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FRENCH REGULATORY REQUIREMENTS CONCERNING SEVERE ACCIDENTS IN PWRs AND ASSOCIATED RESEARCH PROGRAMME

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The Safety of a nuclear reactor depends mainly on good design, efficient construction and effective operating rules. However, the possible failure of major safety arrangements designed and implemented respecting design rules does not rigorously have a negligible probability. Accordingly, a safety assessment was carried out on a "beyond of design" basis. in order to identify failure modes and accidents for which one is not fully sure that they are really inconceivable, and probability and consequences of which could justify particular arrangements liable to reduce either of the above terms.

I - APPROACH TO SAFETY DOCTRINE

The appended table is an overall view of the French approach to safety doctrine ; it stresses those aspects relating to severe accidents.

I - Safety objectives as regards populations

The French safety authorities rely on the following safety objective (set in 1977) :

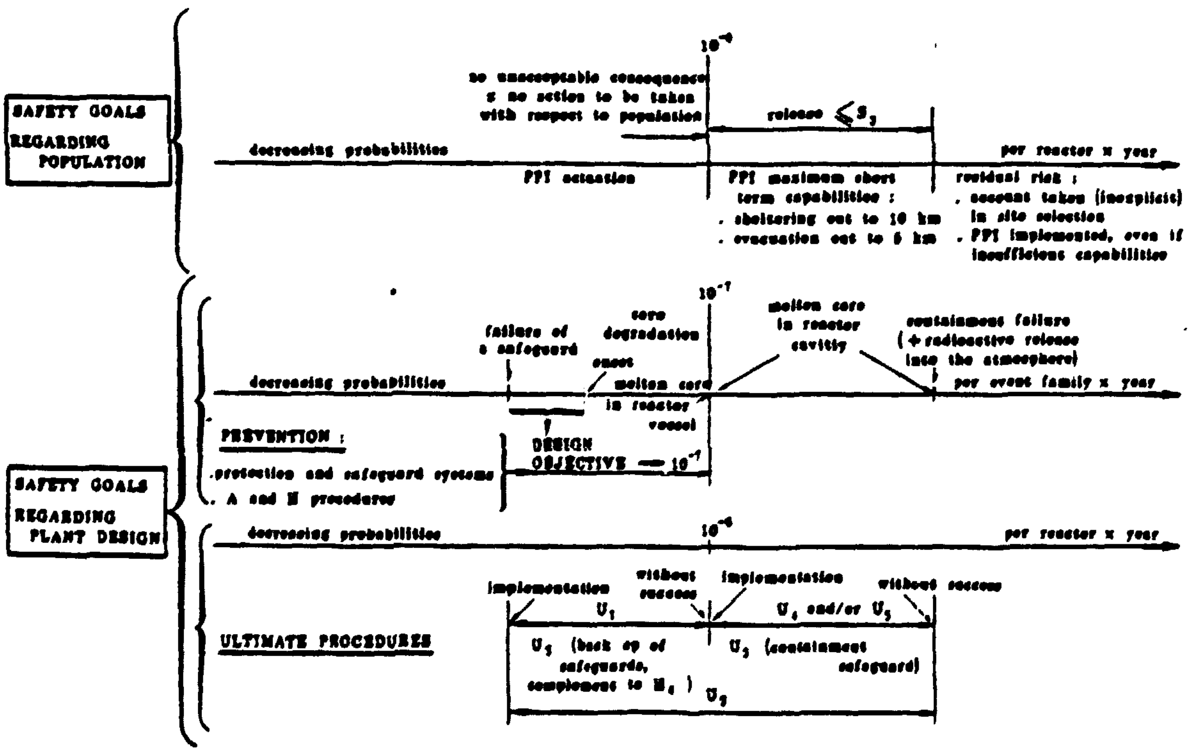
"Generally speaking, the design of the installations of a single unit containing one pressurised water nuclear reactor should be such that the overall probability that the said unit can induce unacceptable consequences will not exceed 10^{-6} per year".

However this probability level must be understood more as an indication of the objective to be sought rather than a strict limit : "the probability figures... have of course to be considered as providing orders of magnitude as concerns the overall safety goal and as concerns each family of events..."

"Unacceptable" implies the obligation of taking steps to provide protection for people outside the site. In the first place this concerns actions intended to safeguard the life and health of people on a short-term basis, as part of a Particular Emergency Plan (called "Plan Particulier d'Intervention" (PPI), in french).

Studies into more severe accidents than those allowed for in the design show that on the basis of realistic assessments, certain categories of accident could still be considered conceivable, although

PRESSURIZED WATER REACTORS - SAFETY GOALS



highly unlikely. Some of these accidents could have an important impact on populations and the environment.

The concept of importance bound with the consequences of such events depends on the ability of setting up the "Particular Emergency Plan". On the basis of the first studies carried out on the consequences of accidents, the characteristics of the PPI have been established as follows, in view of insuring its setting up with a strong presumption of success : people up to 5 km must be able to be evacuated and people up to 10 km must be confined within the first 12-14 hours after the beginning of the accident before any major radioactive releases, without the subsequent obligation, after the actual emission of radioactive releases, of taking other measures immediately (i.e within the first few days). In these conditions, the PPI will be compatible with the magnitude of the radioactive releases in case of accident, if those do not exceed the characteristics of a limit source term, or reference source term, so called S3 later on, for which appear the following constraints : firstly the concept of a delay (12-24 hours) before any important radioactive releases, and secondly the concept of a maximum radioactive release level (no obligation to provide immediate protection for population beyond 10 km).

The studies have shown that the source term S3 was situated at the level of the consequences of severe accident categories, for which the radioactive products were, for a very low part, rejected outside, taking into account the design tightness specifications of the containment, and for the remaining part filtered, except the noble gases almost totally rejected, notablely through the ground in case of passing through the base mat by the melting core. But the studies of other accidents, among the most severe in the category of those judged still conceivable (in the meaning given earlier) have shown the possibility of direct releases, beyond the 12-24 hour period after the beginning of the accident, noticeably higher than those of S3. In the same time, the elaboration of the arrangements necessary to the setting up of the PPI has shown that, even in assuming relatively high thresholds, the necessary interventions could not be correctly led with releases of such magnitude. Consequently further arrangements on the plant have become as to be considered, however at the condition that their importance remains moderate, considering the very low probability of such extreme events.

Subsequently a number of important developments took place as regards the source term. It appears that the source term based on recent studies and assessments regarding radioactive transfers inside the plant should (except in the case of noble gases) be much lower than that produced by the first studies. But at the same time it appears that thresholds should also be fairly low, to allow proper control of the PPI implementation requirements whose objective is clearly to prevent off-site populations from receiving ill-effects ; from this standpoint, the values in the guide prepared under the framework of the European Community and entitled "Health physics criteria for restricting public exposure in the event of accidental release of radioactive substances" (July 1982) is a good reference. Moreover this concept of an intervention threshold should also be combined with other considerations such as :

a) allowance, given that assessments are supposed to be based on realistic assumptions, of possible unforeseen circumstances at the time of the accident affecting the actual conditions under which the radioactive substances are transferred to the environment, depending on weather conditions and ground distribution of contamination ;

b) the disruption caused by long-term contamination (e.g cesiums) on the conditions of living in the neighbourhood of the plant on a very long-term basis.

In the current state of our knowledge, we must consider that the reference source term S3 is simply a guideline to the arrangements to be made both for the plant and for the PPI, whose characteristics should not be fixed too early as long as the results of the research programmes and corresponding studies are not yet in. To avoid difficulties in the compatibility of source term S3 with PPI feasibility, it was decided to make arrangements, of moderate extent, for the plant which would reduce by a factor 10 releases (except noble gases) caused by some of the most severe accidents under consideration ; this gives an order of magnitude which is considered significant to the extent that a gain of factor 10 for release corresponds to a gain of about factor 16 for contaminated areas in the environment, allother things being equal.

This approach does not include other totally different hypothetical accidents, since these cannot be described on the basis of realistic assessments and consequently cannot be considered as realistically possible. These include accidents leading to sudden break of the containment after a violent explosion ; this relates in particular to break modes α and very probably γ (RSS WASH-1400 terminology) after a steam explosion or a hydrogen explosion respectively. Such accidents and others equally imaginable but just as hypothetical cannot however be passed by in an examination of the acceptability of a site or in the creation of emergency plans.

2 - Plant safety objectives

The Safety Authorities also prescribe the following measures : "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...".

a) For design purposes, the design objective logically aims at the large-scale non-degradation of the core.

As regards the safety objective of 10^{-7} per group of events per reactor and per year to avoid unacceptable consequences, there are good reasons for supposing that preservation of containment integrity has a far from negligible probability of success both from the point of view of direct releases to atmosphere and for example from that of contamination of underground water via the subsoil.

However, to allow for the very partial qualification of current PRA mehtods, and also of common modes liable to simultaneously affect core and containment safeguards and even of a concurrent failure independent of containment leaktightness

(3 Mode), we currently consider that the safety objective will be met when the design objective corresponds to large-scale non-degradation of the core whose probability should not exceed 10^{-7} per group of events, per reactor and per year.

To reach this design objective, probabilistic assessments underline the need for measures supplementing the automatic protection and safety systems normally provided ; this is especially true for systems which are in permanent or frequent operation. To remedy this situation :

-either increased redundancy or diversification is sought for the systems : this approach is continuously under analysis ;

-or additional arrangements are made to promote the implementation of special operating procedures, called H, which overcome the difficulty by preventing major core degradation in staying in a safe state during a sufficient period, allowing to increase significantly the chances to recover the faulting function.

In this second approach, the probability of an accident degenerating to the extent of major core degradation is roughly the product of three terms : the probability of the initiating event, the failure probability of the system called on, and the failure probability of procedure H itself.

As for the latter, given firstly the assumption that additional arrangements cannot strictly comply with normal design rules and secondly that human-factor uncertainties should be considered in the development of the procedure, it is considered that at all events this probability cannot be counted at less thant 10^{-2} per group of events, per reactor and per year.

Five H procedures have been defined up to now (corresponding to five categories of events) :

- H_1 : Loss of external heat sink.
- H_2 : Total loss of main and auxiliary feed-water to the steam generator,
- H_3 : Total loss of off site and on site power source.
- H_4 : Actuation of the containment spray function by safety injection system and vice versa during the recirculation phase,
- H_5 : Protection of along-the-river sites from a flood overtoping the thousand year flood.

For current reactors under construction or in operation, these procedures are considered at design boundaries, whereas for future reactors they are part of the design as is shown by the table. Procedures H_1, H_2 and H_5 are already in operation for reactors. Procedure H_3 will begin in a year's time and procedure H_4 a little later for the 1300 MW reactor at PALUEL.

b) "Degradation-melt core" is supposed to be an almost direct result of the "safeguard failure" event, irrespective of the cause(including possible failure of an H procedure); these are the safeguards designed to ensure timely core reflooding (basically safety injection, direct or recirculation) and residual heat removal. The "safeguard failure" event is clearly identified here since it is the cause of the implementation of the so-called "ultimate" operating procedures or U, beginning with those (U_1 and U_3) designed to maintain the core inside the reactor vessel (non-degraded fuel if this is still possible, or degraded fuel if not).

From the moment when the degraded core can no longer remain contained inside the reactor vessel, in case where all previous measures have failed to have the anticipated effect, it is highly probable that the containment function itself will ultimately be highly threatened, especially by possible loss (under highly-degraded containment environmental conditions during the course of the accident) of the safety injection and containment spray safeguards during the recirculation phase. The integrity of the containment could also be compromised by a defective working of containment isolation system, in correlation or not with containment aggressive environmental conditions. Analysis of corresponding scenarios shows that assuming this happens radioactive releases could exceed (by about a factor 10) the reference source term S_3 used to evaluate the possibilities for implementation of the PPI.

The solution to this difficulty was found in the possibility of setting up other ultimate procedures : U_2 procedure, extension of procedure U_3 to containment safeguard, procedures U_4 and U_5. These procedures are supplemented by special arrangements allowing their implementation. It appears that for French-designed reactors these arrangements may be on a small scale.

Current U procedures are as follows :

- U_1 : Use of any available means to prevent core degradation, or, in the event of degradation, to keep the core inside the reactor vessel ;
- U_2 : Procedure to follow in the event of failure to isolate the containment ;
- U_3 : Use of external mobile equipment to remedy any long-term failure of the safety injection system (RIS) and containment spray (EAS) ; other types of safeguard system also envisaged ;
- U_4 : In the case of 1300 MWe reactors under construction (P4 and P'4 series), possibility of flooding the drainage device inside the concrete of the raft under the reactor pit ;
- U_5 : Possibility of controlled and filtered releases by means of a special filter system (release gain by a factor of about 10), useable in medium term (beyond 12-24 hours) for preventing loss of containment integrity after internal overpressure.

For procedures U_1 and U_2, the corresponding operating rules are currently set up for reactors in operation. Procedure U_3 will be set up for the first 1300 MWe reactor (PALUEL) two years from now. The operating rule concerning procedure U_4 will be set up in 1983. Procedure U_5 will be completed over the next year.

II - DESCRIPTION OF ULTIMATE OR "U" PROCEDURES, INVOLVING THE PROCEEDING OF PHYSICAL PHENOMENA INDUCED BY SEVERE ACCIDENTS

The H procedures, although classified at the design boundary, are basically of a preventive nature towards situations which could otherwise become highly degraded. They will not be described here.

However, the U procedures, except for procedures U_2 and U_3 which are also basically preventive, are closely linked to the proceeding of the physical phenomena induced by severe accidents.

1 - Procedure U_1

The operating procedures under accident conditions of the current type, i.e A (with protection and safety systems operating normally) or H (failure of certain systems) are based on a sequential analysis of clearly-identified accident categories and on an initial diagnosis of the cause of the accident (a single cause per category).

However this approach turned out to be insufficient to cope with all possible situations, especially those resulting from multiple failures or combined accidents, with human or technical origins. As a result on the initiative of Electricité de France a new approach was conceived based on the recognition of reactor cooldown thermohydraulic states, including in particular the ultimate procedure U_1, specifically applying to potentially serious situations liable to lead to core degradation or still applicable if the core is already degraded.

The proposed approach includes the following stages :

- the exhaustive identification of all possible reactor cooldown states, their ranges of stability and transitions ;

- the characteristics of states by measurable physical parameters ;

- the identification, for each state, of corrective and/or remedial actions to be carried out by the operator ;

- determination of the corresponding state diagnosis process and corresponding operating rules ;

- identification of physical measurements and of control room data processing required to the diagnosis process.

This method has been proved to be feasible.

Procedure U_1 will be used either because the clearly-identified (even partial) failure of an essential safeguard will lead to the abandonment of a sequential procedure during its course, or because the reactor cooldown state appears to be in a degradation phase giving cause for concern.

It is probable that entry into procedure U_1 will be a difficult choice for the operator. Accordingly EDF has decided to set up a "Permanent Post-Incident Monitoring" process ("Surveillance Permanente après Incident, SPI, in french) for all plant incidents; monitoring will be supervised by the "Safety and Health Physics Engineer" (Ingénieur de Sûreté et de Radioprotection, ISR, in french), whose role is to advise the operator on the procedure to follow as long as controlling events is no problem, but who takes overtotal responsibility for operations when he considers it necessary to enter procedure U_1.

Additional instrumentation is provided to make the procedure fully operational : temperature under the reactor vessel dome, level in the upper space between the core outlet and the reactor vessel discharge nozzles,void fraction in the hot legs. Corresponding technical approaches are currently being studied.

Despite these deficiencies, procedure U_1 is already applicable to all existing reactors.

2 - Procedure U_4 and U_5

Raft attack by the molten core and containment pressurisation due to steam and incondensable gas production are considered inevitable in cases where the containment spray system (EAS) is no longer operable. Furthermore, even if this system is initially available, containment conditions during the recirculation period could not allow a durable operation of the system, which is not qualified for these conditions.

As already indicated, the hypothesis of a violent explosion leading to loss of containment integrity has not been retained, whether it is due to water-molten fuel interaction or to hydrogen. However, it is assumed that severe reactions of the same origin are possible and may be an aggravating factor in the accident.

Procedure U_4

This procedure is only used in 1300 MW reactors. These reactors are provided with a double containment system comprising two walls for the off-ground part and a net of drain pipes in the raft, including the section under the reactor pit. During core melt through the raft, these drains can provide a path for direct radioactive release to the atmosphere at a given time.

Procedure U_4 involves flooding the drains as soon as raft attack has begun, using a special-purpose mobile device supplied from site water reserves.

Although there is much uncertainty about the effectiveness of this method, a large proportion of the iodine and cesium could be trapped by the water.

For future reactors, EDF proposes a new drain system design, so this type of weakness will not exist anymore.

Procedure U_5

As regards accident scenarios involving both core melt through the raft and loss of containment spray, research shows that internal containment pressure will increase until it reaches after a few days a value beyond which containment resistance is no longer guaranteed.

Meanwhile radioactive substances (excepting noble gases) will gradually leave the containment atmosphere by deposition on walls or in the sump water. Knowledge of transfer and distribution of of radioactive substances inside the containment are still highly deficient and target values depend greatly on the conditions in which the accident takes place.

Even if the containment is in principle capable of withstanding a higher pressure than design pressure (cracking as from 7-8 bar for double concrete containments without liners used in 1300 MW reactors, clean break as from 10-12 bar for single containments with liners used in 900 MW reactors), it could be a reason to consider the possibility to proceed to controlled releases outside the containment from, as it is thought now, a pressure of 5 bar, given also the uncertainty existing on the possible leaks through the containment penetrations beyond the design pressure. Since a gain of a factor of about 10 is required for corresponding potential radioactive release -the reasons for which were given above - all reactors are programmed to incorporate a special filter system designed to control and filter releases.

EDF is studying the installation of such a system, for 900 MWe plants, which will be located above the nuclear auxiliary building (shared by both units). It will be manually controlled and the system will be connected to penetrations already used for other needs.

Filter characteristics have not yet been fully determined. It will use sand through which will be processed the air-steam mixture and other non-condensible gases without trapping of the evacuated energy. Mixture flowrate will not be allowed to exceed 3 kg/sec. Current results of studies and tests show that the target factor 10 will be obtained from a system with reasonable size and weight dimensions, compatible with the considered implantation, and even do not exclude the possibility of reaching with this system figures 2 or 3 times higher.

III - R and D PROGRAMMES IN RELATION TO THE VARIOUS STAGES OF SEVERE ACCIDENTS

The strongest motivation in setting up R and D programmes is the acquisition of knowledge and the tools required to optimise control of the events occurring during proceedings of accidents, including the most severe accidents. In this respect it must be stressed that the term "procedure" used above must be understood in its widest sense. It is not a question of simply drawing up a check-list, although this aspect is included, but of setting up an overall operational guide providing wide scope for deliberation about the diagnosis and corrective actions, with a large part of the final studies being devoted to theoretical and practical training of operating personnel and other people affected by an accident.

R and D areas are numerous and there is insufficient space here to mention all of them.

1) At the limit of the area of <u>prevention</u>, problems related to diagnosis and the correct day-to-day monitoring of accidents should be studied and analysed.

a) <u>The human factor</u> deserves a special mention. Its implications are difficult to control ; in-depth knowledge calls for the close observation of actual behaviour. However, the lessons drawn from initial studies have led to improvements in certain arrangements in the following areas : qualification of procedures (drafting methods, operability), work station design (especially control rooms), personnel education and training methods in particular by highly effective means (instructions assisted by computers, simulators under normal and incident conditions).

b) <u>Thermohydraulics in double phase transient</u> occupies an important place. Extensive efforts have been made throughout the world since the beginning of the launch of the PWR plant series, starting with studies into Design Basic Accidents. Today, all conceivable types of accidents have to be considered : depressurisation through breaks of all sizes in the reactor primary circuit, all types of transients potentially leading to insufficient water inventory in the reactor primary circuit, assuming also failure of the engineered safety features.

The French approach to data acquisition and calculation methods in this field has been consistent from the very beginning : the aim has been to have as physical as possible a computer code, qualified on the basis of analytical experiments, then checked by overall experiments. EDF and CEA, with the help of FRAMATOME, are jointly finalising the CATHARE code (with a mixed team of about twenty engineers at CEN/Grenoble). This code is now capable of computing loss of coolant accidents. When the code has the informatic structure making it operational for any user, i.e normally within about a year, it should then join the wide range of second-generation advanced codes existing worldwide.

The three French partners, EDF, FRAMATOME and CEA/IPSN have just jointly opted for the construction at the Grenoble Nuclear Research Centre of a loop system (BETHSY), which will be operational in 1986. This loop will contribute to improvements in the validation of the CATHARE code, but its major aim is to simulate all the categories of phenomena liable to occur during an accident and thus contribute to the qualification of operating procedures under accident conditions, whether these procedures are based on the sequential development of accident or on the recognition of reactor cool-down thermohydraulic states, as it is the case for procedure U_1.

With the same objectives in mind, also note the EDF venture aimed at the construction over the next few years of an accident simulator for the training of plant personnel based on the CATHARE physical moduli and capable of real time representation of the most complex reactor accidents.

2) Development of phenomena, from the beginning of fuel degradation, presents the problem of the <u>limitation of consequences</u>.

a) <u>The behaviour and even the major degradation of fuel rods</u> in the event of an accident is covered by extensive worldwide research programmes. In France the experimental reactor PHEBUS, which is devoted to study these problems, is now operational. As a result initial tests have been performed which simulate both reactor coolant system depressurisation conditions (within about 20 seconds in the case of a large break accident simulation) then refilling and reflooding successively ; the values reached by clad and fuel temperatures reach and even exceed the values used in the criteria ; the claddings broke and significant initial clad oxidation was observed.

When this phase of the programme is completed halfway through 1984, the plant will be adapted for the implementation of a programme involving studies of phases including much greater fuel degradation, which will begin in the autumn of 1985.

b) It is well-known that the assessment of radioactive releases during accidents is based on highly complex phenomena.

The order of priority in which the various types of problems are tackled depends on the type of safety related preoccupations, as explained above.

A programme of studies and tests is under way at CEN-Cadarache and aims at determining technical specifications for the filtration system designed to handle releases as part of procedure U_5. Current results show that a filtration factor of 10 can be obtained through a layer of sand 80 cm thick with a grain size less than 1 mm, which is crossed under steady state conditions by a downflow of a mixture of steam (1/3) and incondensables (2/3) at 135°C, including simulated aerosols of about 1μm (AMMD) at atmospheric pressure and at a rate of about 10cm/sec. which corresponds to a head loss of about 500mm of water. The test programme also includes an analysis of the influence of plugging and of thermal transients.

The solution to be retained by EDF after these results will then be tested in the PITEAS loop, which includes a tank 1 m in diameter for the sand and will be equipped as required for qualification of procedure U_5, also with the aim of acquiring knowledge of filtration system limit conditions as a function of accident parameters.

The PITEAS loop will then be used to analyse aerosol problems. As regards this type of preoccupation in reactor containments, the CEA has developed two types of computer code : one, called JERICHO, for the analysis of thermodynamic phenomena in the containment during the accident, taking into account in particular of the presence of molten corium in the raft causing the formation of hydrogen and other incondensables ; the other, called AEROSOL, for the analysis of aerosol behaviour. The problem now is to be able to qualify both these codes in relation to accident conditions.

We may also add here that a programme of research into radioactive releases outside the fuel (emission rate and physical-chemical nature of aerosols formed) could be begun using the FLASH loop tested in the SILOE pile at CEN-Grenoble.

c) A number of other phenomena are to be considered in the development of accidents : the consitions (including time) under which the reactor vessel is traversed by the highly degraded core, the vitiating effect of small steam explosions outside the vessel and of hydrogen on the succession of phenomena and fuel movements and radioactive substances, the special problem of reflooding during the accident (procedure U_3 "containment safeguard"), the thermodynamics of molten core reactions on the concrete raft etc... There are no theoretical and experimental studies in France concerning these various problems ; the adopted safety approach has made their priority more relative.

d) However special mention must be made of the hydrogen problem.

On the basis of worldwide information on the risks of detonation or local or widespread deflagration inside the containment, it appears that effects are not crucial given the characteristics of the 900 MW and 1300 MW reactor containments built in France. The following observations can be made :

- given containment size, loss of integrity does not reasonably appear possible, either in straight sections or at penetrations ;

- nevertheless, hydrogen combustion could be a vitiating factor during the accident, with a potential risk of disrupting certain procedures, especially if some sensitive items of equipment are affected (instrumentation, controls, isolation systems,...) ;

- as a result :

all equipment required during such accidents must be clearly identified and performance tested under the conditions to which they may be subjected ;

- appropriate instrumentation must be developed and tested with a view to detecting the appearance of hydrogen ;

- techniques must be researched and developed with a view to minimising the consequences of a hydrogen release into the containment.

For these reasons the following research programme was set up :

- EDF and CEA are taking part in the American programme conducted by EPRI, which includes the vast installations located in Nevada ;

- calculations concerning containment mechanical behaviour (straight sections) are being carried out based on detonation or deflagration different hypotheses ;

- penetration behaviour is being analysed, especially on the basis of any effects caused by high temperatures ;

- some R and D work is being carried out on instrumentation capable of detecting the appearance of hydrogen ;

- following on is done on actions conducted elsewhere concerning devices capable of minimising the consequences of a hydrogen release.

IV - GENERAL CONCLUSION

The approach to safety doctrine in France as regards severe accidents affecting light water reactors has now been clearly outlined. Its main aim is to take all possible measures to optimise control of the most severe accidents that could occur.

The special resulting additional plant arrangements are still being defined and some are being set up by EDF on all its reactors. In particular it has turned out to be possible to lay down arrangements on a small scale compatible with the safety of populations and the environment, with allowance being made for the possibilities of safe implementation of the emergency plans.

The R and D programmes have become more detailed according to the priorities resulting from this approach.

We may add that the various tasks called for by all these problems also appear compatible with other concerns, such as the overall assessment of the risk caused by the presence of these reactors, whose expected potential consequence is to put us in the position of analysing the consistency of the safety arrangements both on a design and on an beyond-of-design basis.

Systématique f + e

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