FR 8803212

COMMISSARIAT A L'ENERGIE ATOMIQUE

INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE

DEPARTEMENT D'ANALYSE DE SURETE

CEA - DAS - 497

RAPPORT DAS N° 497 e

RESEARCH ON PWR SAFETY IN FRANCE

ZAMMITE R. *

IAEA technical committee meeting on
thermal reactor safety research
(Vienna, 14-17 june 1988)

CEA - DAS - 497

RAPPORT DAS N° 497 e

RESEARCH ON PWR SAFETY IN FRANCE

ZAMMITE R. *

IAEA technical committee meeting on
thermal reactor safety research
(Vienna, 14-17 june 1988)

* CEA/IPSN/DAS

Juillet 1988

RESEARCH ON PWR SAFETY IN FRANCE

R. Zammite[1]

Status report for the IAEA Technical Committee Meeting on Thermal Reactor Safety Research Vienna, 14-17 June 1988

PART 1:

CONTEXT AND ORIENTATIONS

The French nuclear safety arrangements form a centralized system characterized by cooperation between the government authorities, their technical advisers and the operators of the installations, especially between the **Commissariat à l'Energie Atomique** (CEA) and **Electricité de France** (EDF). This cooperation in no way contradicts the respective responsibilities of the different parties, in particular those of EDF regarding the safety of its installations and those of CEA as the government's technical adviser and safety analyst. However, it considerably affects the research on reactor safety, which is mainly performed by the CEA Institute for Nuclear Safety and Protection (IPSN), in collaboration with EDF.

For PWRs, the safety preoccupations concerning their development, commissioning and operation can be divided into the following three categories:

A. Safety in design and construction

B. Safety in operation and the control of potential accidents

C. Maintaining safety — aging problems.

The effort consecrated to each category has varied in the past and will continue to do so in the future. At the present stage, emphasis is being given to categories B and C as, due to the standardization of the facilities, most of the design and construction adequacy problems have already

[1] Commissariat à l'Energie Atomique
INSTITUT DE PROTECTION ET DE SURETÉ NUCLÉAIRE

been solved in a generic manner for each of the successive series of units. Obviously, category C will become increasingly important, and category A will become a major preoccupation once more, when it will be necessary to establish the design bases of the reactors which are to replace the existing ones.

The general objective is to attain a sufficient level of safety by means of defence in depth strategy. This principle is applied with allowance made for the variation of the allowable release limits as a function of the accident probability.

IPSN reactor safety research is particularly concerned with fuller understanding and the development of evaluation tools so as be able to assess the adequacy of the lines of the defence in depth. It it, of course, evolving and depends on **questions remaining to be solved** at a given point in time, brought to light in safety analysis, operating experience feedback and the progress already made. For this reason, it should not be interpreted as reflecting the permanent major safety objectives which are essentially reached and must be retained, but rather that of the **additional progress** required to reduce, quantify and assume a residual risk which is already considered acceptable.

The appendix includes tables which indicate, for categories A, B and C, the relationship between the existing research programmes and the questions remaining open that they are intended to solve.

For category B research, which is mainly concerned with accident control, without dwelling on the trends which affected the programmes at the beginning of the eighties and which were in fact latent considerably earlier, it can be globally stated that R&D has been, and continues to be, strongly influenced by the sequel to the Three Mile Island accident. The acquisition of the resources and knowledge necessary for controlling severe accidents has progressively become the principal objective of the work currently being carried out by CEA, much of which is in collaboration with EDF. The French context is characterized by the fact that the knowledge acquired must be applied to a large and homogeneous set of facilities, and hence with pragmatism and prudence. It should not be deduced from this effort to develop procedures for severe accidents that prevention is no longer the first priority. This has always been and remains the principal objective of the defence in depth strategy, which constitutes the basis of the regulatory safety analyses. Beside, most of the new procedures are of an essentially preventive nature, ie they are intended to prevent any damage to the core, even in cases where they apply to situations previously considered to lie beyond the design limits.

As concerns the procedures for last resort safeguarding of a core and limiting of the consequences of severe accidents, it must be admitted that the lack of understanding of the mechanisms of damage to a core in degenerated accident situations, of the progression of molten corium, of hydrogen and steam explosions, or of the routes of fission products released in various forms, was such at the end of the seventies, that the efforts since made have not made it possible to obtain accurate and full answers to all the questions and in all circumstances. This explains why a substantial part of the present programmes are concerned with obtaining fuller understanding in these fields. One highly specific objective is the availability in emergency centres, in the event of an accident, of means of rapidly obtaining diagnostics and prognostics which are as reliable as possible. This necessitates a major effort for qualification and validation of the system of codes used for the **realistic calculation** of accidental releases. The PHEBUS fission product in-pile global experiment project, for example, results from this requirement.

In the second part of this paper, we describe the present situation and the perspectives for research on the management of severe accidents which are occupying the greater part of the research resources of the IPSN (category B).

In category A research, it may be considered that the primary circuit large breach reference accident is satisfactorily covered for existing reactors. The knowledge acquired by means of the PHEBUS large breach and FLASH programmes relating to the behaviour of fuel and the emission of fission products are now processed and can be used as a basis for giving consideration to changes in the design criteria. The validation of the large breach version of the thermohydraulic CATHARE code is however less advanced than that for small breaches, and will require further effort. The quantification of the safety margins offered by the design with regard to external hazards has made significant progress due to more sophisticated modelling of the effects of missiles and external explosions. Nevertheless, in the seismic field, work is continuing with the collection of existing data as well as new data concerning high intensity movements throughout the world (measurement systems are in the process of being installed in California for example), work has also continued with developing a methodology for seismotectonic zoning to better characterize the potential movements at the power plant sites.

In category C research, concerning the retention of safety over the life of the plants, the main research initiative is, at present, devoted to the potential weakening of the primary circuit envelope, particularly as concerns the pressure vessel and the steam generator tubes. Particular efforts have been made to improve the performance and reliability of non-destructive testing. Tests and studies concerning the applicability of the leak-before-break criterion, partly within the framework of the international IPIRG programme, are considered essentially as a means of making progress in evaluation of weakening and the earliness of its detection.

A second research initiative is devoted to evaluating the life span under irradiation of the electronic components and equipment, as well as of the polymer-based cables used in the systems important to safety, particularly in accident cases. Concerning this last point, a test programme is being conducted to determine the effect of accident conditions on the operability of equipment.

Thought is currently being given to the possibility of extending the scope of the studies and research concerning aging to other components and structures important to safety.

PART 2:

RESULTS AND PERSPECTIVES OF RESEARCH AND DEVELOPMENT CONCERNING ACCIDENT MANAGEMENT

2.1 Core cooling faults and associated procedures: primary circuit two-phase thermohydraulics

Electricité de France, the operator, the Commissariat à l'Energie Atomique and Framatome, the vendor, have jointly developed the CATHARE code. This has been operational since 1984. The latest version released (CATH 1 Version 1.3) has been, in accordance with a rigorous quality control process, qualified in some 200 experiments of an analytical nature and checked in a large number of experiments on system loops for the accident configurations of all plausible types.

The code is currently being used in safety analysis studies by the three French partners, particularly in the fields of accidents beyond design limits.

A new version, CATHARE II, featuring new modules and also a number of new numerical and data processing methods, which is also easier to use, is being developed and should be ready within one year.

A User Club links the French partners and members from other countries, with a view to exchanging operating experience and passing on requirements for code improvements to the CATHARE team (about twenty engineers) in charge of its development and all validation work.

Furthermore, at the request of EDF, with the participation of CEA/IPSN, the CATHARE models are being integrated into a simulator designed for training plant safety engineers in both severe accidents, as long as the core is not seriously damaged, and in design studies and safety analyses. This is to be operational within two years.

Of the supporting experimental programmes, the BETHSY loop is particularly noteworthy (simulation of a reactor circuit in 1/100 scale, but using full scale heights), of which the experimental programme has just begun.

2.2 Core damage and associated procedures

The behaviour of a core during damage at the moment at which the re-obtaining of water enables it to be cooled is probably highly dependent upon the kinetics of the cooling process. The tests already performed, and the Three Mile Island accident itself, show this: in the event of certain forms of cooling, there is a risk of large parts of the core collapsing and forming a mass of debris which is difficult to cool, even under water, which begins to melt and progress downwards. Certain aspects of ultimate operating procedures could be affected.

The PHEBUS severely degraded core in-pile programme is designed to make it possible to study these phenomena, particularly to determine whether oxidation of the cladding is a dominant phenomenon or not. Three tests have been carried out which correspond to these two situations, the fuel being brought to a temperature of approximately 1800°C. Four other tests are to follow by mid-1989 in which the following parameters will be varied: degree of cladding oxidation, maximum temperatures attained, kinetics of cooling by steam and by water.

The ICARE code, developed for the purpose, is used in precalculation of the tests and for interpreting the results. It will then be used as a basis for the VULCAIN code of the ESCADRE system.

2.3 Safeguarding of confinement and associated procedures

Whatever the safety approach adopted with regard to the phenomena liable to lead to extreme pressure in the containment, whether slowly or suddenly (particularly explosion and/or detonation of hydrogen), studies of **containment ultimate strength** are necessary. The mechanical problems involved make it necessary to model the behaviour of the structures in the fields in which non-linear phenomena are predominant: plasticity of metals, propagation of cracks, failure of civil engineering materials etc. The work carried out has led to the development of the CASTEM code system, which has been validated by numerous experiments in France. This system correctly predicted the burst pressure of a reinforced concrete containment vessel model on which experiments were performed by SANDIA in the USA during the summer of 1987.

Apart from the capability of withstanding extreme loads, the prior presence of leaks exceeding those specified in the design constitutes a serious risk. In France, particular care has been paid to the means of diagnosing and locating such leaks, as well as forms of action to control them. These different aspects are made allowance for in the drafting of ultimate operating procedure U2. Research and development initiatives are being conducted to determine the retention ratio for radioactive substances on passing through small cracks and openings, in order to obtain far more realistic values than those used at the present time.

Furthermore, Electricité de France and the safety authorities have decided to provide the pressurized water reactors with a system by means of which the operator can avoid failure of the containment in the event of excessive internal pressure. This is the filtered containment venting system which, more specifically, makes it possible, with greater safety, to initiate accident management long before there is any real threat to the integrity of the containment: for example, as the operator knows that, whatever the case, the pressure rise in the containment can be controlled, it could be interesting to inject water to cool the corium to slow down or prevent penetration of the raft. The research and development programme itself, pursued by EDF and CEA/IPSN jointly, includes the following stages for finalization of the sand filters:

- a series of laboratory scale tests to determine the principal properties of the sand and the manner of its use: this programme was completed in 1984;

- a series of tests using the PITEAS loop with a sand bed 80 cm deep as planned for the reactor filter, in a drum with a diameter of 1 m, making it possible to verify the results of the laboratory tests and to investigate different operating conditions in transient and steady-state regimes, this programme was completed in 1986;

- the performance of a few tests on a complete full scale system as built and installed at the reactors: this programme is planned to be carried out at the Cadarache Nuclear Testing Centre between 1989 and 1990.

Should the reactor vessel be breached or ruptured, all, or more likely part, of the corium, will quickly reach the bottom of the reactor pit. Whether or not there is water there, a process of erosion of the concrete will begin, of which the kinetics will essentially depend on the quantities of water and corium present. In addition, there will be an increase in pressure in the containment due to the evolution of non-condensable gases and of steam created by the interaction between the corium, concrete and water, in the event of loss of cooling by spraying. Management of the accident concentrates on avoiding penetration of the concrete foundation raft below the reactor vessel well and/or rupture of the above-ground part of the containment. The principal parameters involved are the rate of erosion of the concrete and the rate of pressure rise in the containment.

As concerns R&D, **the interaction between corium and concrete** has been the subject of major experimental programmes. CEA has contributed to certain of these by the modelling and validation of codes, particularly the German WECHSL code which is now extensively used. On the other hand, few experimental studies of corium cooling capability have been carried out. In France, and elsewhere, experiments have been carried out the possibilities of cooling a bed of debris with sodium. This would appear to be ineffective. The difficulty resides in the feasibility of carrying out experiments in water which are sufficiently representative. Besides, theoretical studies shows that it would be difficult to ensure cooling by submersion alone.

The conditions under which **an explosion or detonation of hydrogen** could take place with hydrogen homogeneously distributed in an atmosphere of a given composition are now quite well known. The remaining questions relate to the kinetics of the forming of a given distribution, the explosion-detonation transient as a function of the local fuel-to-oxidant ratio and the geometrical configuration of the location.

CEA has developed means of calculating the distribution of hydrogen (TRIO code) and the effects of an explosion on the containment vessel (PLEXUS code).

2.4 Evaluation of accidental releases — Management of accident consequences

Study of the short and long term behaviour of radioactive products for accidents of relatively slow kinetics, with a potential for a relatively heavy release outside the installation after one day, constitutes a current objective.

The corresponding phenomena are extremely complex. A considerable amount of research has been carried out, both concerning modelling and supporting experiments. In certain fields, a basic understanding has now been achieved.

A number of programmes of an analytical nature are nevertheless being continued and scheduled in France, of which the most significant are:

- the HEVA out of pile experimental programme for studies of the emission of radioactive products (quantities and physicochemical nature) with preirradiated then reirradiated fuel elements; five tests are currently being performed;

- the PITEAS Aerosol Physics programme for the accurate observation of condensation of water vapour on soluble and insoluble aerosols, the coalescing of the droplets formed and the settling of the aerosols on walls by diffusiophoresis. The PITEAS experimental chamber is a cylinder with rounded ends of 3 m^3 (diameter 1.20 m). Relatively sophisticated instrumentation has been installed. The tests have just begun;

- laboratory experiments with iodine: radiolysis phenomena: division between atmospheric and liquid phases, trapping ratios of gaseous iodine for paints, steels, concrete etc.;

- the TUBA programme, which is planned to start in 1989, concerning the trapping of aerosols in lines, simulating the conditions in the reactor circuits;

- the out of pile experimental study of the emission of radioactive substances during interaction between corium and concrete with, in addition, consideration being given to future tests with real irradiated fuel (energy supplied *in situ* by the fission products themselves).

All these tests, as well as the others carried out outside France of which the results are or are soon to be available, are intended to validate the corresponding codes of the ESCADRE system.

The main research effort is nevertheless concerned with preparation of the **PHEBUS fission product** programme (PH/PF), a series of experiments of an integral type to be carried out in the Phebus experimental reactor from 1990.

The purpose of this programme is to reproduce in a single installation, under conditions as close as possible to those prevailing in a power reactor, the succession of phenomena from the emission of radioactive products outside the fuel to its arrival in the containment and the exterior, after passing through certains part of the primary circuit. Insofar as the problems of scale of the circuits and the volumes reproduced do not compromise the representativeness of the phenomena and it is possible to install suitable and sufficiently accurate instrumentation, it should logically be possible to verify all the corresponding design codes. However, certain phenomena will not be represented, such as the interaction between corium and concrete, steam or hydrogen explosions and direct heating of the containment atmosphere, all of each can contribute to additional emission of radioactive products or their partial re-emission.

For the tests, we will use pre-irradiated fuel which will then be re-irradiated in PHEBUS, in the form of a cluster of 20 pins with a useful length of 80 cm, in order to obtain a representative spectrum of fission products at the moment of the simulated accident. The primary circuit will be represented by the upper internals of the core, by parts of the circuit with pressurizer and pressure relief tank, steam generator tubes and different devices, according to the type of accident tested. The containment will be simulated by an enclosure of approximately 30 m^3.

It is planned to carry out five tests, at a rate of one per year from the end of 1990. The experimental programme, of which the principal orientations are established (typical scenarios of breaches and transients, parts of primary circuit involved, dwell time in the containment etc.), will be laid down on completion of the full description of the experimental circuit itself (the principal characteristics are already known and the complete design should be available in 1989), of which the exact nature depends on preliminary studies concerning both the determination of the phenomena to be represented and the effective simulation capabilities.

The cost of this programme, to be performed between 1990 and 1995, but initiated in 1986, should be in the region of 500 million francs. The first initiatives in cooperation with the countries of the European Economic Community are in progress, a major participation by the Commission of the European Communities being envisaged.

CONCLUSION

The research work carried out throughout the world has already greatly contributed to improving firstly the understanding of the safety margins in design and secondly the phenomenology of severe accidents. Nevertheless, the quantification of important parameters in the latter field remains a problem.

The intrinsic improvement of safety afforded by the relatively recent introduction of the notion of accident management, particularly with the preparation of a guide for action, is based on the knowledge thus acquired but has led to priority being given to certain research and development programmes.

A major effort is to be maintained to properly adapt the management of potentially severe accidents by correct diagnostics of the situation and by prognostic capabilities, as a function of the action possible.

In this same perspective, care must be taken to provide and maintain a very high level of skill, beyond that of the field of research, amongst the experts who would be directly involved in the event of an accident. This approach essentially of inculcating, in all those involved, a safety culture relating to potentially severe accidents, essentially linked to a R&D effort.

The prevention of accidents nevertheless remains the principal preoccupation of those responsible for safety, and the need to durably maintain the level of safety which has now been reached will inevitably lead, in the coming years, to progressively intensifying the R&D effort concerning the problems raised by the aging of power plants.

APPENDIX

TABLE INDICATING THE PRINCIPAL TOPICS IN PRESENT RESEARCH AND DEVELOPMENT CONCERNING THE SAFETY OF PRESSURIZED WATER REACTORS

A. Safety in design and construction

(essentially relating to determining the safety margins with regard to postulated events)

Questions open (1988)	Existing programmes (1988)
1. **External hazards**	
Better characterization of earthquakes to be postulated in different French sites.	• permanent collection and processing of data concerning strong movements (California, Japan etc.) • development of new methodologies for calculation of spectra, • a first approach to establishing sismotectonic zonation
2. **Response of structures and components to external hazards**	• improvement of design tools for structures (containment, primary loop)
a) earthquakes, more accurate calculation of the resistance of structures and weakening of equipment	• behaviour of pipes (elbows): own study and participation in IPIRG programme (leak before break)
b) margins with regard to collapse for different types of missiles	• development and finalization of advanced modules of the PLEXUS and INCA codes (3D codes)

Questions open (Contd.)	Existing programmes (Contd.)
c) Explosions: propagation of shock waves within installations from openings	• development of a calculation code (and subsequent experimental validation)
3. Response of structures and components to internal design basis accident conditions	
• effects of depressurization by large primary breach in internals of reactor vessel and reactor vessel supports	• development of adapted versions of TRISTANA and PLEXUS codes for non-linearities
• temperature and pressure rise in containment in the event of protected large breach	• for reference, programme completed (eg ECOTRA, REBECA, GRUYER code)
4. Leak before break	
Applicability of the concept to primary circuit and to steam-water circuit (possible withdrawl of anti-whip devices)	• tests on pipe sections and elbows • participation in international programme IPIRG
5. Control and instrumentation	
Reliability and performance levels of programmed safeguard systems (microprocessors)	• development of a semi-automatic test in tool (OST), on computer, for programmed microprocessors • studies and tests of local data transmission networks

Questions open (Contd.)	Existing programmes (Contd.)
6. **Response of primary cooling system and core to design basis accident conditions (LOCA)**	
• cooling function: improve understanding of safety margins, operating safety systems and procedures to be implemented	• development of the **CATHARE code** (particularly versions G.B) on the basis of numerous analytical experiments now completed • validation of CATHARE with integral experiments. Particularly the **BETHSY tests**: protected single small breaches.
• behaviour or fuel and core (for reference, programmes completed)	• EDGAR out-of-pile programmes hot and cold (irradiated) • PHEBUS phase II in-pile checking programme (LOCA/GB) • development of CATHACOMB code
• release of fission products (for reference, programmes completed)	FLASH in-pile tests 1 to 5 (fresh and irradiated fuel)
7. **Reactivity accidents**	
Confirmation of the harmlessness of the envisagible reactivity accidents (post-Chernobyl context)	Adaptation of the PHYSURA calculation code (FBR) for calculation of the maximum interaction energy in the event of dispersal of fuel

B. Operating safety and control of accident situations

Questions open (1988)	Existing programmes (1988)
1. Human factor	
• perception of risk by operators	• study of risk perception correlation — operating mode (*in situ* studies and during training sessions)
• validation of diagnostics and operating aid resources	• sessions on simulators diverted to the application of procedures A, H and U (full scope 900 and 1300 MWe PWR simulators)
• ergonomic assessment of control rooms	• study of cognitive aspects of man-to-man communication and of organization
• rough evaluation of human reliability	• use of the expert system technique to model the operational capability of an operating team
2. Loss of core cooling and associated procedures (prevention of core meltdown)	
• validation of early procedures H (H1 to H4) and U3	• development and validation of CATHARE code (v. A.5)
• validation of procedure U1 and the state-oriented approach	• BETHSY programme (system loop) to validate CATHARE, validate physical basis of procedure determination, study of special cases (v. A.5)
	• development of an advanced study simulator for all the systems, enabling interactive study of all accident situations necessitating operator action

Questions open (Contd.)	Existing programmes (Contd.)
3. **Damage to core and associated procedures**	
• phenomenology of core damaged	• PHEBUS core severely degraded in-pile test programme (PH/CSD) simulating the behaviour of given core areas for 4 typical accident scenarios and certain final cooling modes
• possibilities of ultimate cooling of core-corium in reactor vessel	• development and validation of ICARE code (core severely damaged) • theoretical cooling studies of corium
4. **Safeguarding of confinement and associated ultimate procedures**	• development of an adapted version of the TRIO 3D code for the distribution of hydrogen
• risk of hydrogen explosion	• improvement of means of calculating effects of detonation and of explosion
• cooling of corium outside reactor vessel and risk of violent corium and water interaction	• theoretical study of risk of interaction between corium and water • hydraulic behaviour of a self-heating bath
• calculation of pressure and temperature changes in the containment	• validation of JERICHO code on results of LACE, DEMONA and MARVIKEN tests
• ultimate strength of containment and associated pressure and leakage	• theoretical and experimental study of leak rates at special points and at penetrations

Questions open (Contd.)	Existing programmes (Contd.)
• interaction between corium and concrete: long and medium term behaviour	• participation in the ACE programme • preliminary study of an in-pile experiment
• filtered containment venting (procedure U5)	• full scale validation of filtered containment venting system
Accidental FP releases and management of accident consequences	
• release of fission products outside core in severe accident	• **HEVA out-of-pile test programme** on irradiated pin sections
• behaviour of aerosols	• **PITEAS AEROSOL out-of-pile tests** on changes in a spectrum of particles in a damp atmosphere, validation of the AEROSOLS/B1 code
• behaviour of iodine	• laboratory tests on the forming of molecular iodine by radiolysis, on the hydrolysis of the iodine and its retention by concrete and painted surfaces and modelling — participation in the ACE programme
• validity of calculation resources for fission products transfers and releases in severe accidents	• **PHEBUS fission products project (PH/PF)** experiments for overall validation of accidental discharge design code systems (in pile) • development and validation of the **ESCADRE code system** for the evaluation of accidental discharges

Questions open (Contd.)	Existing programmes (Contd.)
• Conditions of implementation of containment filtered venting procedure U5	• full scale validation of results of PITEAS filtration programme and development of procedures for full scale use of the system
• evaluation — prediction of accident situations at emergency centre	• **SESAME Project**: development of means of evaluating and predicting the availability of systems, the condition of the core and the primary circuit and the state of the containment. Creation of an expert system for aid in analysis and decision making
• possibilities for accelerating the recovery of contaminated soil after a PWR accident	• **RESSAC Programme** of laboratory tests, at large scale and in situe, to evaluate the potential nuisances, transfer from soil to plants, and development of countermeasure and decontamination techniques

C. Aging — Weakening of equipment and structures

Questions open	Existing programmes
1. Primary system enclosure	
• embrittlement of reactor steel by irradiation: accurately quantify the effect of the parameters, optimize surveillance	• irradiation programme in OSIRIS reactor • study of rupture criteria • study of the role of phosphorus
• conditions of weakening of components showing faults	• development of calculation tools for dynamic cracking, cracking in plastic media and crack stability
• improvement of non-destructive test performance – ultrasonic technique (reactor vessel): stainless steel pipes – eddy current technique (distorted zones of steam generator tubes)	• development of a method of detecting inter or transgranular cracks associated with corrosion starting on the insides of tubes • processing of ultrasound signals relating to flat defects • processing of eddy current signals in distorted zones and development of a point sensor • participation in expert examination of SURRY 2 steam generators • contribution to PISC 3 programme

Questions open (Contd.)	Existing programmes (Contd.)
2. **Control and Instrumentation — Protection system: determination of test conditions**	Experimental studies in irradiation enclosures and thermodynamic test enclosures on:
• aging conditions of optical fibres	• the effects of low dose radiation, of illumination, of temperature (fibres)
• aging of cables (polymers) determination of test profiles	• changes in mechanical properties of oxygen consumption and the effect of electrical power (cables)
	• the effect of beta radiation

DESTINATAIRES

DIFFUSION CEA

M. le Haut Commissaire
DSE
DDS
IPSN
IPSN : M. CANDES
OSSN : M. GUILLEMARD
DRSN : M. BUSSAC
DRSN : M. PELCE
DAS/DIR
DAS/SASICC
DERS Cadarache
SES Cadarache
SERE Cadarache
SESRU Cadarache
SRSC Valduc
SEMAR
DPS/FAR + DPS/DOC : Mme BEAU
DPT/FAR
DSMN/FAR
CDSN/FAR : Mme PENNANEAC'H
UDIN/VALRHO
DEDR Saclay
DERPE/DIR Saclay
DRNR Cadarache
DRE Cadarache
DEMT Saclay
DMECN/DIR Cadarache
DRE/STT Grenoble
DRE/SETH Grenoble
Service Documentation Saclay : Mme COTTON (3 ex.)
Service Documentation Cadarache : Mme REY

DIFFUSION HORS CEA

Secrétariat Général du Comité Interministériel de la Sécurité Nucléaire
Conseil Général des Mines : M. DE TORQUAT
Service Central de Sûreté des Installations Nucléaires : M. LAVERIE (+ 3 ex.)
Service Central de Sûreté des Installations Nucléaires - FAR
Monsieur le Président du G.P.d. : M. GUILLAUMONT
Direction Générale de l'Energie et des Matières Premières : Mlle TISSIER
FRAMATOME : M. le Directeur Général
NOVATOME : M. le Directeur Général
TECHNICATOME : M. le Directeur Général
TECHNICATOME : Service Documentation
EDF / L'inspecteur général de sûreté et de sécurité nucléaires : M. TANGUY
EDF / SEPTEN (2 ex.)
EDF / SPT
VIENNA INTERNATIONAL CENTRE LIBRARY : Mrs ROSLYN M. STIRLING
M. HOHLEFELDER } Bundes Ministerium für UMWELT, NATURSCHUTZ
M. BREEST } und REAKTORSICHERHEIT - BONN (RFA)
M. KREWER - Bundes Ministerium für Forschung und Technologie - BONN (RFA)
M. BIRKHOFER - Gesellschaft für Reaktorsicherheit - KOLN (RFA)
M. JAHNS - Gesellschaft für Reaktorsicherheit - KOLN (RFA)
M. HAUBER - U.S.N.R.C. - WASHINGTON (E.U.)
M. BECKJORD - U.S.N.R.C. - WASHINGTON (E.U.)

.../...

M. J.S. MACLEOD - Nuclear Installations Inspectorate - LONDON (G.B.)
M. GONZALES - Consejo de Seguridad Nuclear - MADRID (ESPAGNE)
M. José DE CARLOS - Consejo de Seguridad Nuclear - MADRID (ESPAGNE)
M. C. BORREGO - Département de l'Environnement - Université d'AVEIRO (PORTUGAL)
M. E. HELLSTRAND - STUDSVIK ENERGITEKNIK AB -
Nuclear Division, Safety and System Analysis - NYKOPING (SUEDE)
M. NASCHI - Direttore Centrale della Sicurezza Nucleara e della Protezione Sanitaria -
ENEA - ROMA (ITALIE)
M. P. VANNI - Direttore relazioni esterne e informazione -
ENEA - ROMA (ITALIE)
M. ZHANG YU MAN - National Nuclear Safety Administration (CHINE)
M. MA FUBANG, Director of the Nuclear Electricty Office - MIN (CHINE)
M. Itsuro MISUMI - MITI (JAPON)
M. EIICHI TSUJI - Science & Technology Agency -
Director of the Nuclear Safety Division (JAPON)
M. ICHIHASHI - Science & Technology Agency - Nuclear Safety Division (JAPON)
M. FUKETA - JAERI - Center of Safety Research (JAPON)
M. Mohamed KARBID - Secrétaire général du Ministère de l'Energie et des Mines (MAROC)

M. CHAVARDES (Attaché près de l'Ambassade de France aux Etats-Unis)
M. MORIETTE (Attaché près de l'Ambassade de France au Japon)
M. LALERE (Conseiller nucléaire auprès de l'Ambassade de France en Chine)

COPIE (SANS P.J.)

SRDE
LEFH
BAIN
GCSR
SASR
SACP
SAEP
SGNR
SAREP
SAPN
SASLU
SASLU/VALRHO
SEC
SAET
SAED
STAS
SASC
SAEG
SAM
SPI

M. GOURIEVIDIS (Conseiller Nucléaire près de l'Ambassade de France en RFA)