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ACCIDENT MANAGEMENT ON FRENCH PWRS

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1. <u>FRENCH SAFETY RATIONALE</u>

1.1. <u>Design bases of French PWRs for radioactivity retention in plausible situations</u>

The French Nuclear Power Plant Program is based on the design, construction and operation of identical series of PWRs. The only differences to be found in reactors of the same series involve adaptation to the site.

American reactors under construction were used as a reference for the construction of the first French reactors (Beaver Valley for Fessenheim, North Anna for Bugey). At this stage, Electricité de France and the French safety regulatory authority essentially based themselves on American safety regulations (10 CFR 50 and Regulatory Guides) for ensuring and evaluating reactor safety.

The experience gained in the operation of the two Fessenheim and four Bugey units formed the basis for the design of the 900 MWe reactor series (CP1 and then CP2), the 1,300 MWe reactor series (P4 then P'4) and now the 1,400 MWe reactor series (N4).

In that way, accident prevention relies essentially on a deterministic approach, the objective of which is to demonstrate that, in the situations considered plausible (normal

operation, incident and accident situations), the retention of the radioactive substances is sufficient. Confining radioactivity is provided by "barriers" and the situations to be allowed for result from application of a "defense-in-depth" concept.
In French-built PWRs, three barriers between the radioactive substances and the plant staff and the general public are schematically considered : the cladding of the fuel, the pressure boundary of the primary system and the containment. The integrity of the barriers is checked for normal operation and for the incident and accident situations considered plausible. Radioactive substances can only be released if all three barriers fail.

The "defense-in-depth" concept classically involves three levels.

- Prevention by quality

The design, fabrication and operating range of the equipment are to be such as to provide the installation with sufficient safety margins with regard to specified limits, to ensure its proper behavior.

- Monitoring and protection

The installation is to be equipped with monitoring and protection systems aimed at restoring it to its normal operating range in all foreseeable transient and incident cases.

- Safeguarding

Regardless of the above preventive and protective measures, plausible accidents are to be allowed for, and safeguard systems have to be designed to limit the consequences of such accidents.

When applying this concept, the following points must be borne in mind :

1) The fact that a component or system is designed for a given situation does not mean that its failure in that situation can be disregarded. If the consequences of such an event are considered unacceptable, additional provisions must be made to mitigate or prevent them. In this way, appropriate arrangements are made to ensure that the pressure systems can withstand the maximum stress to which they are liable to be subjected, the case of their failure nevertheless being given consideration in accident studies. No exception to this rule is allowed unless the risk is sufficiently minimized by adequate preventive measures. The catastrophic failure of the reactor vessel of a pressurized water reactor is thus excluded in the light of precautions taken during its design and fabrication, and of the tests carried out during the service life of this component, to ensure timely detection of any faults which may be forerunners of more serious failures ; in addition, specific regulations apply to this component which is the subject of special scrutiny by the relevant government body.

2) As it is not possible to examine all the accident situations considered plausible, operators and safety authorities have agreed to examine a limited number of them, selected as being representative of the risks. Each situation is chosen and studied in such a manner that its consequences are conservative compared to those of the events of the same nature that are intended to be represented ("envelope accident" approach).

3) It is necessary to identify the failures which can simultaneously jeopardize the arrangements made to prevent the accidents and mitigate their consequences ; provisions have been made to avoid such failures being the origin of unac-

ceptable consequences. In this way, the total failure of the onsite and offsite power supplies could lead to a LOCA (leakage at primary pump seals) which the safeguard systems, having no power, would be unable to compensate for. Similarly, fire can be a source of "common mode" failure. Here the problem is to decide how far to go, and what accident situations are to be allowed for in designing the installation. In the deterministic approach, a conventional list of situations is usually established, such situations being grouped in frequency categories : the lower the probability of occurrence of a category, the higher the upper limit for the corresponding radiological consequences.

For each site, French regulations require authorizations for the gaseous and liquid radioactive effluent releases ; these authorizations set the maximum admissible global activities for the releases on a case-by-case basis and specify the limits of activity for some radioactive species. Conversely, French regulations do not set limits on the equivalent doses likely to be received by the public under accident conditions. The radioactive consequences induced by the conventional operating conditions and conditions resulting from external events are calculated without reference to upper limits of dose equivalents, but their assessments are submitted for each unit to the safety authority for approval and are generally deemed acceptable during the licensing procedures in which the agreement of the Ministry of Health has to be obtained. Nonetheless, for the existing plants and the plants under construction, Electricité de France has proposed the following limits, which have been accepted by the safety authority for PWR design purposes.

Frequency Category	Estimated Frequency (per year)	Maximum Radioactive Consequences
1 Normal operation	1	limited by the radioactive waste release authorizations
2 Minor but frequent incidents	10^{-2} - 1	
3 Unlikely incidents	10^{-4}-10^{-2}	0.5 rem (whole body) or 5m Sv 1.5 rem (thyroid) or 15m Sv
4 Hypothetical accidents	10^{-6}-10^{-4}	15 rem (whole body) or 0.15 Sv 45 rem (thyroid) or 0.45 Sv

Appendix 1 gives the conventional list of operating situations chosen for standardized 1,300 MWe nuclear units.

1.2. Complementary probabilistic approach

1.2.1. Safety objectives

The probabilistic approach was first used in France to define safety measures to be taken against external events. This approach was used to establish a relationship between such events which had to be taken into account in the plant design and conventional operating conditions.

For example, on the basis of a probability analysis of an aircraft crashing on a PWR, the French safety authority accepted that the various series of reactors be designed solely to withstand the crash of a general aviation aircraft (based on a 1.5 ton Cessna 210 as a "hard" projectile and a 5.7 ton Lear Jet 23 as a "soft" projectile), therefore not taking into account the risks deriving from military and commercial aircrafts ; such a policy may result in deleting some possible PWR sites.

As early as 1977, an examination of the general technical options for the 1,300 MWe PWR series led to set forth an overall probabilistic objective in the following terms :

"The design of a nuclear unit comprising a PWR should be such that the overall probability that the said unit can induce unacceptable consequences will not exceed 10^{-6} per year".

"From hereon, when a probabilistic approach is to be used to assess whether a group of events should be allowed for in the design of a unit, it should be assumed that this group of events must be allowed for if the probability that it may lead to unacceptable consequences exceeds 10^{-7} per year ; such a threshold cannot be exceeded for the said group unless it can be proved that the calculation of the relevant probabilities is sufficiently conservative".

"Moreover, Electricité de France has to pursue its efforts to extend as early as possible the use of probabilistic approaches for the broadest possible range of events".

"In application of the above, Electricité de France shall examine on a case-by-case basis, whether the simultaneous failure of the redundant files of the systems essential to safety should be taken into account in the design of power units using PWRs ... For these studies, "realistic" assumptions and calculations methods may be used".

Such statements have to be supplemented by the following comments for clarification.

1) The overall objective is set forth in terms of "unacceptable consequences" ; in accordance with the above, these "unacceptable consequences" are not defined by any legislative or regulatory text. In fact, such consequences are to be assessed in political terms, taking into account site-related effects and the possible impact of measures aimed at protecting the general public.

2) The probability of 10^{-6} per year is a "target" value for a reactor and Electricité de France was not required to demonstrate that such a target value is really met ; nevertheless, this objective was considered reasonable, based on the results of the WASH-1400 Report and on the improvements made in the design of French reactors with respect to the PWR power plant examined in this report. The justification of the design provisions adopted to prevent any unacceptable risk still relies heavily on deterministic analyses rather than on an overall probabilistic analysis.

   In this regard, in a letter addressed to Electricité de France in 1978, the Ministry of Industry clearly specified the framework for the probabilistic analyses required from Electricité de France :

   "I emphasize ... that my concern to extend the use of probabilistic analysis to the greatest possible number of groups of events does not imply the direct use of this approach for the design of pressurized water reactors. Probabilistic evaluation may be run afterwards to show that the assumptions made for the design provisions are well founded, and may furthermore be used, if need be, to improve the definition of the deterministic criteria used for the design of future reactors".

"Neither do the terms of my letter ... (of 1977) ... imply that the safety of a pressurized water reactor be demonstrated today through an exhaustive probabilistic analysis. Conversely, the use of a probabilistic approach should allow a better justification, or even an improvement, of the definition and classification of the events taken into account in the design of a pressurized water reactor".

3) The value of 10^{-7} per year is more directly used in an operational way ; the above-mentioned approach regarding external events does use this value by considering for example several groups of events for aircraft crashes : the probability of a general aviation aircraft crashing on a nuclear power plant in France is such that provisions are taken to protect nuclear units systematically, wherever they are located. On the other hand, the probability of a commercial airliner crashing on a reactor in France outside airport approach areas is low enough to obviate the need for protective measures against this type of crash. Regarding military aviation, the matter is examined for each site to make sure the site is indeed suitable for a PWR power plant. The value of 10^{-7} per year can also be used for treating problems involving combinations of external events and conventional operating conditions.

   Nonetheless, it should be underlined that the value of 10^{-7} per year is no longer considered a "cut-off" value, above which design provisions must be automatically made. The question of whether or not such design provisions are to be made is examined on a case-by-case basis by making a criticism of the assumptions made, based on the following two major considerations :

   a) the overall risk objective : for example, so as to remain within the external event field, greater vulnerability can be attributed to aircraft crashes given a lesser vulnerability to explosions ; in other words, the number of families of events which lead to unacceptable conditions and have a frequency greater than 10^{-7} per year has to be taken into account.

b) the cost of the extra design provisions envisaged versus the expected benefit as regards safety.

4) In contrast with the conventional deterministic approach, which is based on conservative assumptions and calculations regarding conventional events, the probabilistic approach emphasizes the use of values as realistic as possible for estimating both probabilities and consequences, in order to be fully beneficial and to bring about an improved consistency in the provisions made for preventing the unacceptable from happening.

1.2.2. Implementation of the probabilistic approach on French PWRs

As stated above, in France, the probabilistic approach has been directly applied to the assessment of the measures to be taken regarding external events for which the probability of occurrence can generally be assessed. Such an approach was used for determining the external events to be adopted for the design of the Gravelines nuclear power plant, located close to a large crude oil storage, nearby an oil terminal and not far from a projected LNG terminal. In particular, the probability of the explosion of a drifting gas cloud close to the NPP, resulting in unacceptable consequences, has been assessed and led to adopt further design provisions with regard to external explosions.

Moreover, the probabilistic approach has also shown the necessity of complementary provisions to ensure a satisfactory level of safety for some situations which are not included in the list of conventional situations. The 1977 letter of the

Ministry of Industry to Electricité de France requested thorough examination of the probabilities and the consequences of :

- anticipated transients without scram
- total loss of the ultimate heat sink
- total loss of electrical power supplies (off- and on-site).

On completion of these studies, additional arrangements were effectively implemented on the different series of standardized units with allowance made for the state of progress of their construction.

More generally, in order to meet the safety objective set forth, the examination of the probabilities and the consequences of the total loss of redundant and safety-related systems was required. Such studies have shown the necessity of additional measures to complement the automatic systems normally provided by the initial design.

This necessity has led to the definition and development of specific operating procedures designated as the "H Procedures", which will be further examined hereunder.

2. REACTOR OPERATION AND SEVERE ACCIDENT MANAGEMENT

The first domain of application of operating procedures on French PWRs under incident or accident conditions was mainly linked to conventional design basis situations ("I" and "A" procedures) ; this initial domain has been progressively extended to a series of events not allowed for in the original design, but which were identified as requiring consideration under the probabilistic approach ; the procedures to be applied to this new area are the "H" procedures. All the above event-oriented operating procedures were then supplemented by a symptom-oriented ultimate procedure ("U1" procedure) for preventing core melt and three ultimate procedures ("U2", "U4" and "U5" procedures) to mitigate the consequences of a core melt, each one addressing a preferential containment mode of failure.

2.1. The "I" and "A" operating procedures for conventional incident or accident situations

The "I" and "A" operating procedures are essentially related to design basis situations ; such procedures have been defined by considering each failure of active/passive components used in normal operation, liable to jeopardize a major safety function such as :

- the control of the nuclear power,
- the evacuation of the core energy,
- the confinement of fission products.

The consequences of such accidents are limited by the operation of protection and safeguard systems.

The rules for treating these situations, in particular the single failure criterion, result in redundant safeguard systems (2 x 100 %), which are emergency power supplied. "A" procedures are linked to breaches (A1 for primary system breaches, A2 for those on the secondary circuit and A3 for breaches at the interface of the two circuits) whereas "I" procedures address partial failures of support systems (electric sources and compressed air). "I" and "A" procedures are event-oriented, which implies that a diagnosis of the accident sequence in course has to be made before any initiation of such procedures.

2.2. The "H" and "U3" event-oriented procedures for situations at the limit of the design

The first results of the probabilistic studies performed by EdF in 1978 showed that the probability of unacceptable consequences associated with situations of total loss of redundant safety-related systems was higher than the safety objective, considering, in this particular case, that the core melt was inducing unacceptable consequences. The safety authorities asked EdF to propose design modifications and adapted procedures to reduce this risk to acceptable values. This resulted in the definition and the development of the five following "H" operating procedures :

H1 for the total loss of the ultimate heat sink

H2 for the total loss of steam generator feedwater (normal and auxiliary)

H3 for the total loss of electrical power supplies (off- and on-site)

H4-U3 for the mutual back-up of the spray system and emergency low pressure injection system during the recirculation phase

H5 for the protection of sites along rivers against floods exceeding the reference level (millennial flood).

The initial letter "H" stands for "hors dimensionnement", that is "beyond design" : actually, the designation "at the limit of the design" would be more appropriate.

2.2.1. Total loss of the ultimate heat sink : H1 procedure

The H1 procedure specifies the actions to be taken in the event of a failure of the component cooling system, the residual heat removal system or the cold source itself, following the failure of the pumping station or the failure of the four service water pumps. If these events are initiated at full power, there is a risk of damaging the primary pump seals, which could induce a small break and lead to a core melt due to the unavailability of the safety injection pumps.

The purpose of the procedure is to bring the plant to a stand-by situation (T $\leq$ 180°C, primary pressure $\leq$ 45 bar), which allows the cooling injection to pump seals to be shut down with no risk of damage. To reach this stage, the operator cools down the primary system using the steam generators, the steam being dumped to the atmosphere.

The water reserves on site and the procedures specified to resupply the auxiliary feedwater tank allow considerable time for repair of the heat sink (one month).

If these events are initiated at shutdown, the loss of heat sink leads to a total loss of reactor cooling and accordingly to core melt. The purpose of the procedure in these situations is also to bring the plant to a standby condition where the reactor cooling is provided by a steam generator (availability of one or two steam generators is required by the technical specifications). When the primary system is open, as in the case of maintenance during shutdown, it may be necessary to start injection by the chemical and volumetric control system, taking water from the refueling water system tank, because the steam generators are not operational. Resi- dual heat is evacuated to the atmosphere by keeping the containment open.
All these actions are described in the H1 procedure which is available on all the 900 MW and 1,300 MW plant sites.

2.2.2. Total loss of the steam generator feedwater : H2 procedure

This situation results either from the failure of the main feedwater system, followed by the failure to start the auxiliary feedwater system, or from the failure of the auxiliary feedwater system when it is in operation.

Due to the high pressure in the primary system, the safety injection system is inefficient, and this situation would lead to core melting.

The H2 procedure consists of a voluntary opening of the operated valves of the pressurizer before the steam generators are completely dried. The safety injection signal, triggered by depressurization, is confirmed manually. An estimated time of

40 mn is available to the operators to make such a decision. The standby condition to be reached consists of evacuating residual heat via :

- the auxiliary feedwater system, if it has returned to operation,
- the shutdown cooling system, as soon as permitted by the temperature and pressure of the primary system.

The H2 procedure is available on all 900 MWe and 1,300 MWe sites.

2.2.3. Total loss of electrical power supplies : H3 procedure

This situation results either from the loss of external power supply, followed by the loss of the two diesel generators, or from the loss of 6.6 kVolt electric switchboards.

At full power, these events could lead to the damage of the primary pump seals due to the loss of cooling and water injection on these seals.
This situation would result in a small break and lead to core melt due to the unavailability of the safety injection system.

The major objective of the H3 procedure is to maintain the water injection on the pump seals by using another small pump powered by a turbine-generator driven by the steam of the steam generators (Fig. 1). This turbo-generator also produces power for the control of the plant. In addition, one gasturbine or one additional diesel generator is installed on each site, which can be on line three hours after the beginning of the accident.

In reactor shutdown situations where the reactor is cooled by the residual heat removal system, these events would lead also to core melt due to the loss of reactor cooling. The procedure consists of cooling the reactor by the steam generators whenever possible.

The processes specified in the H3 procedure for 900 and 1,300 MWe PWRs have been justified by a probabilistic study of the risks resulting from a failure of the emergency power supplies ; these studies allowed for the various states of the unit and for failures of power sources and switchboards. The following results were obtained :

	Risk of unacceptable consequences* without H3	with H3
900 MWe Reactor	$1.1.10^{-5}$	$1.3.10^{-7}$
1,300 MWe Reactor	$4.7.10^{-6}$	$7.2.10^{-8}$

(*) The term "unacceptable consequences" in the study means core uncovering.

All the design changes related to the H3 procedure have been taken into account during the construction of the 1,300 MWe plants and have been implemented upon for the 900 MWe plants.

2.2.4. Loss of the safety injection system in the recirculating phase : H4-U3 procedures

After a LOCA, when the break cannot be isolated and when the residual heat removal system is not available, the long-term decay heat removal is ensured by recirculating borated water from the containment sump by means of the low pressure injection pumps ; the heat transferred from the core to the containment is evacuated to the cold source by the containment spray system heat exchanger. Taking into account the fact that this situation can last for months, the probabilistic studies showed that is was necessary to improve the reliability of required functions by increasing the redundancy of the pumping systems after a few days.

The idea was, in case of total loss of the containment spray system (CSS) pump, by using connexion sleeves between the low pressure safety injection system (LPSIS) and the CSS, to use LPSIS pumps to assure the functions of the two systems and vice versa.

For the 1,300 MWe plants, flanges are installed on the pipes of the systems and the connecting sleeves can be installed after a period of 15 days after an accident. In addition, a mobile unit including one pump and one heat exchanger can be installed 15 days after an accident in case of the loss of all pumps and CSS heat exchangers.

For the 900 MWe plants, this modification is nearly completed.

2.2.5. Protection of the sites along rivers against floods exceeding the millennial flood : H5 procedure

This procedure allows for a flood 15 % higher than the millennial flood. Such an event would result in the loss of external power sources and of the heat sink about three days. An advance warning of the flood, provided two days before reaching the millennial level, makes it possible to put in place mobile means aimed at protecting necessary material and to bring the NPP to a safe standby state, depending upon its initial state.

2.3. The U1 symptom-oriented ultimate operating procedure for core-melt prevention

The objective of the measures described in chapter 2.2 is to attempt to fulfill the overall safety goal in the particular case of the loss of redundant safety related systems. However, all the design measures taken at the conception level may be inadequate due either to multiple equipment failures or to inappropriate previous actions.

In order to attempt to stop the development of potentially serious situations which could lead to core degradation, after the TMI-2 accident, EdF has proposed a new approach, based on the characterization of every possible cooling state of the core, which will provide an exhaustive coverage of all accident situations. Such an approach, which necessitates a water level measurement in the vessel, already installed on all 1,300 MWe plants, has been implemented for the start-up of the Penly and Golfech plants, now in progress, and will be achieved progressively on all other 1,300 MWe plants and 900 MWe plants. This delay is due to the time needed to develop the corresponding set of procedures and to train the operators.

Nevertheless a limited application of the state approach has already been implemented on the 900 MWe and 1,300 MWe plants where the U1 procedure is used by the safety engineer in accidental situations.

Figure 2 gives a description of the organization of the work in the control room between the operator team and the safety engineer. The safety engineer is called to the control room in case of shutdown or loss of subcooled margin. He is in charge of post-incident supervision and carries out monitoring of criticality, primary and secondary parameters, safety injection and containment spray systems and containment activity. The safety engineer, using given criteria, can decide to adopt the U1 procedure, which specifies the actions for each of the nuclear steam supply system (NSSS) states defined by functional and by physical criteria. The U1 actions are performed by the operator team, and during this time the safety engineer is in charge of permanent ultimate supervision to verify the efficiency of the actions.

In addition to the preparation of the U1 procedure and the appointment of safety engineers to ensure a human redundancy, the lessons drawn from the Three Mile Island accident led to a great number of modifications of the installations and more especially to re-examine the man-machine interface, even for the units already built.

2.4. The U2, U4 and U5 ultimate procedures for the mitigation of the radiological consequences of a severe accident

The principle of incorporating into French PWRs ultimate procedures devoted to the mitigation of the radiological consequences of severe accidents was accepted in 1981 by the involved parties - the Safety Authority and the utility - in order to meet a requirement which can be summarized as follows :

- in case of a core melting, the third barrier, i.e. the containment and the various systems passing through it, must constitute an ultimate line of defense, which must reduce the radioactive releases to the environment to a level compatible with the feasibility of the off-site emergency plans.

Deriving from the studies made on the basis of the WASH 1400 report, one was led to the definition of three typical source-terms to be used for the assessment of severe accidents. Ultimate procedures were then developed to make the fission product releases compatible with emergency plans.

2.4.1. Reference source terms versus external emergency plan feasibility

In France, the expression "source term" is used in a restrictive sense with regard to radioactive releases. A "source term" is a typical release, characteristic of a reactor type and of an accident class. Possible defense against these accidents is sought for in view of the ultimate protection of the population ; they are therefore essentially a reference for defining emergency procedures on the plant and assessing the validity of emergency plans : "Plan d'Urgence Interne" (internal emergency plan), abbreviated PUI, of the power plant and "Plan Particulier d'Intervention", PPI, (off-site particular emergency plan) beyond the site limits. Thus the notion of source term cannot be associated with a specific accident sequence, but rather represents a class of releases.

As shown in Table I, three source terms are defined in France for PWR severe accidents, and they all assume a complete core melt-down.

In order of decreasing severity, they are :

- S1, which corresponds to a total and very early loss of containment tightness. Research results tend to indicate that early containment failure is unrealistic. If fast interactions between molten fuel and water, as well as local hydrogen detonations or overall explosions are still considered physically realistic, sudden steam explosions, overall hydrogen detonations, direct overheating, theoretically capable of inducing early containment failure, are deemed physically unrealistic for the large "drywell" containments used in France. Despite this conviction, studies on this subject are going on, with a view to achieving a better understanding of the phenomena involved and enhancing even further the "defense-in-depth", by investigating for instance the possibilities of corium cooling by water injection,

- S2, which corresponds to a large and direct release of radioactivity to the atmosphere, beginning one day after the accident onset (for example δ mode in the Rasmussen terminology),

- S3, which corresponds to an indirect release to the atmosphere, starting one day after the accident onset, through leakpaths between the containment and the atmosphere involving a substantial fission product (F.P.) retention ; S3 also incorporates the minor, normal releases of the containment before its impairment.

These source terms derive from U.S. assessments established more than ten years ago (essentially the WASH-1400 report), which were adapted in the late seventies to PWRs built in France.

Feasibility studies on PPI in France were completed in the early eighties ; they resulted in the following conclusion : for French PWR sites, when using classical operational means, it appears feasible to evacuate the population within a radius of about 5 km around the plant, and to confine it within a radius of about 10 km, provided there is at least a 12-hour advance warning before the postulated releases.

This being considered, in compliance with ICRP-40 recommendations, it appears that S3 corresponds to release characteristics that can be correctly accommodated by the current PPIs.

This means that steps had to be taken to mitigate the consequences of still conceivable core-melt sequences that could otherwise result in a S2-type release. This is the purpose of procedures U2, U4 and U5.

2.4.2. U2, U4 and U5 procedures for consequence mitigation

2.4.2.1. U2 procedure

This procedure addresses the search for and repair of abnormal containment tightness defects (β mode).

The U2 procedure must in fact cover a wide range of accident severity because it is obviously desirable to activate it as soon as any threat of significant release of radioactivity inside the containment has been discovered. It defines :

- the condition of containment surveillance (radioactivity at the stack, in the sumps and inside the containment, state of containment isolation systems),

- the action to be taken to mitigate the radioactive releases (for example : isolation of unit, reinjection of liquid waste inside the reactor building).

This having been accepted by the Safety Authority, the U2 procedure is currently operational on all 900 MWe and 1,300 MWe units.

2.4.2.2. U4 procedure (ε mode)

During the studies devoted to the analysis of the consequences of the basemat melt-through by corium, it appeared that, in the 900 and 1,300 MWe standard basemats, direct pathways to the atmosphere of early releases, not filtered by the ground (basemat auscultation, draining systems), existed.

For the N4 standard, these pathways are eliminated at the design stage. For the 1,300 MWe plants, various arrangements have been completed, covered by the general term of U4 procedure, aiming to suppress or to mitigate the presence of these pathways.

2.4.2.3. U5 procedure (δ mode)

The U5 procedure uses a device making it possible to do planned and filtered releases, conceived :

- to lower the internal pressure of the containment to the design value,

- to decrease significantly the release of some radioactive products into the environment,

- to direct the filtered gases towards the stack, where their radioactivity is counted before dispersion into the environment.

- Filtering System description (Fig. 3)

The device includes mainly a tight container, holding a 40 m^2 sand bed, 80 cm deep, isolated by valves, connected upstream to the containment atmosphere by a pre-existing penetration (used to perform the containment tightness tests) and downstream to the stack.

A research and development program, called PITEAS filtration, was performed on the sand filtration in the CADARACHE Nuclear Research Center ; it made it possible to define the system, to check the efficiency of the device and its ability to accomplish its task under conditions representative of accidents liable to occur. An efficiency factor of 10 for aerosols is obtained, so that a S2 release can be reduced to the S3 level.

All 900 MWe and 1,300 MWe plants on the grid are now equipped with this device and the U5 procedure is currently operational.

With the U2, U4 and U5 procedures, source terme S1 can be excluded and source terme S2 can be lowered to the level of S3, against which, technically speaking, the population can be satisfactorily protected.

In the course of ongoing studies, longer terme management of severe accident consequences is being investigated, taking into account contamination of surface, soil and ground water.

3. R. AND D. AIMED AT DEVELOPING ACCIDENT MANAGEMENT PROCEDURES

A joint R and D effort between the utility and CEA/IPSN has been developed in the following main areas.

3.1. Development of the physical code CATHARE

This code, a version of which is operational, permits a realistic description of the accident physics and kinetics. Such knowledge is essential to define the criteria for initiating the actions anticipated in the procedures, particularly in the symptom-oriented procedures currently developed. In this type of approach, the operator actions are indeed defined at each time on the basis of the actual course of events affecting the NSSS, and not on a supposed sequence resulting from an initiator. Therefore, correlations between the measurable physical parameters and the various states of the NSSS have to be established, so as to define criteria for operator action. Besides, based on the CATHARE code models, the SIPA software is developed as a simulator for studies, safety analysis of incidents and emergency drills.

3.2. Construction and operation of the Integral Test Facility BETHSY

The Integral Test Facility BETHSY has been designed for the analysis of PWR accident situations controlled by automatic circuits and/or operator actions.

The main technical objectives are :

- the validation of the physical assumptions made for the definition of operating procedures, whether event-oriented or symptom-oriented,

- the global validation of the CATHARE code.

The various PWR circuits and systems are modeled, which will provide adequate initial conditions and a physical evolution similar to that on the power reactor. Operator actions will be automatically implemented according to the criteria in the procedures, taking into account the time allowed for intervention, which can be adjusted. Such an option eliminates bias that an interface and various operating crews implementing the actions could generate. Besides the above objectives, the BETHSY facility should provide elements for appreciating the post-accident reactor operation.

3.3. Achievement of the PITEAS program

This research program was aimed at examining the feasibility of the filtration device to be used for containment venting under U5 procedure. Laboratory-scale tests were carried out to specify the filtering material under specific conditions representative of the reference accident scenario. These tests were followed by a new series of tests on a loop at a representative scale, to analyze the filter's thermal behavior, particularly its possible clogging by condensates, and to check that the expected efficiency was met for any accident conditions liable to occur. A filtration efficiency of at least a factor 10 can be guaranteed for the aerosols under the above accident conditions. Full scale tests will be completed in 1990 in the FUCHIA facility in order to confirm the results gained from tests on smaller scale test rigs. The first results indicate that the observed filter efficiency is higher than the design value.

3.4. Severe accident management

To ensure the best management of any accident condition, it is essential to carry on with the studies on the phenomena likely to occur in case of severe accident, so as to consider these phenomena and their possible consequences as realistically as possible ; these studies will also contribute to the improvement of the design of future reactors.

Among the major problems deemed to require further investigation in the severe accident area, the following can be highlighted :

- possible coolability of a molten core,
- effect of the raft concrete type on aerosols produced when the concrete is attacked by the molten core,

- remobilization of the aerosols deposited in the event of an explosion, of limited amplitude, in the containment,
- processes likely to entail a loss of containment leaktightness.

This list is not exhaustive. In France, and allowing for the results obtained in other countries, the research program is conducted along three lines :

1) Continuation of a program of analytical experiments and development of a system of codes using the models developed from the results of these experiments and validated by them. This program covers, among others :

   - physical behavior of the degraded core and the cooling conditions remaining in this case which has been investigated by experiments on the PHEBUS test reactor ; these experiments are used to validate the VULCAIN code,
   - behavior of the containment : hydrogen stratification, effects of deflagration or detonation (PLEXUS and JERICHO codes),
   - characteristics of the aerosols produced by means of out-of-pile tests HEVA carried out on irradiated fuel heated at 1800°C in an induction furnace,
   - study of aerosols in the primary system and in the containment, deposition and remobilization of these aerosols : effort has still to be made in the qualification of models (in France : code AEROSOLS B1), especially in damp atmosphere (steam condensation on soluble aerosols, collision of droplets) ; in addition, thorough experiments on the behavior of iodine have shown the importance of the effects of radiolysis on the compound CsI and of the reactions with the paintwork on the walls.

   A realistic system of computer codes has been progressively developed from the beginning of the eighties ; this system, named ESCADRE (Tables 2 and 3) is used for characterizing -quantitatively, qualitatively and for all the accident duration- the fission products possibly released to the environment ; it is also used to define and evaluate the means for severe accident mitigation and management (limitation of core degradation and containment failure). The ESCADRE modules have been qualified -for most of them- against separate effect experiments.

2) Need for Integral Experiments : the method which consists in qualifying the physical models one by one, from the results of analytical experiments, is not sufficient ; it has the disadvantage of introducing hypotheses relating to the additivity of effects, the nature of physicochemical species, etc., and does not prevent essential phenomena from being overlooked. There is therefore a need for integral experiments in order to validate the overall response of a system of codes, as well as the procedures. The aim of these experiments is to approximate reactor-responses in accidental conditions as closely as possible :

- in the field of prevention, H and U1 procedures, in which thermo-hydraulic effects computed by the CATHARE code are predominant, the overall check is to be made in the BETHSY system loop, as previously mentioned,
- in the field of the source term, a modification of the PHEBUS reactor is in progress. The objective is to allow a representation of all the phenomena, from the melting of a fuel assembly up to the release out of a simulated containment, including the transport out of the primary circuit and the stratification, deposition and remobilization effects within the containment, and using actual fission products. It is a very important program requiring the use of pre-irradiated and re-irradiated fuels and for which an international cooperation has been settled (PHEBUS-FP) (Figure 4).

3) Examination of measures to be taken in the event of off-site contamination. After evacuation or sheltering of populations in case of emergency, it would be necessary to control contaminated areas. Medium and long term decisions would have to be made for the recovery of contaminated

soils and return to normal conditions of living. This is the aim of the RESSAC program which is now in progress in order to study the different actions to be taken and estimate their effectiveness.

4. EMERGENCY ORGANIZATION IN FRANCE FOR THE CASE OF A NPP ACCIDENT

Organizations have been set up and are regularly tested to ensure an adequate management of an accident on a NPP for the short-term period (a few days) after the accident onset. These organizations are based upon a clear definition of responsibilities and roles of the utility and government bodies involved. On each side, there is a local organization -the internal emergency plan (PUI, for "Plan d'Urgence Interne") for the utility, the particular (off-site) emergency plan (PPI, for "Plan Particulier d'Intervention") for the government representative at the "département" level- and a national centralized organization.

4.1. The plant internal emergency plan (PUI)

A three-step PUI exists for each NPP site, which is initiated by the head of the plant whenever an accident occurs : level 1 addresses conventional accidents, whereas levels 2 and 3 correspond to events with actual/potential radiological consequences on- and off-site, respectively. These levels correspond to those of the PPI.

The PUI is initiated at the onset of a series of events requiring the application of a procedure on a pre-established list, so as to provide the operating crew with substantial support for longer term actions : this list comprises, among others, the A1, A2, A3, H1, H2, H3, U1 and U2 procedures previously examined.

After initiating the PUI, the usual plant organization is turned into an emergency organization aimed at :

- making the right decisions and implementing rapidly the relevant actions to bring the NSSS back to a safe state, and mitigate the consequences,
- collecting any information contributing to diagnosing the accident and making a prognosis for its evolution, with the support of the utility expert groups at the national level,
- providing information to the administration.

Putting in place the PUI results in the constituting of four emergency management teams (PC for "Poste de Commandement") and one emergency technical team (ELC for "Equipe Locale de Crise") :

- The local emergency management team (PCL for "PC Local") is placed in the plant control room ; it controls the actions of the crew on shift so as to save the NSSS.

- The plant emergency management team (plant PCD for plant "PC Direction"), which can be evacuated to an on-site bunker (BdS for "Bloc de Sécurité"), is the only team in charge of the plant safety and of the staff protection ; in this prospect, it coordinates the actions of the three other emergency management teams on site. The PCD also ensures the official connections with the local government representative, who is regularly informed of :
  . the plant condition and its anticipated evolution,
  . the radioactivity transfers to the environment, if any, and their expected evolution.

  The plant PCD is connected, at the national level, with the utility PCD at the "Service de la Production Thermique", or SPT (division of power production by thermal units), the Safety Authority PCD (Service Central de Sûreté des Installations Nucléaires, or SCSIN) and the appropriate body of the Ministry of Health (Service Central de Protection contre les Rayonnements Ionisants, or SCPRI).

- The emergency management team for logistic matters (PCC for "PC Contrôles") is responsible for gathering and synthesizing all data regarding local weather conditions and radioactivity, and making previsions of the releases ; the PCC can be sheltered in the BdS.

- The on-site emergency technical team (ELC) is a reflexion group of specialized engineers, the role of which is to assess the real-time situation of the NSSS and its probable evolution, so as to provide the plant PCD with technical recommendations for the short/medium terms accident management actions ; it receives the data from the impaired unit, in particular those of the safety panel. The ELC also transmits the necessary plant-related information to the two national level emergency technical teams, one at the utility SPT, the other at the CEA/IPSN, the latter acting as the technical support of SCSIN : the on-line transmission of data from the safety panel and the plant calculator and continuous connections between the three technical assessment teams permit the analyses to be compared and synthesized (Fig. 5).

4.2. The national-level emergency organization

For accidents involving levels 2 or 3 of the PUI, the utility activates a national-level organization at the SPT. The "département"-level government representative (the "Commissaire de la République"), when implementing the PPI, is supported by a national-level organization. This organization includes the SCSIN, the SCPRI, the "Direction de la Protection Civile" (civilian protection branch of the department of the interior) and the CEA/IPSN.

4.2.1. The national-level emergency organization of the utility

This organization comprises an emergency management team and an emergency technical team, both located at the utility headquarters building in Paris.

- The emergency management team (national-level PCD)

This team, which is in permanent communication with the plant PCD, is the interface with the concerned government bodies, in particular the head of SCSIN (Fig. 5).

- The emergency technical team (ENC for "Equipe Nationale de Crise")

Its role is to supplement the information of the above PCD and to give advice and recommendations to it. The ENC is in close contact with the plant ELC which provides information ; it compares its analyses with those of the other emergency technical teams (plant ELC and CEA/IPSN).

The ENC comprises specialized engineers on call, who are expected to arrive at the emergency technical room within an hour. A representative of Framatome also joins the team when the support of the vendor is requested ; his role is to maintain a continuous connection with the Framatome technical support team.

4.2.2. The emergency organization of SCSIN

Three teams are set up in case of emergency :

- The emergency management team (PCD), chaired by the head of SCSIN, is installed in the emergency center of the Ministry of Industry in Paris,

- The emergency technical team is located on CEA/IPSN premises at Fontenay-aux-Roses, near Paris,

- A team is detached locally, partly to the impaired plant, partly to the relevant Prefecture (office of the government representative at the "département"-level).

On the basis of the information gathered on the plant situation and of the analysis elaborated by the CEA/IPSN, the head of SCSIN verifies the adequacy of the actions taken by the utility ; he makes a prognosis regarding the releases of radioactivity and provides assessments of possible radioactive transfers into the environment. Such previsions, as well as those from the utility, should allow the local government representative to take, after the SCPRI advice, the appropriate actions for protecting the public.

Besides, it should be noted that a "guide d'intervention accident grave" ("severe accident intervention guide") has been written by Electricité de France taking into account the present knowledge on the phenomenology of severe accidents.

A significant number of technical exercices involving at least the utility, SCSIN and CEA/IPSN have been carried out up to now : the lessons learned constitute a major contribution to the improvement of the emergency organization.

Once the accident has been settled on the plant, the issue of longer term protection of the public and minimizing the economical impact of the accident is addressed by the off-site post-accidental plan (PPA). This implies an extensive scrutinizing of the areas downwind, possibly contaminated by the releases and the control of food, water and agricultural products. Local and national technical divisions of many government departments are involved, coordinated by a "Préfet" of the area concerned by contamination.

5. WAYS OF IMPROVING THE SAFETY OF FUTURE NUCLEAR POWER PLANTS

a) In the space of 10 years, the notion of "unacceptable consequences" has become more restrictive. By today's standards, it would appear necessary, all probabilities being otherwise equal, to reduce, for future reactors, the quantities of radioactive substances released, in view of the particular sensitiveness of public opinion to this aspect.
On another hand, up to now, surveys on the radiological consequences of accident situations only dealt with dose

equivalents which could be received by the populations concerned. Since the Chernobyl accident, it has become necessary to examine possible consequences regarding contamination of the ground and of human food products. In the European Community countries, reference can obviously be made to the food marketing standards determined after this accident, even if they have no strictly medical relevance. This doubtless constitutes an additional reason for reducing radioactive releases.

b) In the French context and insofar as the overall safety objective remains the same, it appears today to be no strong incentive for a radical design change in French nuclear power plants. On the contrary, it would seem indispensable to derive maximum benefit from operating feedback, which is considered by all the various organizations concerned as an essential means of improving safety. Moreover, the probabilistic studies performed for the 900 MWe and 1,300 MWe plant units, now completed, provide a basis for identification of relative weak points in the French nuclear units, thus indicating topics for deeper discussion in the context of relatively constant design features.
However, the impact of new projects developed in other countries must not be underestimated, taking into account that certain such projects have interesting safety features and also the possible influence of a decision to build such a new type of nuclear power plant in another country. At the very least, the solutions envisaged in these new projects have to be carefully investigated ; achievement of greater simplicity in plant design and greater passivity of systems used to limit the consequences of a primary break are essential topics for prospective discussion relating to the PWR 2000 standardized series, and this opinion is substantiated by the results of the probabilistic studies mentioned above.

c) Rethinking appears necessary for the man-machine interface. Besides installation simplicity and passivity of systems, thought must also be given to the degree of automation of some actions ; it is particularly likely that for some accident sequences, a higher level of automation in French nuclear power plants would be beneficial from a safety point of view.

d) But the essential point is that, from the present time, even in a evolutionary context, and in connection with the previous point a), an in-depth appraisal of confinement appears necessary. It has already been shown how, without altering the basic design of French power plants, additional measures were taken or planned to improve the role of the containment in the event of a beyond-design-basis accident.

Today, the clear boundary line which used to separate design-basis and beyond design-basis conditions, has completely disappeared. Investigation of severe accidents must be integrated in the design itself along with appropriate rules.

In short, confinement must be the best achievable in all conditions, from normal operation up to severe accidents.

This concerns, in the first instance, the containment itself and a comparative analysis of the various types of containment already used in France (particularly, single pre-stressed concrete containment with leaktight liner and double containment with inner enclosure in pre-stressed concrete and outer enclosure in reinforced concrete) ; but the possibility of improvements to the containment should also be investigated at the design stage, for example : preventing basemat melt-through by corium and, assuming that it is still necessary, integration of the filtered containment venting system.

This also concerns the possibility of containment bypass, with current containment design, particularly with regard to problems arising with steam generator tube breaks. Design modification of the steam generator secondary enclosure must, in this connection, be investigated.

APPENDIX 1

Incidents of moderate frequency, the consequences of which must be extremely limited :

- Uncontrolled withdrawal of RCC assembly, with reactor subcritical,
- Uncontrolled withdrawal of RCC assembly, with reactor at power,
- Incorrect position, drop of RCC assembly or group of RCC assemblies,
- Uncontrolled dilution of boric acid,
- Partial loss of primary coolant flow,
- Startup of an inactive loop,
- Total load rejection, turbine trip,
- Loss of normal feedwater,
- Malfunction of normal feedwater,
- Loss of offsite power,
- Excessive load increase,
- Inadvertant opening of a pressurizer valve (momentary depressurization of the primary circuit),
- Inadvertant opening of a secondary valve,
- Inadvertant startup of safety injection or emergency borification.

Very infrequent accidents, the consequences of which must be sufficiently limited :

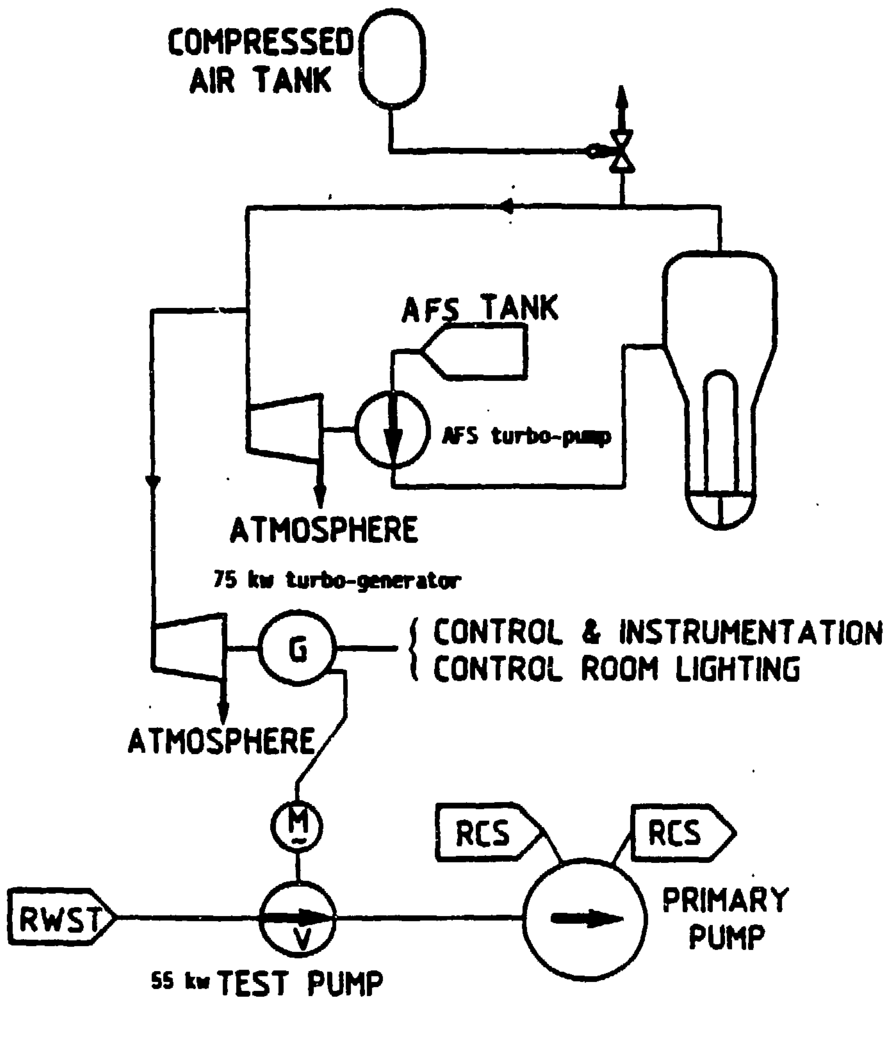
- Loss of primary coolant (small breaks),
- Inadvertant opening of a pressurizer valve (long term depressurization of the primary circuit),
- Small break on secondary piping,
- Total loss of primary coolant,
- Incorrect position of a fuel assembly in the reactor core,
- Withdrawal of an RCC assembly at full power,
- Rupture of chemical and volume control system tank,
- Rupture of gaseous waste treatment system tank.

~~Severe and~~ hypothetical accidents, the consequences of which must remain acceptable :

- Fuel-handling accident,
- Serious rupture of a secondary circuit (water or steam pipe),
- Motor-driven primary pump rotor blocked,
- RCC assembly ejection,
- Plausible loss of coolant accident.
- Double-ended break of a steam generator tube.

FIGURE 1

H3 PROCEDURE



AFS : Auxiliary Feedwater System
RCS : Reactor Coolant System
RWST : Refueling Water Storage Tank

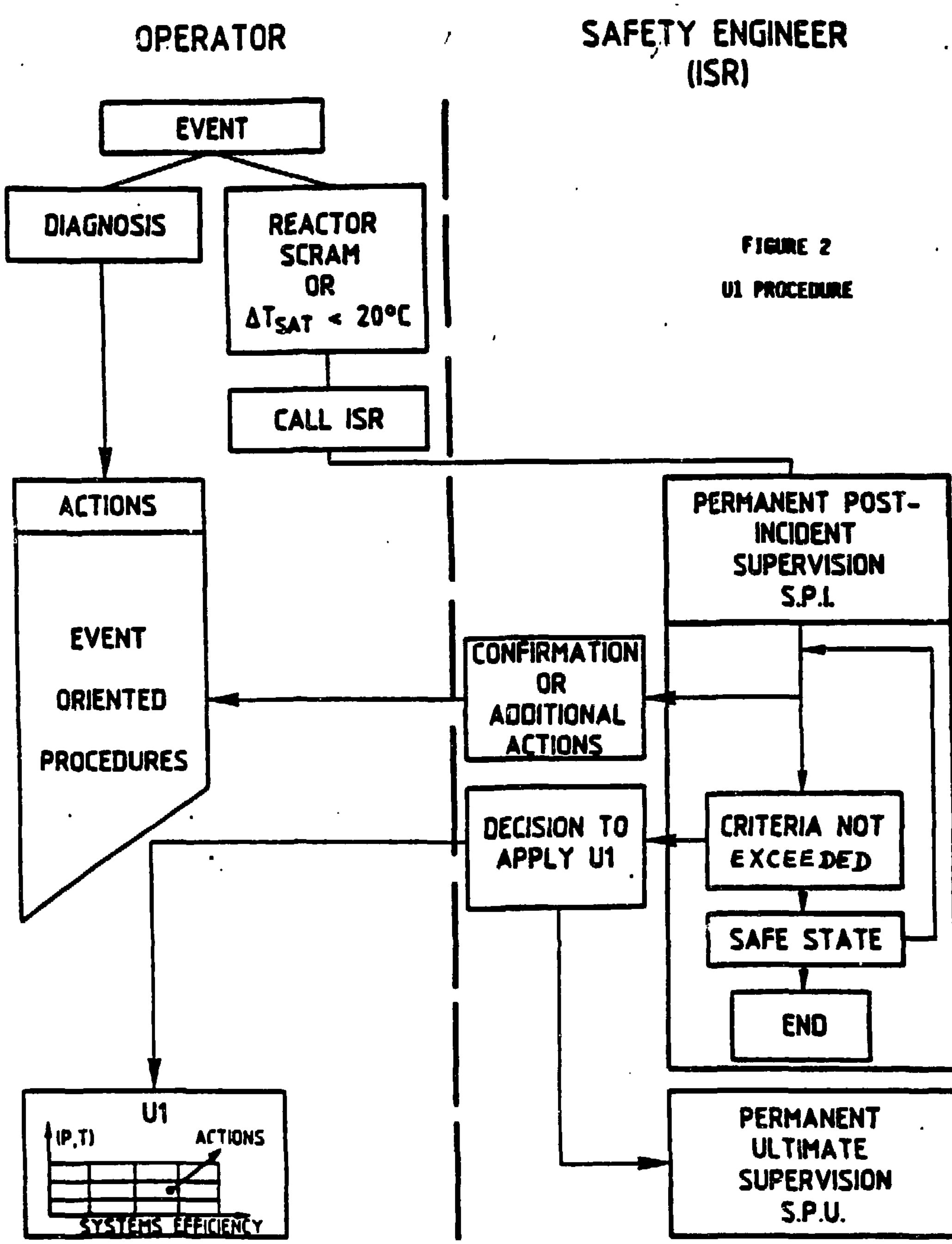


FIGURE 2

U1 PROCEDURE

TABLE 1

CALCULATED SOURCE TERMS INTO THE ENVIRONMENT (INTEGRATED VALUES IN % OF CORE INVENTORY AT REACTOR SCRAM) FOR ALL PWRs AS BUILT IN FRANCE

Source Term	Noble Gases (1) as Xe 133	Iodine (1) as I 131		Cs (1) as Cs 137	Te (1) as Te 132	Sr (1) as Sr 90	Ru (1) as Ru 106	Lanthanum Actinides as Ce 144
		Inorganic	Organic					
S1	80	60	0.7	40	8	5	2	0,3
S2	75	2.7	0.55	5.5	5.5	0.6	0.5	0.08
S3	75	0.30	0.55	0.35	0.35	0.04	0.03	0.005

(1) - For other isotopes of the same chemical category adequate decay half-lives may be taken into account where appropriate.

FIGURE 3

U5 - FILTRE A SABLE

U5 SAND BED FILTER

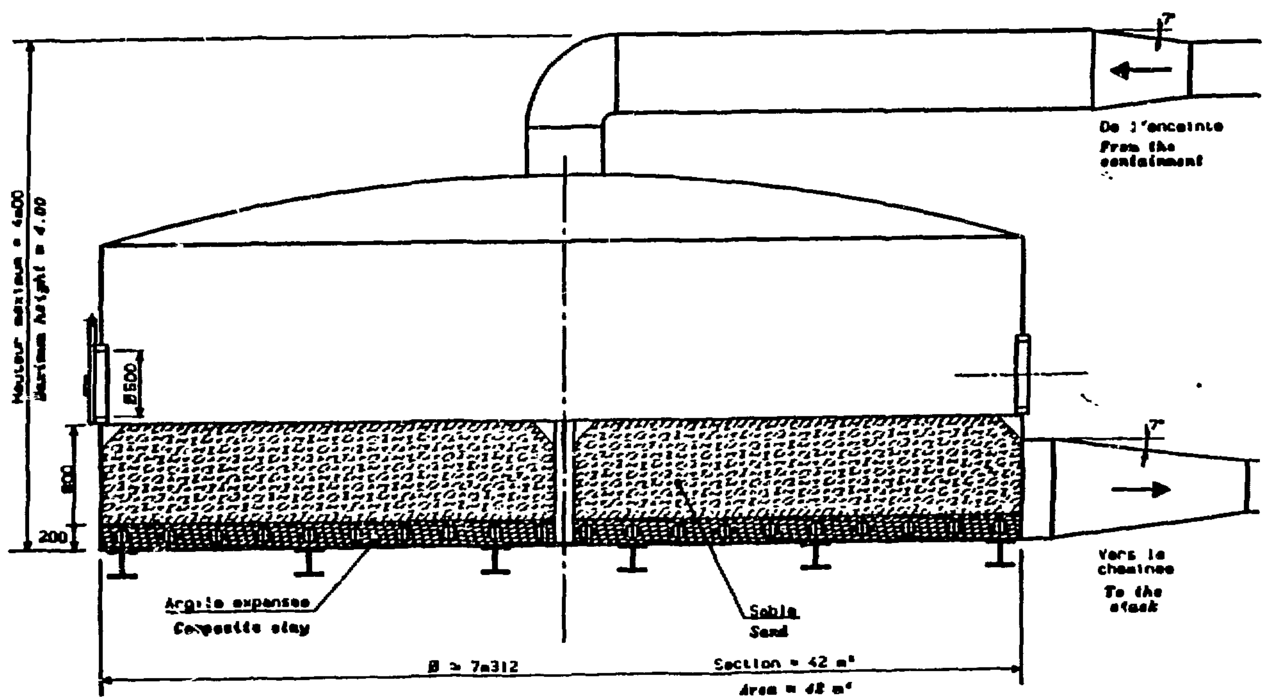


TABLE 2

ESCADRE	06 - 1988

DESCRIPTION OF CODES

THERMALHYDRAULICS

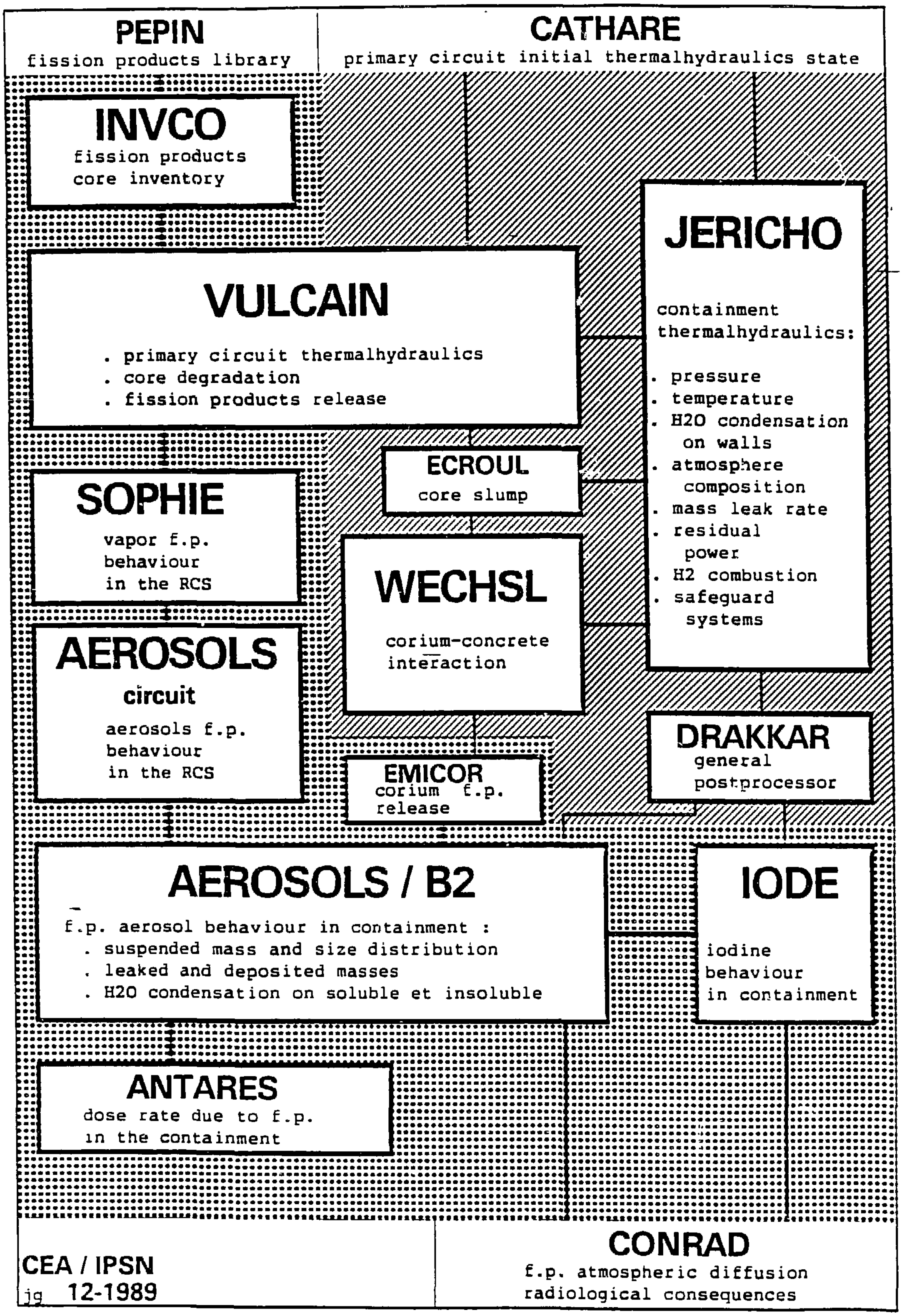
VULCAIN	: RCS THERMALHYDRAULICS BEFORE CORE SLUMP
ECROUL	: CORE SLUMP AND VESSEL FAILURE
WECHSL	: CORIUM-CONCRETE INTERACTION
JERICHO	: CONTAINMENT THERMALHYDRAULICS

F.P. TRANSPORT

INVCO	: F. P. CORE INVENTORY
VULCAIN	: F. P. RELEASE DURING CORE DEGRADATION
EMICOR	: F. P. RELEASE DURING CORIUM-CONCRETE PHASE
SOPHIE	: VAPOR F. P. BEHAVIOUR IN RCS
AEROSOLS/B2	: AEROSOLS BEHAVIOUR IN RCS AND CONTAINMENT
IODE	: IODINE BEHAVIOUR IN CONTAINMENT
ANTARES	: DOSE RATE FROM F. P. IN THE CONTAINMENT

CEA IPSN / DAS

TABLE 3



the ESCADRE system

System of Codes for PWR severe Accident Analysis

PHEBUS PLANT

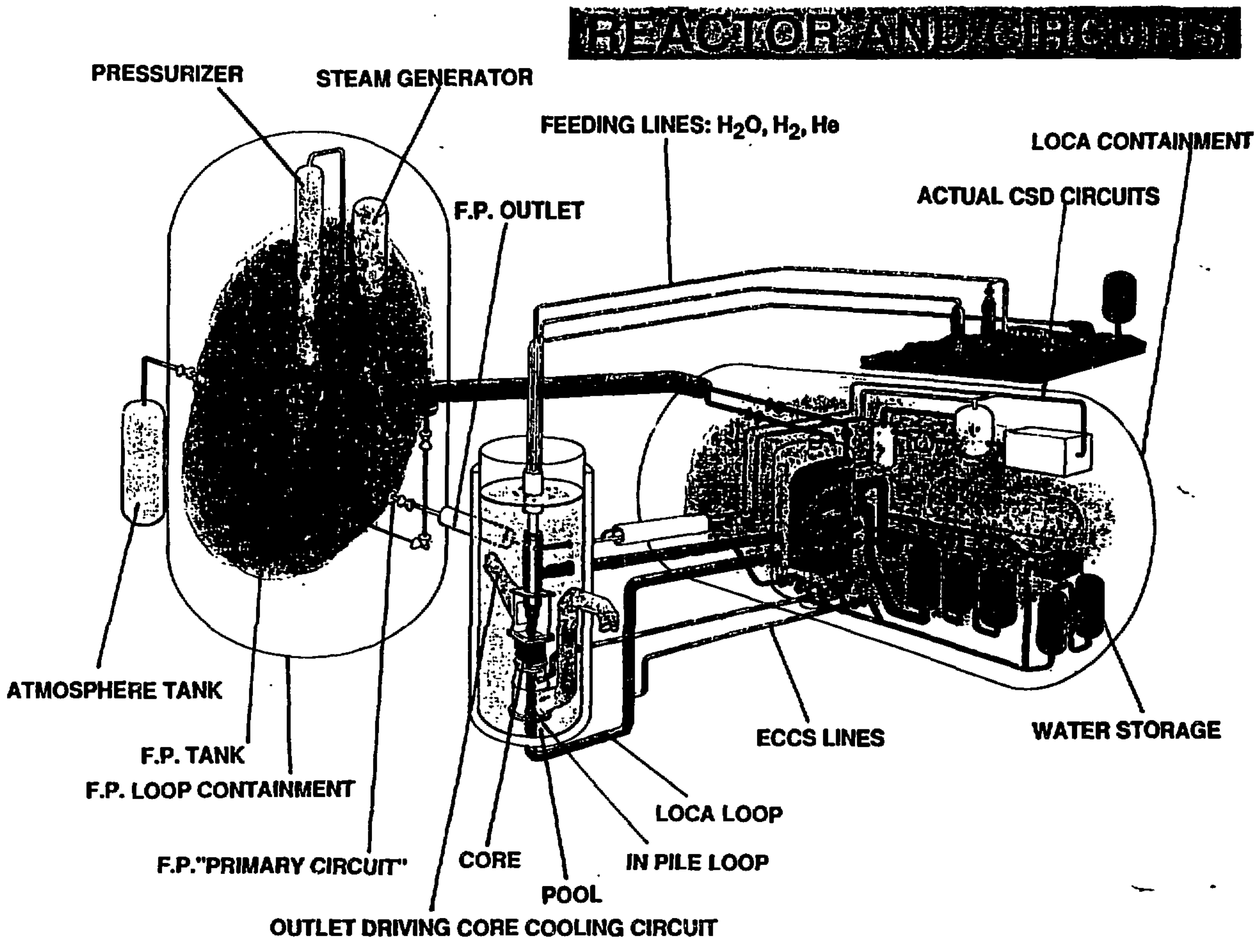
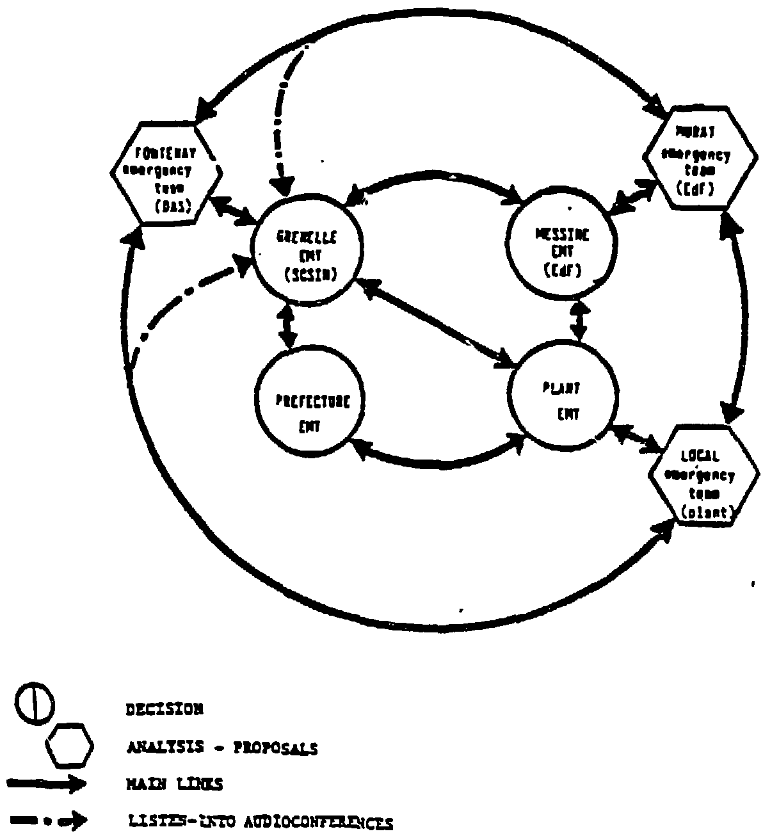


FIGURE 4

FIGURE 5

ORGANIZATION DURING AN ACCIDENT
IN AN ELECTRICITE DE FRANCE REACTOR



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