same manner using TK-13 containers with 12 assemblies capacity.

SAFETY

Present technological schemes ensure safety of storage and transportation. The pools are provided with duplicate systems for water cooling and make-up, air ventilation above the water level and in the surrounding areas, purification of ventilation exhausts from radioactive aerosoles. The base and the walls of the pools are constructed of thick reinforced concrete, lined with plates of carbon steel, then stainless steel. To avoid pool water leakage into environement, sumps for water collection in case of leakage are provided under the pools. Due to the increased water leakage of the pond of the NPP Kozloduy unit-1 after 10 years of operation a repair was carried out after the fuel had been transferred to the pond of unit-2. After the repair the leakage is minimized. Water level and temperature in the pools are automatically controlled but visual control by operators is possible as well. Chemical analyses carried out periodically for our research reactor and power reactors indicated that pool-induced degradation do not occur regardless of some relatively long periods of storage - up to 27 years for IRT-2000 assemblies. Before the shipment to the supplier all the F.As. have also been tested for hermetical state and results confirmed the previous conclusion. To date there have been no serious incidents during transportation such as temperature increase or radiation release.

STORAGE

Storage of the spent fuel in water pools "wet storage" is adopted as a more developed and proven technology. For Kozloduy AFRS and probably for the next one the storage of fuel in baskets is considered to be more reliable than putting the assemblies in racks.

Water quality is controlled and adjusted by using purification systems and adding ammonia and hydrozine. Attention is paid to keep pH = 6÷8 and halogen content less then 500 ppb.

It is not planned at present to increase the ponds capacity by using high density racks or rod consolidation for both AR and AFR storages.

RECYCLING OF THE FUEL

The concept adopted nowadays is to transfer the spent fuel back to the supplier. According to it the fuel, after some years of storage, and the rights to it are transferred to the supplier. So there are no reprocessing plants and plans to construct such facilities for uranium and plutonium reprocessing.

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Canada

1.0 SPENT FUEL STORAGE

In Canada approximately 600,000 CANDU spent fuel bundles (18,000 MgU) are now stored in water pools at the reactor site. Additional storage will be required at many of the nuclear generating stations beginning in the

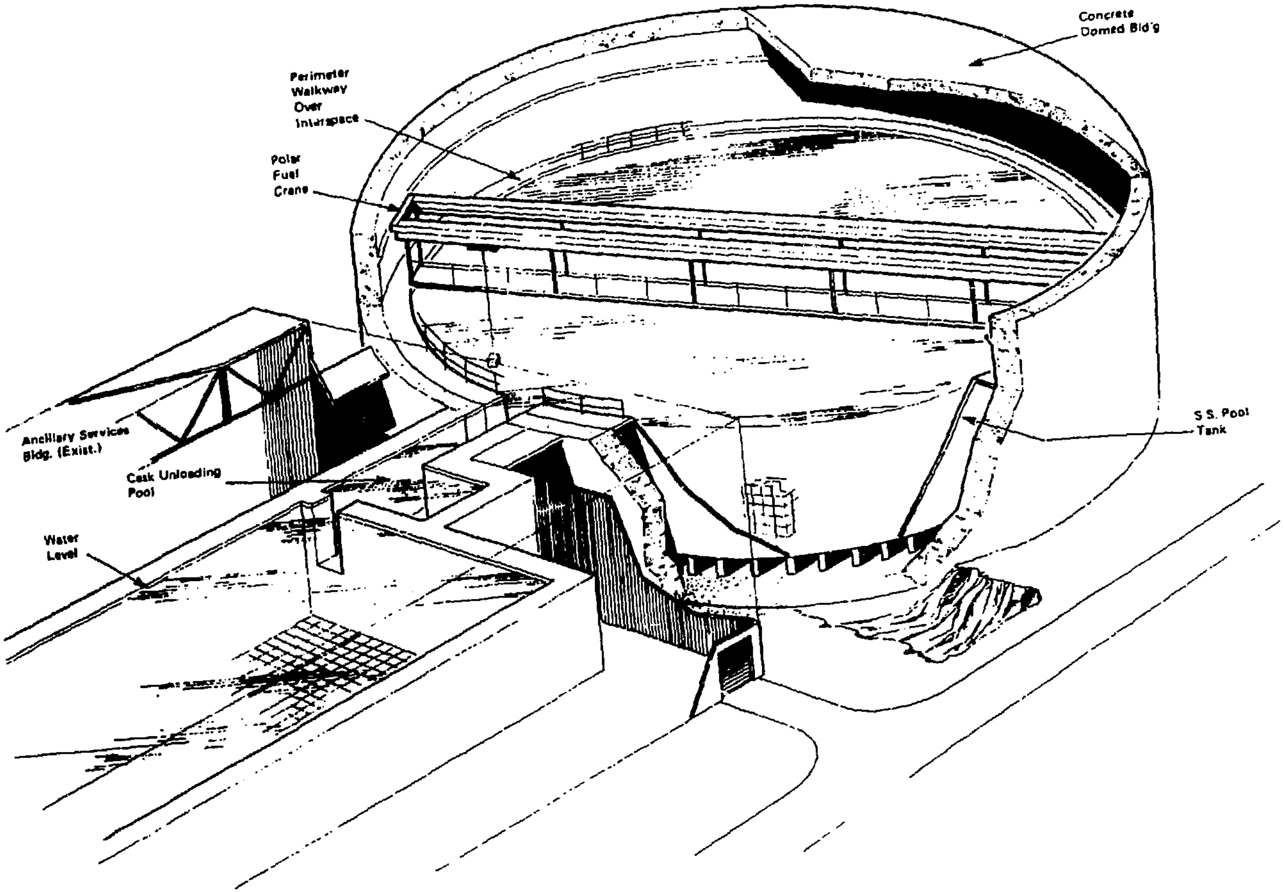

Fig. 1 Circular Water Pool

early 1990s. The additional storage will be provided by water pools or by dry storage in concrete containers. Following is a description of the current status of the water pool storage and dry storage programs.

1.1 Water Pools

All of the water pools being used for storage of spent fuel in Canada are located at the reactor site. There are two basic types of pool currently used, the primary pool which receives fuel directly from the reactor via the on-power fuelling system, and the secondary or auxiliary water pools which provide additional storage capacity as required. Depending on the capacity of the primary pool the spent fuel may be stored for nine years or more, or for as short a period of time as six months before transferring to the secondary pool. The secondary pool may be interconnected to the primary pool or may be some distance away at another location on the site. Transfer between bays can be under water in the former case but an on-site transport cask is required for the latter.

All water pools currently being used are rectangular in shape and are approximately 9 m deep. The pools are concrete structures which are lined with stainless steel or epoxy.

Ontario Hydro, which will have an installed nuclear capacity of approximately 15,000 MWe with the completion of Darlington NGS, is currently developing improved water pools which can reduce storage and maintenance costs. One such pool is shown in Figure 1. This pool is deeper (12.5 m) and is a circular stainless steel tank structure; typically the capacity of the circular pool would be 500,000 CANDU fuel bundles.

1.2 Dry Storage

Atomic Energy of Canada Limited (AECL) developed the concrete canister shown in Figure 2 during the early 1970s. The canister has proven to be a successful concept and is now used at the reactor sites shown in the following table.

Location	No. of Canisters	Stored Fuel Capacity
WNRE	19	23 MgU
Gentilly	11	67 MgU
Douglas Point	46	300 MgU

In addition to the above applications the concrete canister will be used to store 75 MgU from Canada's first demonstration power reactor (NPD). Additional storage capacity will also be provided by concrete canisters at Point Lepreau, a 600 MWe reactor owned by New Brunswick Power.

Ontario Hydro currently uses only water pools for the storage of spent fuel but is developing a dry storage concept called the concrete integrated container (CIC). The concrete integrated container is similar to the AECL concrete canister, but is designed to be a universal container that can be used for the storage, transportation and, possibly, disposal of Ontario Hydro's used fuel. This concept is referred to as an "integrated system", and has been under development since 1981.

The current CIC design (Figure 3) is based on the following specifications: It will contain 384 CANDU fuel bundles (7 years old or

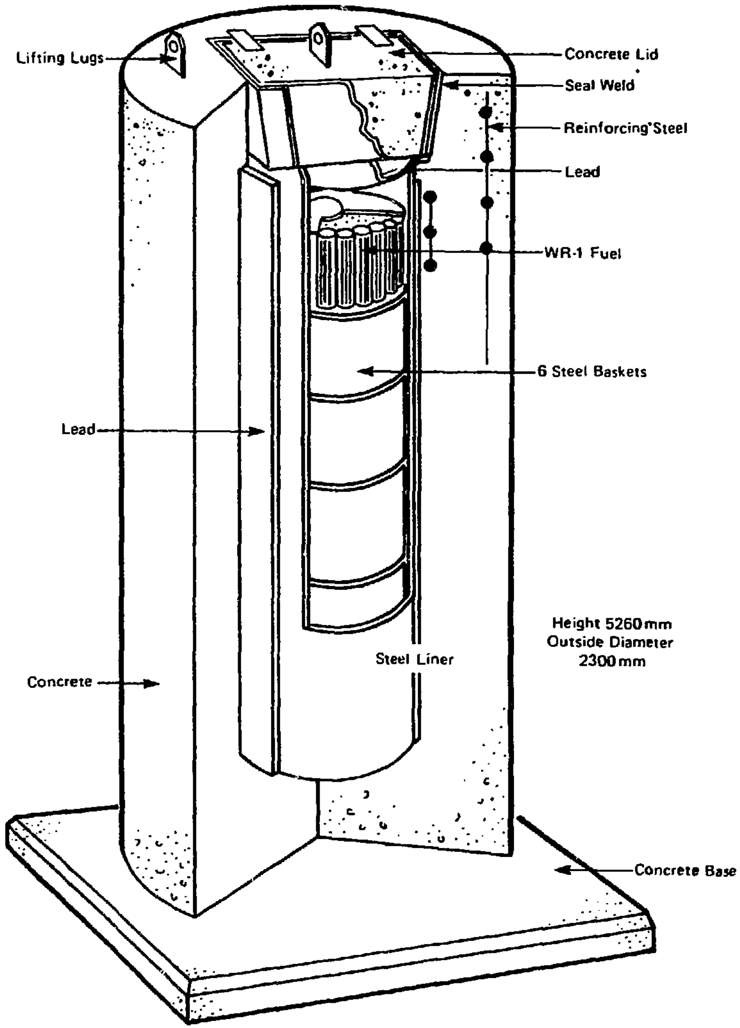

Fig. 2 Concrete Canister for Fuel Storage

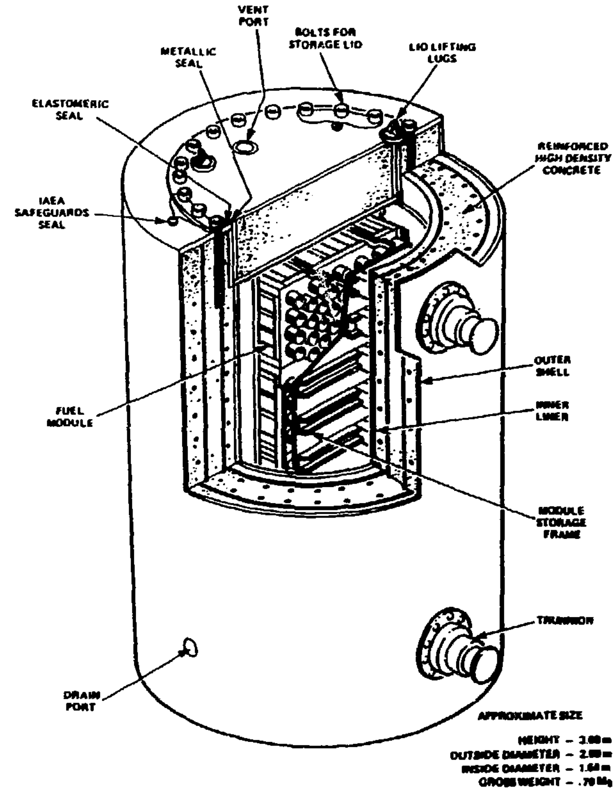

Fig. 3 Concrete Integrated Container (CIC)

older), can be loaded under water, has inner and outer steel liners for ease of decontamination, has separate storage and transportation closures, and is constructed from a durable high-density concrete, which provides good radiation shielding. A two-stage CIC demonstration program has been initiated. One CIC was built and loaded in 1988 in a used fuel storage demonstration at the Pickering nuclear generating site; one more is planned for construction in 1989 for additional tests. Successful demonstration will likely lead to full-scale deployment of the system at Pickering.

1.3 Fuel Durability in Storage

In support of the spent fuel storage program research was initiated by AECL in 1978 to investigate the long-term performance of spent fuel in wet and dry storage. The dry storage experiments consist of the easily retrievable basket (ERB) experiment (dry storage in air at seasonally varying temperatures), and the controlled environment experiment, Phase 1 (CEX-1, storage in dry air at 150°C) and phase 2 (CEX-2, storage in moisture saturated air at 150°C). Fuel is periodically retrieved from the experiment (annually at first) for non-destructive and destructive examination.

The wet storage program was also initiated in 1978 and includes fuel which has been stored in water for as long as thirty years. Fuel is retrieved from the wet storage program for destructive and non-destructive examination every ten years, the first re-examination occurring in 1988.

Results from both programs suggest fuel performance will be good in both modes of storage and are providing valuable data for the development of storage system design.

2.0 TRANSPORTATION

To date the transportation of spent fuel in Canada has been limited to small scale shipments between operating nuclear stations and experimental facilities. Large scale shipments will likely not occur until a disposal facility is established sometime in the next century.

Nevertheless Ontario Hydro has designed and built a road transportation cask (Figure 4) capable of carrying 192 (4.2 MgU) ten-year old spent fuel bundles. The cask is a forged stainless steel structure weighing approximately 35 Mg and will be used to demonstrate the practicality of large-scale transportation. The cask will also be used to transport smaller quantities of more recently discharged spent fuel.

In addition to the road cask Ontario Hydro have also developed larger designs carrying approximately 600 fuel bundles intended for shipment by rail or water. Construction of the large casks will not commence until large-scale transportation is imminent.

3.0 DISPOSAL

Studies to develop a safe concept for the underground disposal of nuclear fuel wastes began in Canada in the early 1970s. During the early phases of the research program, prior to 1975, an evaluation was made of the potential of Canadian salt deposits and other geological media for nuclear fuel waste disposal. A primary consideration in this evaluation was the fact that, because the province of Ontario was, and would

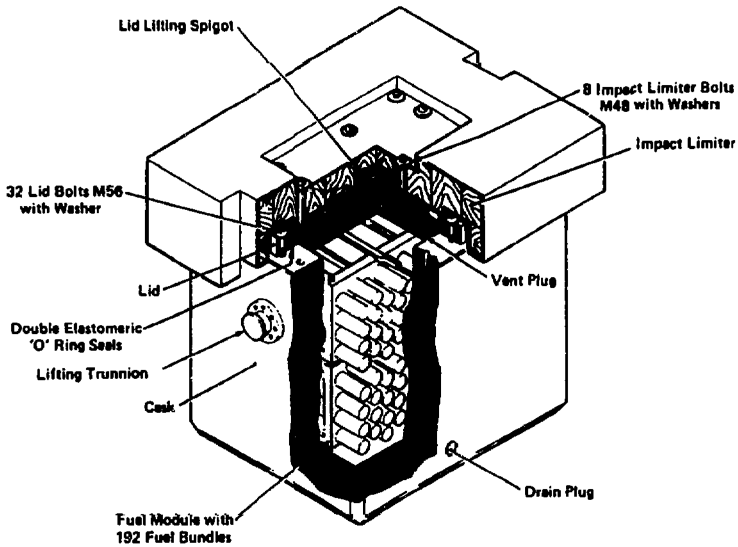

Fig. 4 Spent Fuel Road Cask

continue to be, the principal region in Canada for the development of nuclear power, the first disposal vault would most likely be located there. Therefore, the research program would have to be relevant to an assessment of a disposal concept for Ontario. It was also clear that available resources would not permit intensive research on a number of rock types. Therefore, in 1975 the decision was made to focus research on the concept of disposal in plutonic rock of the Canadian Precambrian Shield, which is predominant over a large portion of Ontario.

In 1978, the governments of Canada and Ontario entered into an agreement to cooperate in the development of technologies for the safe, permanent disposal of Canada's nuclear fuel waste. Under this agreement, Ontario Hydro, the provincially owned utility, is responsible for research on interim storage and transportation, while Atomic Energy of Canada Limited (AECL), a federal crown corporation, is responsible for research on immobilization and disposal. In April 1981, the Canadian government approved, in principle, a ten-year generic research and development program on nuclear fuel waste management. The results of this generic research phase will be submitted to regulatory and environmental agencies in 1988. The formal review of this submission will include public hearings, and is expected to result in a recommendation to government regarding the acceptability of the concept by 1991.

The conceptual disposal facility consists of an engineered excavation (vault) 500 to 1000 m deep in plutonic rock together with the associated surface facilities to handle and package the nuclear fuel waste. The disposal vault will consist of arrays of rooms, each several metres high and wide to receive the waste. The rooms will be connected by haulageways for transportation of the excavated rock, waste containers and backfill materials. Fuel wastes would be isolated in corrosion-resistant metal containers (ASTM Grade 2 titanium is the current reference material) and emplaced in boreholes drilled in the floor of the disposal rooms. A minimum container design lifetime of 500 years has been specified to ensure isolation of the fuel waste during the period of high fission product activity. The containers are surrounded by a compacted buffer material (50 wt percentage sodium bentonite and 50 wt percentage silica sand) that will swell on saturation with groundwater and ensure that transport of contaminants from the containers is controlled by diffusion. After the waste emplacement, the rooms will be mainly backfilled with a mixture of 75 wt percentage crushed and graded host rock and 25 wt percentage glacial lake clay. On completion of vault operations, the remaining volume, including all the shafts and exploratory boreholes, will be backfilled and sealed. Once the facility is sealed, no further actions are to be required to ensure adequate isolation of waste.

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Problems of back end of the fuel cycle are becoming important due to the increasing number of reactor units of VVER type in Czechoslovakia. There are 8 units in operation and another 4 units will be put into operation within five years.

The reactors of VVER 440 type are working in one year cycles with duration of a campaign approximately 290 power days. About one third of the reactor core is exchanged when the campaign ends. The amount of 120 spent fuel assemblies is regularly exchanged. They have an average burn-up of about 29 MWd/kgU.

The spent fuel is stored for the period of three years in "the reactor pools" which are situated in a reactor hall. For the next two years the spent fuel is stored "away from the reactor storage facility" at Jaslovské Bohunice site. After five years storage, according to the regulations set by the USSR in the year 1984, the spent fuel is transported to the Soviet Union.

The worldwide experience shows different approach to this problem due to different criteria which were taken into account, e.g. environmental aspects, low costs of uranium ore and high costs of reprocessing, etc.

A number of small countries have agreed to the long-term storage. Information given in IAEA bulletins and reports indicates, however, that in the near future reprocessing costs will be cut down.

In Czechoslovakia preliminary analyses have been also carried out, considering following possible options about back end of the fuel cycle:

- final storage of spent fuel on the USSR territory without Czechoslovak participation,
- construction of a new re-processing plant or final disposal facilities in the USSR with Czechoslovak participation; in the meantime the spent fuel would be stored in the CSSR,
- final long-term storage of spent fuel in the CSSR.

The choice of one of the above mentioned alternatives for the final solution of the back end of the fuel cycle in Czechoslovakia, which is expected to be realized after the year of 2000, is being considered in co-operation with the USSR.

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Finland

GENERAL

In Finland, about 30 % of electricity is produced by nuclear power. There are four reactors in operation, two PWRs at Loviisa (2 x 445 MWe, start of operation in 1977 and 1980) operated by Imatran Voima Oy (IVO) and two BWRs at Olkiluoto (2 x 710 MWe, start of operation in 1978 and 1980) operated by Teollisuuden Voima Oy (TVO).

The Nuclear Energy Act and Decree, passed in 1988, form the central regulatory basis for the management of spent fuel and other nuclear wastes. Responsibilities, licencing procedures and financing principles are defined in them. The objectives and schedules of nuclear waste management have been defined in more detail in the Government's policy decision of 1983.

The utilities are responsible for the safe management of their spent fuel: research and development work, implementation and financing of spent fuel management. A number of research institutes, universities and consultants participate in the programme as

contractors, such as the Technical Research Centre of Finland, the Geological Survey of Finland and the University of Helsinki.

The progress of the waste management programme is supervised by the Ministry of Trade and Industry. The Finnish Centre for Radiation and Nuclear Safety is responsible for the supervision of the safety of plans and activities. The major facilities are licenced by the Government. Part of the research work is funded by the Ministry in order to maintain independent expertise for the supervision of the activities of the waste producers.

SPENT FUEL ARISINGS

The Loviisa power plant produce annually 28 tU (tons Uranium) of spent fuel. IVO has an agreement with the Soviet fuel supplier on the return of the Loviisa spent fuel to the Soviet Union. Fuel assemblies are stored for five years at Loviisa before transportation. At the end of 1988 the amount of spent fuel stored at Loviisa was 143 tU.

The annual production at Olkiluoto is 45 tU. The amount stored at Olkiluoto was 363 tU at the end of 1988.

INTERIM STORAGE

IVO stores spent fuel assemblies in the water pools of the Loviisa power plant until they are transported to the Soviet Union.

Also TVO stores assemblies in the storage pools of power plant units. In addition, TVO has constructed an interim storage facility (KPA store) at the Olkiluoto power plant site. The three water pools of the store have a storage capacity of 1200 tU. The storage facility was commissioned in 1987.

FINAL DISPOSAL

The alternatives for Olkiluoto spent fuel management after the interim storage phase are

- direct disposal of spent fuel in Finland
- foreign reprocessing and return of wastes to Finland
- foreign reprocessing including waste disposal, or foreign direct disposal services

So far, no agreements have been signed on the foreign services. Preparations are made for final disposal of spent fuel in the Finnish bedrock.

An updated plan for domestic direct disposal of spent fuel was presented to the authorities in 1985. The repository concept comprises horizontal tunnels with vertical holes in the floors at a depth of several hundred meters in the crystalline bedrock. The repository is planned to be constructed in the 2010's.

TRANSPORTATIONS

The spent fuel of Loviisa is transported to the Soviet Union by train in the Soviet wet flasks with a capacity of 30 assemblies.

TVO has one flask for transfer of spent fuel assemblies from power plant units to the on-site interim storage facility. The wet flask has a capacity of 41 assemblies.

RESEARCH AND DEVELOPMENT WORK

The site for final repository will be selected by the year 2000. Field investigations were started at five areas in 1987. The programme consists of airborne survey, deep and shallow drillings as well as measurements and sampling from the surface and in boreholes. Field work is followed by laboratory studies as well as modelling and evaluation activities.

Parallel to bedrock investigations, the technology of final disposal is being developed and optimized. Various long-term experiments are carried out for performance assessments of the repository system.

FINANCES

The utilities have to present annually updated cost estimates for nuclear waste management, including spent fuel, low- and intermediate-level wastes and decommissioning. Based on these estimates, the Ministry of Trade and Industry each year confirms the fee to be paid into a government-controlled fund.

The estimate includes the future costs for the management of the waste amount already produced. The costs are calculated at current prices without discounting. The funded money need not immediately cover the total costs. The difference has to be covered by securities.

Each annual fee has to increase the fund to the level corresponding to the ratio of the cumulative produced electricity and the total production during 25 operation years. By March 1989, IVO has paid MFIM 395 in the fund and TVO MFIM 1479. A utility is entitled in borrowing back 75 % of its own share of the fund.

INTERNATIONAL COOPERATION

The utilities TVO and IVO have signed information exchange agreements with the Swedish SKB, the Swiss NAGRA and the Canadian AECL. Furthermore, the Finnish organisations have a number of contacts with the other waste management organisations especially in the countries studying crystalline rock.

The Finnish utilities, research institutes and authorities are participating in several international research projects, e.g. the Stripa project. There is lively exchange of information also within the expert groups of international organisations, e.g. the IAEA, the OECD/NEA and the Nordic NKA.

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1. INTRODUCTION

France has been confidently pursuing its policy, undertaken for more than thirty years, which consists of immediate reprocessing of spent fuel, associated with recycling of recovered fissile materials (U, Pu) and with waste selective treatment and storage.

This policy for spent fuel management results from :

- the French engagement in a significant electronuclear equipment programme,
- the lack of French energetic resources and the concern for ensuring in the best way the fueling of present and future power reactors, which is a responsible approach with regard to future generations,
- economic evaluations which desmontrate that Pu recycling in PWR is an interesting solution, awaiting for FBR introduction.

The important R&D programmes, undertaken by CEA have allowed to master quite fully reprocessing and waste treatment processes, both on pilot plants and on intermediate size plants. Accumulated satisfactory operation experiences related to different types of fuel (GCR, LWR, FBR) always according to the PUREX process have permitted to demonstrate industrial feasibility and to launch large size plants (UP3, UP2-800). Simultaneously, studies and experiments especially for alpha-bearing and high activity waste storages have been developed owing to medium and long term plannings.

This spent fuel management programme is carried out inside CEA Group, according to an appropriate structure including COGEMA, SGN, CEA and ANDRA.

The present status of the different steps of this programme can be summarized as follows.

2. SPENT FUEL PRODUCTION AND STORAGE

The French nuclear power capacity which contributes for the largest part to the irradiated fuel arising reprocessed in France, has reached 51.4 GWe, corresponding to 48 PWR, 4 GCR and

2 FBR units. Moreover 9 PWR units are under construction, for 13.4 GWe.

Also, most of the spent fuel presently produced in France is PWR fuel, irradiated up to 33000 MWD/t, for a tonnage which amounted to about 800 T (H.M.) in 1988 and will reach more than 1000 T in 1990.

The storage facilities of the La Hague Centre, which are useable both for EDF power stations and foreign customers, have reached a 10000 T capacity, with the commissioning of a new pond in 1988, the E-pond.

3. REPROCESSING

French reprocessing industrial experience for GCR and LWR fuels has been gained through the UP1 (Marcoule) and UP2 (La Hague) plant operations by COGEMA. For FBR fuel, this experience results from :

- . reprocessing in UP2, by dilution with GCR fuel,
- . operations in the CEA specific pilot plant, called APM.

At the present time, UP1 which is characterised by a recent mechanical decladding facility (MAR 400), is dedicated to French and Spanish GCR fuels and has cumulated 3578 T GCR fuel, at the end of 1988.

UP2-400, initially devoted to GCR fuel, has been extended since 1976, with an oxyde unit called HAO and has operated, from that time, until January 1987 by alternate campaigns, either with GCR, or LWR fuel, or even with FBR fuel diluted in GCR fuel. At the end of this period, 4894 T of GCR fuel were reprocessed. UP2-400 has been then entirely devoted to LWR fuel and has been operated at a regular average througput of about 40 T/month, and even up to 50 T/month for several months periods. At the end of 1988, it has reached the cumulated amount of 2440 T reprocessed LWR fuel.

The Marcoule Pilote Plant (APM) has been restarted on January 1988, after an interruption period for refurbishment.
Its normal capacity has been increased to 5 T/year and at this time, new R&D possibilities are available for FBR and LWR fuel.
APM operated a first compaign with FBR fuel, in a check-out objective. Other compaigns are scheduled with LWR fuel, and especially with MOX, then with high burnup LWR fuel, but this does not change its prime vocation for FBR fuel reprocessing, which will lead to a new Phenix fuel campaign during 1989.

As for the two new production units UP3 and UP2-800, which will allow COGEMA to increase its reprocessing capacity in La Hague from 400 to 1600 T/years, with improved safety conditions for operators and environment, and at reduced costs, the situation is as follows :

. the construction of UP3 is in its last step. The plant is planned to start hot tests during 1989 and to be commissioned in 1990. This delay is due to modifications following operation experience of UP2. Indeed, during the last years, UP2 has necessited important changes and improvements, which have allowed UP2 capacity to be increased progressively from 250 T/year to its nominal capacity 400 T/year in 1987 and which have been applied to UP3.

. Hot commissioning of the extension of UP2 (UP2-800) is still scheduled in 1992. This unit devoted to LWR uranium oxyde fuel, will be able to reprocess UO2 fuel with higher burnup and MOX fuel.

4. PLUTONIUM AND URANIUM RECYCLING :

According to the availability of increasing quantities of plutonium, and to technical feasibility studies and tests for Pu recycling in PWR, EDF decided in 1983 to recycle the plutonium issued from reprocessing in its 900 MWe units. The introduction of MOX fuel is scheduled according to progressive steps. A first MOX reloading (8 T fuel, i.e. 16 subassemblies) was operated in November 1987 (St Laurent B1). Two other ones were operated in 1988 and this rate will increase progressively to reach 5, then 10 reloadings/year between 1993 and 1995. In a first period, burnup will be limited to 33000 MWd/T, but the objective is to achieve more than 40000 MWd/T.

5. WASTE MANAGEMENT

In the present realizations, important efforts have been carried out to improve maintenability and safety of components and to reduce effluents and wastes production. Adequate radioactive treatment and conditioning technics are applied in reprocessing facilities (vitrification, embedding in concrete, or bitume, ...). In this field, the vitrification prototype plant AVM has operated in Marcoule since 1978, and has vitrified until 1988, 1265 cubic meters of fission products corresponding to 1605 canisters, each of them containing 150 borosilicate glass liters. This process, in an adapted version, will be used in the T7 and R7 units of La Hague, associated respectively to UP3 and UP2-800, for the vitrification of HLW solutions.

Concerning final waste disposals, the shallow land disposal of La Hague for A-category wastes, including low and medium activity, short lived wastes, will be closed in 1990. A new storage is in preparation at Soulaines and will be available in 1991.

Alpha-bearing (category B) and high activity (category C) waste are temporaryly stored in engineered storages before final disposal :

- four sites for underground storage, relative to different geological configurations, are being studied by ANDRA, one of them will be chosen in 1990, for further investigations and for the realization of an underground laboratory,

- the construction of the deep repositary for alpha-bearing wastes should be engaged about 1995,

- the opening of a deep repositary for high activity wastes remains scheduled around 2010.

For all these areas, CEA carries out important R&D programmes, upstream and in support of present and future realizations.

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Hungary

The history of nuclear research in Hungary dates back to the turn of the century. Substantial R+D activities started, however, only in the mid-fifties. Our research reactor was built in 1959 and opened immediate perspectives to peaceful uses of atomic energy. As to nuclear power, its age started in Hungary with the NPP Paks: four PWR units of the 440 MW WWER type. As to the future, it is planned to set up several reactors of the WWER-1000 type.

1. The research reactor WWR-SM

Our first research reactor (then named WWR-S) started operations in March 1959 and run until 1967 at 2-2.5 MW thermal power level. It used fuel assemblies of the EK-10 type. Each of them contained 128 g of U-235, enriched to 10 per cent. 82 fuel assemblies were discharged, with burnup reaching 25%.

A first reconstruction was performed in 1967. The new core consisted of 36% enriched WWR-SM fuel assemblies, each containing 40 g of U-235; it was surrounded by a metallic beryllium neutron reflector. This core produced a total of 11000 MWdays until May 1986. The average burn-up reached 50 %. During this operational phase, 780 spent fuel assemblies were produced.

The reactor was again taken out of operation in 1986 for its second reconstruction. Henceforth to be called WWR-SM10, it will initially run at the 10 MW level, but in the nineties this will be uprated to 20 MW by means of modernized fuel assemblies.

To the reactor are attached two spent fuel storage ponds: The inner one is situated in the main hall of the reactor. Its stainless steel tank is placed into a pit also provided with stainless steel lining, connected to the reactor tank via a duct. The grid arrangement of this pond is designed to ensure subcriticality under all conditions.

The inner storage pond can accomodate the spent fuel assemblies normally discharged during a two year period plus a total core. It has its own separate cooling system. Upper shielding is provided by 3 m of water and 30 cm of steel.

The external spent fuel pond is situated at a distance of 100 m from the reactor, on site . Its stainless steel tank, sunk into the soil, has a diameter of 2.5 m and is 7 m deep.

The fuel assemblies were placed in this tank on three levels, in storage tubes, each of which can accomodate three 'triple' or nine 'single' assemblies. To prevent corrosion of fuel assemblies during the storage period, the storage tubes are replaced now by hermetic containers of the same size. The containers are equipped with water and air sampling nozzles and their water can be drained and refilled.

Spent fuel is stored in the Institute only temporarily. In the near future the spent fuel will have to be either transported back to the supplier or a new storage pond will have to be constructed. The problem will become acute at the beginning of the nineties.

2. Paks NPP

The Nuclear Power Plant at Paks is equipped with WWER-440 type reactors. The four identical units took up commercial operation in 1983, 1984, 1986, 1987, respectively. The main technical data related to the spent fuel management are as follows:

- Fuel weight in each reactor: 42 t
- Max. enrichment: 3,6%
- Number of fuel assemblies: 349
- Refuelling interval: once a year, one third of the core

In Paks NPP the spent fuel is stored under borated water in ponds located at each unit and connected via a sluice gate to the reactor pits. In the pond heat release and radioactivity of the spent fuel is reduced to values permitting transportation out of the facility.

Two independent technological systems provide cooling of the spent fuel pond water. Corrosion products of the cladding and of pond structural materials as well as other contaminants are removed from the system by water treatment. Due to the high purity of the water and the cleanness of the surfaces this water treatment system has to be run only for a few hours each month.

The ventilation system assures elimination of radioactive gases originating from the spent fuel and from radiolysis of water.

Should the activity of the primary circuit reach specified values during operation, fuel assemblies must be checked for leak-tightness upon removal from the reactor. Until now no inhermetical assemblies were found in Paks NPP.

Originally it was foreseen to keep the spent fuel at the site for three years before reshipping it to the supplier and so the design capacity of each spent fuel pond was 349 fuel assemblies. Later on, storage time increased to five years, according to an agreement with the supplier and therefore the capacity of the spent fuel ponds had to be modified by reconstructing the internal rack structure for compact storage.

The fuel of VVER 440 reactors is of hexagonal cross-section. Therefore the compact rack has a triangular lattice formed by hexagonal tubes. Subcriticality is ensured by geometry and by the rack material that contains 1,1% boron as neutron absorber. The rack was designed and manufactured on the basis of a Kraftwerk Union (FRG) technology in cooperation with Hungarian enterprises. The present capacity of each pond provides 706 places (650 for hermetical and 56 for possibly inhermetical fuel assemblies).

Identity and integrity of irradiated fuel assemblies kept under water can be verified and checked with the aid of an optical device developed specifically for use in spent fuel ponds and in the core. The Hungarian Underwater Telescope (HUT) is a high resolution device with continuously variable magnification. It allows remote viewing and scanning of underwater objects both vertically and horizontally. The greatest water depth can be 20 m. The viewing tube is also water-filled, providing adequate shielding for the viewer. Photo and TV cameras can be coupled to the eyepiece. The complete system includes illumination and supports.

The first reshipment of spent fuel from NPP Paks will take place by rail this year, after completion of the first full five-year storage period. Transport will be performed in accordance with the Regulation for the Safe Transport of Radioactive Material (IAEA, 1985 edition).

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Italy

Recent decisions by the Italian Government have put a temporary stop to the existing nuclear programme. Consequently, the Trino and Caorso power plants have been shut down, the construction of the Montalto station has been discountinued and the EUREX and ITREC reprocessing pilot plants are being closed.

Power Station Fuel

From the beginning of its nuclear power programme Italy envisaged reprocessing the spent fuel arising from its power stations.
All the Magnox fuel from the Latina station (160 MWe GCR) has been and is being reprocessed by BNFL at Sellafield.

Table 1

ITALY : Light Water Reactors' Spent Fuel

STATION	POWER MWe	TYPE	COMMERCIAL OPERATION	FUEL IN CORE,tU	AR STORAGE CAPACITY,tU	SPENT FUEL IN 1988,tU	STORED AR,tU	STORED AFR*,tU	REPROCES-SING **,tU
Garigliano	160	BWR	1964	-	-	68	-	68	-
Trino	260	PWR	1965	35	50	58	23	15	20
Caorso	860	BWR	1981	102	397	117	117	-	-

* Avogadro Facility

** BNFL-Sellafield

Table 1 gives the situation for the fuel from the light water reactors (Garigliano, Trino and Caorso): not all the spent fuel arisings have been reprocessed, because it was not considered economically viable; delayed reprocessing was preferred and the fuel has been stored at the Avogadro Away-from-Reactor facility (capacity: 135tU) in Saluggia and at the reactor pools.
A contract, however, has been entered with BNFL for the reprocessing of 90 tU of spent fuel; within this contract 20 tUs of Trino fuel have been shipped to Sellafield.
Following the interruption of the nuclear programme no decisions have as yet been reached concerning the management of the existing spent fuel.

Prototype and Research Reactor Fuel

With the EUREX (Saluggia) and ITREC (Trisaia) pilot plants Italy could reprocess fuel coming from research reactors (MTR) and prototype reactors (Elk River) and special cruciform PWR fuel from the TRINO station.
These plants are no longer available and it will be necessary to find new ways of dealing with the following fuel:

EUREX :150 MTR fuel elements -152 kgU;

EUREX :52 PWR cruciform fuel elements -1,92 tU;

ITREC :64 Elk River fuel elements -1,69 tUTh

Several possibilities are being considered from reprocessing contracts with foreign companies to dry storage in containers and casks.

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Permanent Mission
of Italy

Japan

1. The strategy of spent fuel management of Japan has its base in reprocessing of spent fuel and subsequent utilization of recovered uranium(U) and plutonium(Pu) as nuclear fuel.

To the end of 1988, Japan operated 35 nuclear power plants with the total installed capacity of 28 GWe. In the year of 2,000, the capacity is estimated to amount to approximately 53 GWe at least and the annual spent fuel arisings will be greater than 1,100 tons per year. In 1987 fiscal year(FY), about 850 tons of LWR spent fuel has been generated. The accumulated quantity of spent fuel by the end of FY 1987(March, 1988) reached about 5,100 tons, of which about 2,000 tons of spent fuel being stored in water ponds at reactor site. The other part of the spent fuel has been shipped for reprocessing in domestic or foreign facilities.

2. Under the policy of reprocessing all of the spent fuel and utilizing recovered U and Pu, a reprocessing facility with capacity of 0.7 ton per day is operated in Tokai-mura by Power Reactor and Nuclear Fuel Development Corporation(PNC). Up to June 1988, about 392 tons of spent fuel, including a small amount(5 tons) of MOX fuel from an advanced thermal reactor(ATR), "FUGEN" has been reprocessed in the plant.

Construction of a commercial reprocessing plant with capacity of 800 tons per year is planned in Rokkasho-mura, Aomori-prefecture. Japan Nuclear Fuel Service Co., LTD is responsible for the construction and the operation. Hot commissioning of the plant is scheduled to start around the middle of the 1990's.

3. On the other hand, most electric companies in Japan have contracts for reprocessing their spent fuel with foreign firms; Compagnie Générale des Matières Nucléaires(COGEMA) of France and British Nuclear Fuels plc(BNFL) of Great Britain. About 4,800 tons of LWR spent fuel is contracted with both COGEMA and BNFL, and 1,100 tons of gas-cooled reactor(GCR) spent fuel with BNFL, for reprocessing. To the end of 1988, about 4,000 tons of spent fuel from both LWR and GCR had been transported to the facilities of COGEMA and BNFL.

4. Plutonium recovered by reprocessing spent fuel will be used most effectively in FBR. A prototype FBR "MONJU" with capacity of 280 MWe is now under construction by PNC, with the target schedule of reaching its criticality in 1992. Pu utilization in LWRs and ATRs is also promoted, since it will take significant length of lead time before the commercial scale deployment of FBR.

As for the utilization in LWRs, a small scale demonstration program is now under way. Under this program, a small number of MOX fuel assemblies were respectively loaded into one BWR in 1986 and into one PWR in 1988.

The pilot-scale R&D of MOX fuel fabrication has been carried out mainly by PNC. More than 90 tons of MOX fuel, mostly for ATR "FUGEN" and FBR "JOYO", has been fabricated at the plutonium fabrication facilities. Another new facility with capacity of 5 tons of MOX per year started operation for "MONJU" in October 1988.

5. The management of the high level radioactive waste, produced by reprocessing, is in the R&D stage in Japan. The waste is presently stored in the form of solution but will be solidified and disposed of in geological formation after cooling the decay heat for a certain period at engineered storage facilities. The construction of a pilot plant for vitrification started in June 1988 for commissioning the hot operation in the begining of the 1990's.

Received from
Permanent Mission
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Korea, Republic of

Nine nuclear power plants (8 PWRs and 1 CANDU), whose total general capacity is about 7.6 GWe. are now in operation or under construction, supplying more than 50% electricity demand in Korea since 1987. And two more 990 MWe class PWRs will be completed in the middle of the 1990's. This nuclear power programme entails the management of spent fuels discharged from plant operation. They are currently stored in each unit's At-Reactor (AR) pools of which total accommodation capacity is about 2.730 MTU. The accumulated amount of spent fuels up to now, since the operation of the first unit (Kori-1) built in 1978, is about 810 MTU and this accommodation is projected to reach about 2.810 MTU by the year 1995, about 3.500 MTU by the year 1997 and about 4.600 MTU by the year 2000. Moreover, the accumulative amount of spent fuel discharged from these 11 units for the whole life time will be approximately 11.500 MTU. This accumulative projection indicates shortage of Full Core Reserve (FCR) storage capacities of those 11 units in mid-nineties except Kori-2.

Under the current situation in which the Korean government has not established a definite policy to either recycle or permanently dispose for the long term management of spent fuels, the best we can do is to store them safely in the meantime. Korean Atomic Energy Commission (AEC) has set forth a resolution (July 1988) that Away From Reactor (AFR) storage be built by the end of 1997 as an interim storage facility to ease out the mid-term spent fuel management problem. Further Atomic Energy Act amended in May of 1986 promulgated the government responsibility for spent fuel management and designated Korea Advanced Energy Research Institute (KAERI) as the responsible body for the construction and operation of the storage facility.

Meanwhile, Korea Electric Power Corporation (KEPCO) will take care of the spent fuel accumulating in the existing AR pools by suitable means up to the time of handover to AFR. Examination of the suitable options for KEPCO's AR storage management shows the possibility of transshipment of Kori-1 excess spent fuel to Kori-3 and 4 at first hand taking advantage of their presence at the same Kori site.

At Uljin site, reracking with poisoned rack system should allow considerable expansion of existing capacities up probably to the end of 1997 as it is planned by KEPCO. Maximum use of existing system of cooling and purification is thought to be adequate without modification, even though further detailed checks would be needed.

For the CANDU fuels at Wolsung site, possible capacity expansion with shorter tray stackup is estimated to be only marginal two more years. Considering the FCR capacity of Wolsung 1 up to early 1992, new option to manage the excess amount up to the target year 1997 is inevitable. KEPCO is presently interested in one of the dry storage concepts and this will probably be implemented in the next year.

AR expansions for spent fuel storage are not the ultimate solution for accommodating the whole spent fuel up to the life time of each unit. KAERI has carried out feasibility studies on this problem for future interim storage strategies.

The interim storage programme by KAERI will soon enter into design phase to be followed by start of construction work on 1993 with the final goal of completion in 1997.

KAERI is proposing the interim storage facility to be wet type with a capacity of 3.000 MTU, but final decision will be made by AEC in the period of 1988-1989 before a definite site, with the consideration of colocation of both the AFR and the low-level waste disposal facilities, that will be selected at the end of 1989.

To implement the spent fuel management programme, transportation system development is an important factor to be integrated in the programme. KAERI has the experience of successfully transporting 4 PWR assemblies with KAERI designed and locally fabricated wet cask. Even though it was the first time to transport the fuel scale PWR spent fuels for PIE purpose, it is an encouraging experience for future spent fuel transportation in that any severe problems have not been encountered in the exercise of loading, transportation and unloading of the spent fuels other than minor technical difficulties.

The major problem by using railway transportation is the distance between the power station sited and railway network and it would require more frequent transport work and a greater number of casks would be needed because of t‥ weight limitation of loaded vehicle on the roads. Some advantageous conditions and given to coastal sea transportation because all power reactor sites are located at the coastal area and thus nearby harbor services can be utilized with some modifications to cover specially designed cask transport ship. An appropriate transportation system will be developed along with the programme implementation of the interim storage facility.

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The U.S. Department of Energy, Office of Civilian Management Radioactive Management (OCRWM) is conducting programs in the following areas related to the storage and management of spent nuclear fuel:

i) Development of spent fuel storage technologies
ii) Generic R&D for spent fuel
iii) Dry fuel rod consolidation

The programs in each of these areas is discussed in more detail below:

1) Spent Fuel Storage Technologies

Dry storage in drywells, silos, vaults, and casks is a primary method for providing additional capacity for spent fuel storage at nuclear power plants. Full-scale demonstrations of two of these technologies, dry storage casks and horizontal concrete silos, are currently under way under cooperative agreements with nuclear utilities and other members of the U.S. nuclear industry, and a third is soon to start. These demonstrations and associated engineering tests provide the U.S. utility industry with information that can be used in the design of dry storage installations and their licensing through the U.S. Nuclear Regulatory Commission.

The demonstration at the Surrey Power Station, under a cooperative agreement with Virginia Power Corporation, resulted in an increase in the number of metal casks being used from five to eight in 1988. Current plans call for the expansion of the number of different cask designs to be used, and continuation of work leading to the utilization of burnup credit in the determination of the safe maximum loading of the casks.

Construction and heater tests were completed for the horizontal concrete silos (NUHOMS) at the H.B. Robinson Facility under a cooperative agreement between the Department of Energy and the Carolina Power and Light Company. The loading of the fuel canisters and testing of the system is scheduled to be completed in 1989.

A preliminary agreement was reached in 1988 with Pacific Sierra Nuclear Associates (PSNA) concerning the fabrication, delivery and testing of a concrete storage cask. PSNA will design and fabricate the concrete cask, and will deliver the cask to the U.S. Department of Energy's Idaho National Engineering Laboratory for testing. The cask is scheduled to be delivered in 1989, and testing is scheduled to be completed in 1990.

2) Generic R&D for Spent Fuel

Generic R&D work in support of spent fuel storage technologies continued in 1988 for the purpose of providing generically applicable licensing information. These studies relate to: 1) the evaluation of the performance of dry storage systems that utilities may implement at reactor sites, and 2) the development of technical bases needed to show that dry storage is a safe technique that will not degrade the integrity of the spent fuel or storage system.

In 1988, the project to examine the behavior of spent fuel in storage continued by initiating tests with LWR fuel rod segments containing intentionally manufactured defects. The testing will determine the extent of fuel cladding degradation and other phenomena with time under a variety of test conditions. The results from these tests serve as input to models being developed to define recommendations of acceptable conditions for storing spent fuel in an anoxidizing atmosphere.

The development and validation of two computer codes, COBRA and HYDRA, was completed. These codes, which will be used by the nuclear industry to evaluate the thermal performance of dry spent fuel storage installations, were submitted to the U.S. Nuclear Regulatory Commission for their acceptance.

Progress was made in 1988 in the on-going project to assemble a base of information for dry storage and rod consolidation that contributes to generic design, licensing, and operations of storage concepts. These data were derived from a multiplicity of laboratory tests, the DOE/utility cooperative demonstrations, and monitoring of dry storage technology experience from outside the U.S.A. Several related reports were completed in 1988 regarding storage of spent fuel, including: characteristics of fuel crud; categorization of failed fuel; cover gas recommendations for LWR spent fuel; recommended temperature limits for zircaloy-clad fuel in inert gases and nitrogen; performance, integrity, storage, and handling behavior of extended burnup fuel; and dry storage licensing issues.

3) Dry Rod Consolidation

This program involves the development and demonstration of a prototypical process line for the dry consolidation of spent nuclear fuel. Equipment for spent fuel rod consolidation in a dry environment was installed at the

U.S. Department of Energy's Idaho National Engineering Laboratory in 1987. By the end of 1987, 48 assemblies (24 metric tons of uranium) had been successfully consolidated. The resulting technical data provided input for the development of prototypical, production-scale, dry rod consolidation equipment. In 1988, a single contractor was selected to fabricate and conduct cold (non-radioactive) checkout of the prototypical dry rod consolidation equipment. Equipment delivery and cold testing are expected to begin during 1990.

In addition to these technical program activities, in response to a requirement of the Nuclear Waste Policy Amendments Act of 1987, the Department of Energy conducted a study of dry cask storage of spent nuclear fuel in the U.S.A., and developed a comprehensive report on the study results. This study was a comprehensive review of at-reactor spent fuel storage requirements, the status and characteristics of dry storage technologies, and the projected impacts of the fuel storage on the overall U.S. Federal Waste Management System in terms of costs, health and safety, and environmental considerations. The Initial Version Dry Cask Storage Study was completed and distributed for comment in September 1988. The Final Version of the Dry Cask Storage Study, which incorporates recommendations of the U.S. Nuclear Regulatory Commission, State and local governments, utilities, other interested parties, and the public, was submitted to the Congress in March of 1989.

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