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COMMISSARIAT A L'ENERGIE ATOMIQUE

INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE

DEPARTEMENT D'ANALYSE DE SURETE

RAPPORT DAS N° 541e

CEA-DAS 541

FRENCH PRACTICE FOR ASSESSING THE FISSION PRODUCT RELEASES FROM THE CONTAINMENT DURING A PWR SEVERE ACCIDENT

J. DUCO, J. DUFRESNE, A. L'HOMME *

OCDE - 10ème réunion du GTP4
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1. INTRODUCTION

French safety philosophy as concerns severe PWR accidents has already been outlined by the Director of CEA/IPSN in an article published in "Nuclear Safety" (1).

Therefore the present paper will focus on :

a) the French reference source terms, as used for elaborating ultimate emergency procedures on PWRs and for emergency planning ;

b) the methods currently developed for more realistic assessments of the release of fission products during a severe accident.

2. REFERENCE SOURCE TERMS

In France three conventional reference "source terms" have been defined ; they correspond to radioactive releases out of the plant which are characteristic of a reactor line and of typical accident classes. Possible defense against these accidents is sought 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, (particular emergency plan) beyond the site limits. Thus the French reference source terms cannot be associated with specific accident sequences, but, rather, represent three classes of releases.

The three source terms defined in France for severe PWR accidents are shown in Table 1 ; 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 ; such catastrophic scenarios are difficult to picture physically for the PWRs with a large dry containment, as built in France, and thus are currently considered as not requiring any specific arrangement ;

(1) Nuclear Safety, Vol. 24, n° 5, Sept-Oct. 1983, p. 589-606

- S2, which corresponds to a large and direct release of radioactivity to the atmosphere, one day after the beginning of the accident (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 fact : 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 addition to the necessary compliance with ICRP-40 recommendations on doses to the population, it appears that S3 corresponds to release characteristics that can be correctly accommodated by the current PPIs. This means that provisions had to be made to mitigate the consequences of still plausible core-melt sequences that could otherwise result in a S2-type release. Such considerations led, in particular, to the specification and installation of a venting/filtering system on all French PWRs (Cf. Appendix 1), which can be manually actuated by the plant staff, protected by concrete shieldings, one day or so after the accident onset.

More generally, such a venting/filtering system is expected to provide the operator with extra flexibility for accident management.

In summary, the French approach for PWR accidents involving core melt-down does not consist in establishing new design rules, but rather in reviewing some particular provisions allowing to mitigate the consequences of some class nine accidents.

3. REDUCTION OF THE S2-LEVEL RELEASES (ULTIMATE PROCEDURES)

The studies on conditions corresponding to the intermediate-level source term S2 have led to the definition of three ultimate procedures, the purpose of which is to restrain plausible S2-level releases to the S3 characteristics, as S2 impact cannot be covered by current emergency plans.

These are :

- The U2 procedure, to be applied in the event of a loss of tightness or isolation failure of the containment (β mode, according to WASH-1400 terminology) ;
- The U4 procedure, to avoid a direct release of fission products to the atmosphere in the event of a basemat melt-through (ε mode) ;
- The U5 procedure, making it possible to avoid a late above-ground rupture of the containment (δ mode) by decreasing the internal pressure by means of voluntary controlled venting through an adequate sand-bed filter, thus retaining a large fraction of the F.P. aerosols present in the containment atmosphere. Actuating the U5 procedure would require a top-level decision on the part of the utility ; such a decision could not be made without a thorough assessment of the pros and cons, after consultation of the technical teams of the national-level emergency organization and of the civilian authorities.

These three procedures are currently being analyzed. In this prospect, a R & D support program has been carried out jointly by EdF and CEA to assess the efficiency of the U5 procedure (PITEAS FILTRATION programme). The results of such a program will be made available to the partners of the ACE (Advanced Containment Experiments) consortium.

4. APPROACH FOR A REALISTIC ASSESSMENT OF FISSION PRODUCT RELEASES

In view of the very low probability of severe PWR accidents, the current French approach for assessing their consequences is characterized by a search for realism. This means that, as far as possible, an attempt is made to identify the physical processes involved and to quantify their effects in the least conservative manner possible.

An experimental approach to the problems raised cannot suffice, mainly for the following two reasons :

- full realistic experiments are not feasible,
- the number of accident sequences to be investigated is very large.

Therefore the assessment of radioactive releases resulting from severe accident sequences necessarily involves the use of a system of computer codes for extrapolating the knowledge gathered from analyses and experiments to the reactor case. The procedure followed for the development, qualification and validation of such a system of computer codes comprises, schematically, the following successive steps :

- a preliminary analysis of the problems aiming at :

• identifying the risk-dominant sequences of events,
• drawing up the list of the major physical phenomena that might be involved,
• identifying the variation ranges of the parameters involved in the physical processes (pressure, temperature, concentration, etc...) ;

- the modelling of physical processes in computer codes ;
- separate-effect experiments designed to qualify the physical models one by one, under realistic conditions for the parameters involved ;
- more integral experiments, considering, for example, parts of the leakpath, to test the validity of the hypotheses made in modelling, the additivity of the effects of the individual phenomena considered and the completeness of the models as regards important phenomena, for the most realistic values of the parameters involved (validation) ;
- application of the code system to the reactor case for the expected risk-dominant sequences ; the associated sensitivity studies should highlight the largest impacts on the radioactive releases into the environment of uncertainties in input data or models of the codes. This should make it possible to verify the pertinence of the assumptions made during the preliminary analysis and to focus the research effort aimed at model improvement on the most appropriate issues.

It is clear that such a procedure is essentially iterative and does not, when extrapolated to the reactor case, allow one :

- to acquire an absolute certainty, due to unavoidable shortcomings as regards the realism of the integral experiments (scale problems, incomplete representation of the complexity of the phenomena) ;
- to make an accurate assessment, as some phenomena are so complicated that they have to be schematized in the sense of an overestimation of the releases (e.g. : the behaviour of radioactive products as they move throughout the upper internal structures of the vessel).

In other words, a good "engineer's judgement" still largely contributes to the evaluation of the reactor calculation tools developed.

5. PRESENT SITUATION

By implementing the successive steps of the approach described in § 4, the following summary can be made.

5.1 Computer code system (ESCADRE)

After a preliminary analysis had been made, resulting in the reference source terms described in § 2, a realistic system of computer codes has been progressively developed from the beginning of the eighties : its present flow chart is given in Appendix 2.

Such a system primarily aims at contributing to the achievement of as realistic as reasonably feasible level-2 PSA studies. Besides, some provisions are currently being made to adapt it to the analysis of integral experiments, such as the Phebus P.F. tests, which are presented below.

The development work recently concentrated on the following three modules :

5.1.1 VULCAIN code

At the beginning of 1985, the VULCAIN code superseded in our system the American BOIL code, of which it is a recast and development. It describes the dewatering and heating of the core, the deformation and oxidation of the claddings before relocation ; it assesses the fission product emission rate using, in particular, NUREG 772 correlations. It also provides the thermal-hydraulic conditions in the primary circuit during the core degradation phase.

5.1.2 JERICHO code

The JERICHO code provides for the calculation of the variations of the containment thermodynamic parameters (temperature, pressure, concentration of various gases, steam saturation rate), taking into account the engineered safety features and physical phenomena indicated in Fig. 1. The internal space of the containment is assumed to be formed of a single compartment, comprising two phases, gaseous and liquid, and including various solid structures participating in heat exchanges.

5.1.3 AEROSOLS/B2 code

This code enables the mass and size distributions of suspended material in the reactor coolant system (RCS) or in the containment to be calculated, as well as the masses of the deposited material on some internal structures.

The starting point of the series of AEROSOLS codes was a version of the American HAARM3 code dating from the end of the seventies. A first step consisted in developing a version which provided for the division of the internal space of the containment into several compartments (AEROSOLS/A1). In a second stage (AEROSOLS/A2), the models related to the presence of steam were introduced, and the numerical and data-processing methods were upgraded. In the third version - AEROSOLS/B1 -, the size distribution of suspended material was discretized in the form of linear finite elements, enabling computations to be made on a leakpath consisting of a succession of pipes and tanks. Ultimately, AEROSOLS/B2 added to the former version the capabilities of treating simultaneously several families of aerosols of differing densities, and of taking into account steam condensation on insoluble and soluble aerosols.

The physical phenomena covered at present by the AEROSOLS/B2 code are given in Fig. 2.

A comparison exercice of the ESCADRE and Source Term Code Package (STCP) models has been recently initiated, following the transmission by USNRC to CEA/IPSN of the STCP ; the first step, currently ongoing, involves MARCH 3 and the corresponding ESCADRE codes (VULCAIN, JERICHO and WECHSL). The CEA objective is to arrive at a good understanding of the STCP capabilities, compared to those of ESCADRE, and possibly to introduce into ESCADRE the STCP models which could turn out to be more appropriate than the current ones, if there are any.

5.2 Analytical qualification of physical models

Separate-effect experiments were carried out - some still to be completed - at the Nuclear Research Centers of GRENOBLE and CADARACHE, relating in particular to :

- the release of fission products from the fuel (hot cell HEVA program) ; more specifically, the objectives of the HEVA program are the following :

  • to simulate a defined accident sequence deriving from a PWR severe accident scenario on a section of irradiated rod ;
  • to study, qualitatively and quantitatively, the kinetics of fission product release during the sequence ;
  • to determine the physical and chemical characteristics of the species released : this type of information provides guidance for elaborating the input data of the SOPHIE code, used for assessing vapor transport of fission products in the reactor coolant system (RCS) ;

- the behaviour of iodine in the containment (impact of radiation, interaction with metal or painted surfaces),
- the behaviour of aerosols in a containment with a dry atmosphere (sodium fire experiments ; EMIS program),
- the efficiency of the ultimate procedure U5 (joint CEA-EdF PITEAS filtration programme : sand-bed filtration) ; full-scale demonstration tests of the efficiency of the sand bed filter as installed on French PWR units are scheduled in 1989 at the CADARACHE center (FUCHSIA program).

Such experiments contributed, in particular, to the qualification of the models introduced into :

- the VULCAIN code, as regards the emission rate of fission products from the fuel, versus temperature and time,
- in the AEROSOLS/B2 code, as regards the behavior of dry aerosols in a containment with a relatively quiescent internal atmosphere (weak convection currents).
- the IODE code, as regards the i dine chemical behavior in the containment.

Other separate-effect studies and/or tests have been recently envisaged or initiated :

- preliminary studies on corium/water interactions after vessel melt-through ;
- aerosol deposition and resuspension tests in a pipe (TUBA program) ;
- vapor deposition and revolatilization tests in a pipe (SOPHIE qualification) ;
- study on the containment response to a localized hydrogen detonation ;
- the participation in the ACE program.

As regards core degradation and relocation under severe accident conditions, CEA/IPSN is currently implementing the PHEBUS CSD - combustible sévèrement dégradé (severely degraded fuel) - in-pile program, which consists of seven tests on 21-rod bundles to be carried out before mid-1989. Three tests are devoted to the case where the cladding oxidation phenomenon is dominant over liquefied fuel formation, whereas two other tests will address the opposite situation. Two more tests are anticipated, one at least incorporating control rod material into the test bundle.

The first test, called B9, was effected on December 9, 1986 ; it was a kind of scooping test, but all the objectives of the test were attained (up to 80 % cladding oxidation, cladding temperatures up to 1900°C).

The second test, the C3 test, was carried out on October 30, 1987 ; its goal was to obtain a significant UO_2/Zy interaction by the collapse of the cladding on the pellets in a hydrogen environment.

The third test, called B9R, was completed on April 14, 1988 ; this test was similar to test B9, but the cooling down - a rapid quenching by water - resulted in rod-temperature-drop kinetics of 10 to 15°C/sec.

PHEBUS CSD semi-integral tests are expected to provide adequate data for the validation of the ECROUL module of the ESCADRE system.

5.3 Overall validation of the code system ESCADRE

CEA/IPSN has contributed to the MARVIKEN V programme, designed for studying the behaviour of radioactive products in the primary circuit of a PWR under accident conditions. Our initial interpretations of these experiments revealed a lack of knowledge of the two basic parameters relating to the processes involved in the agglomeration of aerosols : the collision efficiency coefficient, generally called ϵ in computer codes, and the turbulent energy dissipation rate (ϵ_T). Further computations or experiments are being carried out in France (e.g. the TUBA experimental program) better to reassess these parameters and therefore to enable the interpretation work to be developed.

CEA/IPSN has also participated in the American LACE programme, another overall experiment that is now completed, the object of which was to examine the various modes of release of radioactive products to the environment, and the behavior of aerosols in a containment. The PITEAS Aerosols Physics experimental program is expected to provide, by the end of 1988, improved data, particularly on collision efficiency, for a more developed analysis of the LACE results.

Furthermore, it has been deemed necessary to verify that the ESCADRE system, as well as other appropriate code systems, can describe in a satisfactory manner the whole series of events that control - in the reactor case - the progression of radioactive matter along the leakpath from the core to the environment.

This concern resulted, late in 1985, in the proposal of a new integral test program - the in-pile PHEBUS P.F. program - aimed at providing prototypical benchmarks for the integral, ultimate verification of advanced systems of codes for source-term reassessment.

The PHEBUS PF program refers to large-scale, integral experiments in a set of pipes and capacities connected to an in-pile loop in the PHEBUS reactor, intended to afford an adequate representation of the radioactive matter whole story on its leakpath in a variety of PWR accident scenarios.

To be more specific, the technical objectives of the PHEBUS F.F. program are the following :

- to check the adequacy, additivity and completeness of the ESCADRE models, aimed at representing the important physical phenomena liable to occur in the reactor case ;

- to investigate, under PWR accident conditions, the actual outcome of important issues, which depend on too many parameters to be appropriately assessed by means of a theoretical treatment ;

- more generally, to increase the understanding of fission product behavior during typical severe accident scenarios, and hence to provide the CEA staff with an improved capability for guiding the utility efforts at mitigating the releases, should such an accident occur.

Iodine behavior along its leakpath from the fuel rods to the environment is an example of such important, but confusing, issues. This chemical, according to the French S3-level source term, causes the largest contribution to the dose rate from the groundshine during the first weeks following the releases into the environment (Fig. 3). On the other hand, iodine retention in the systems and buildings, and consequently the source term, strongly depends on the chemical forms of iodine on its leakpath, which are related in complex, not fully understood, ways to such various parameters as the local temperature, pressure and steam/hydrogen ratio, the characteristics of the deposition surfaces, the P_H of the sump water, radiation, ... As of now, all the separate-effect or semi-integral experiments on iodine behavior carried out, and even the TMI-2 accident evaluation, resulted in confusing outcomes.

A complementary lesson learned from the past partial experiments on iodine behavior is that this kind of experiment is not adapted for providing conclusive results, due to the overlapping effects of the governing parameters in the reactor case.

Conversely the PHEBUS P.F. program of large-scale, integral tests under PWR prototypical accident boundary conditions, is liable to provide an improved understanding of the behavior of iodine in some specific cases, as well as that of cesium and other less important chemicals from a radiological standpoint, such as tellurium, strontium, ruthenium, zirconium and lanthanum. Besides, it will provide the physical forms of the source-term contributors, which have a clear impact on further atmospheric transfers and ground depositions ; such knowledge is particularly valuable for on- and off-site emergency planning.

Preliminary analyses of the PWR scenarios of interest to be simulated in the PHEBUS PF program resulted in focusing attention on the following three major boundary conditions :

- the mode of fuel degradation (dominant oxidation of the cladding or prevailing liquefied fuel formation) ; this has a clear impact on F.P. and aerosol in-core emissions ;
- the steam flow in the RCS and the volumic content of steam in the containment, which affect soluble aerosol behavior ;
- the time of residence of F.P. and aerosols within the containment, which has a major effect on natural process removal.

The corresponding tentative test matrix is represented in Table 2. Such a grid is preliminary and subject to alterations in function of ongoing feasibility studies at CEA and of specific interests which could be expressed by potential partners.

Nevertheless, no major obstacle has been identified up to now in such feasibility studies.

The first PHEBUS P.F. test is to be implemented at mid-1990.

5.4 Application of the ESCADRE system

The ESCADRE system has been applied to the assessment of radioactive releases resulting from various accident sequences, aiming at comparing the releases computed in this way with the reference source-terms - S1, S2, S3 - derived from WASH 1400. It is evident that such comparisons are difficult to interpret due to the many uncertainties present, and must therefore be considered as only indicative of a tendency.

The provisional conclusions of these calculations on 900 MWe PWRs can be summarized as follows.

If the containment does not loose its specified tightness before seven days, the S3 source-term appears as a significant overestimation of the releases.

To consolidate such a result, the following points call for further analysis :

- a better assessment of the loss of containment tightness is requisite, in relation with the internal load (temperature and pressure increases, increase kinetics) ;
- measures have to be taken to prevent any early direct connection between the internal atmosphere of the containment and the environment in the case where the basemat is perforated by the corium (by plugging the testing systems of basemats) ;
- the impact of the phenomena likely to affect the conservatism of current release calculations : in particular, the possibilities of resuspension of the radioactive products deposited has to be examined further ; adequate data on this issue are expected to be provided by the PHEBUS P.F. experiments ;

If the containment is by-passed, or its tightness is impaired at an early stage, it is necessary :

- to continue the studies for identifying such accidental sequences and assessing their occurrence probability (PSA) ;
- to assess the radioactive product retention capacity in the surrounding buildings ;
- to develop the studies related to the introduction of the ultimate procedures U2 and U5 on the reactors, to assess their efficiency and their general impact on safety ; it is to be noted that the U5 procedure appears justified in all cases, as it prevents the catastrophic failure of the above-ground portion of the containment.

Valuable data on the two latter issues are expected from the PHEBUS P.F. program.

In case of a severe accident on a 1300 MWe PWR, two particularities appear : the containment has a lower ultimate resistance to internal pressure than the 900 MWe PWR containment, and the basemat structure provides possible early leakage routes to the gap between the containment walls. This makes the additional studies identified above and the installation of the ultimate measures U2, U4 and U5 all the more necessary ; such actions are currently going on.

6. CONCLUSION

The design of French PWRs is not affected by current source-term developments, except for the addition of a venting/filtering system and the plugging of some cavities in the basemats. The ultimate emergency procedures proposed by the utility - such as U2, U4 and U5 - are in the process of being examined for approval by the safety authorities before implementation.

The knowledge gained since the proposal of the reference source terms tends to demonstrate the following :

a) provided no early, large containment failure is induced, the accident scenarios considered up to now - when the above ultimate emergency procedures are implemented - result in significantly smaller releases of radioactivity than the S3 level, the relative abundances of the various radioactive species released being unchanged (S3 pattern) ;

b) conservely, the spectrum of plausible accident scenarios to be considered is presumably larger than previously anticipated (e.g. various β-modes of containment failure to be dealt with using the U2 procedure), and the impact of some adverse physical phenomena (e.g. F.P. resuspension in the containment), has to be assessed better.

Studies are going on to arrive at a realistic value of the releases for the β-mode of containment failure and to adjust the U2 procedure if appropriate.

The tool of the above assessments of external consequences is the ESCADRE system of codes, which is progressively upgraded, step by step, according to the results of the on-going research.

The PHEBUS P.F. program is a masterpiece of the ESCADRE qualification effort, but is also liable to help improve similar code packages.

Besides, the PHEBUS P.F. program, due to the integral and prototypical characteristics of its experiments, should provide global demonstrations regarding iodine and cesium behavior along their leakpath to the atmosphere.

CEA

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.

Table 2

TEST MATRIX (provisional, August 1987)

CONTAINMENT FAILURE MODE → / CORE ↓	RUPTURE MODE / PF RESIDENCE TIME: CONTAINMENT BY-PASS / VERY SHORT	β / SHORT	δ, ε / LONG
A B – large break • rapid heating of the fuel		• Pump, SG • Cold leg break	• Pump, SG • Cold leg break
S D – Small break • dominant oxydation	• Primary circuit, SG • Secondary circuit • By-pass of containment	• Break location either upstream or downstream a SG	
TLB transient with total loss of SG feedwater • eutectics		• Pressurizer • Discharge tank of pressurizer	

CEA

Fig. 1

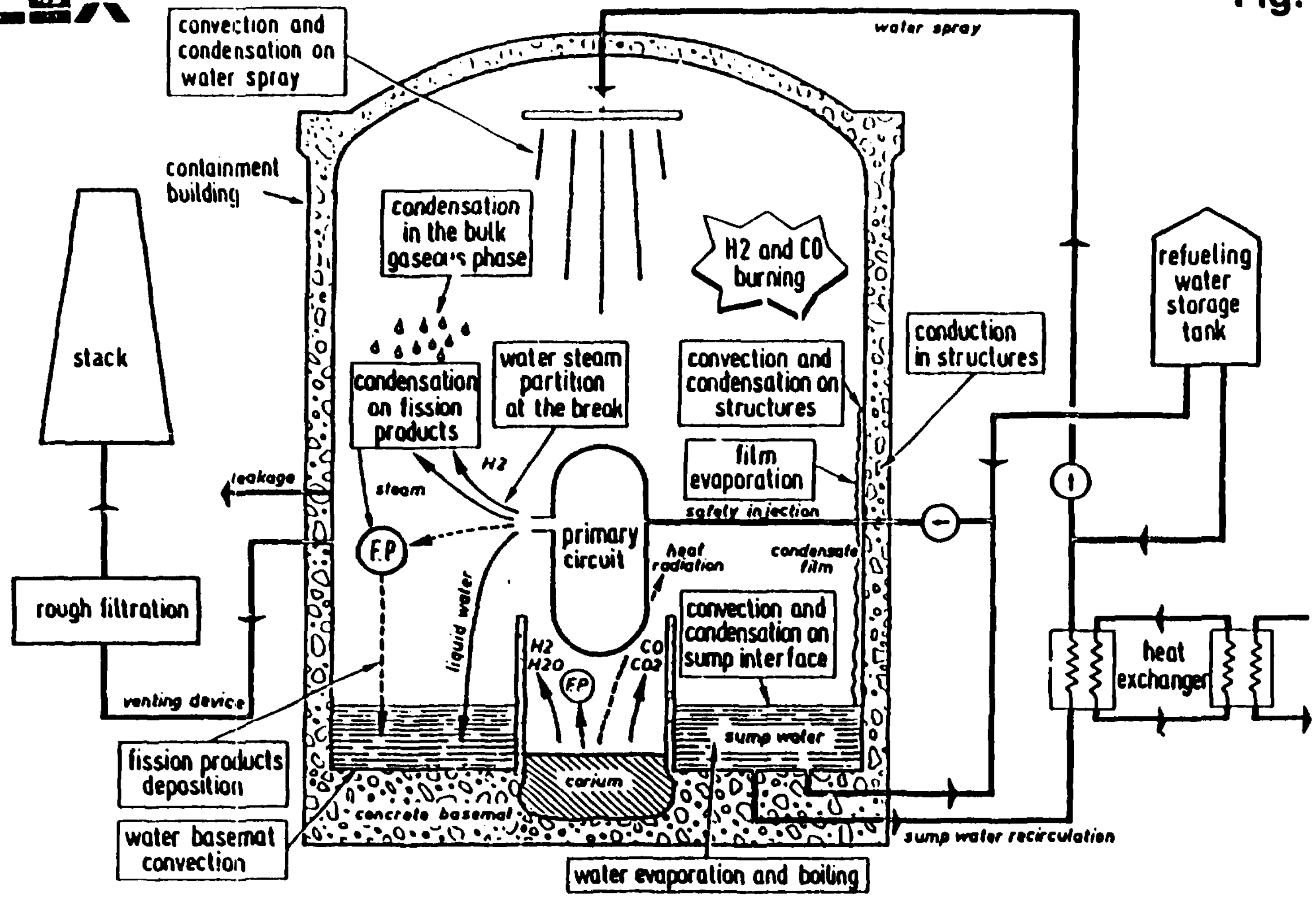
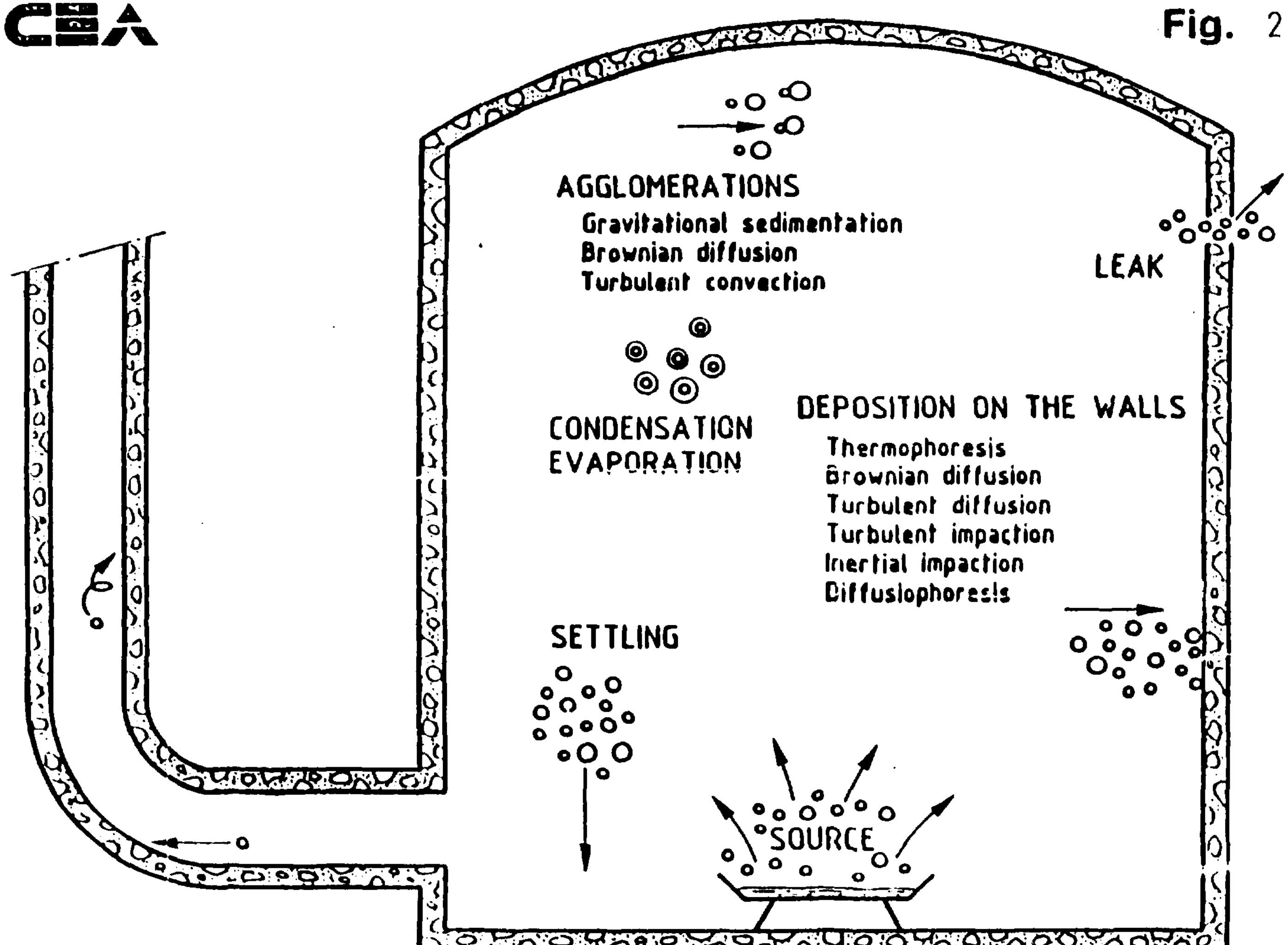


DIAGRAM of SYSTEMS and PHYSICAL PHENOMENA in the JERICHO CODE

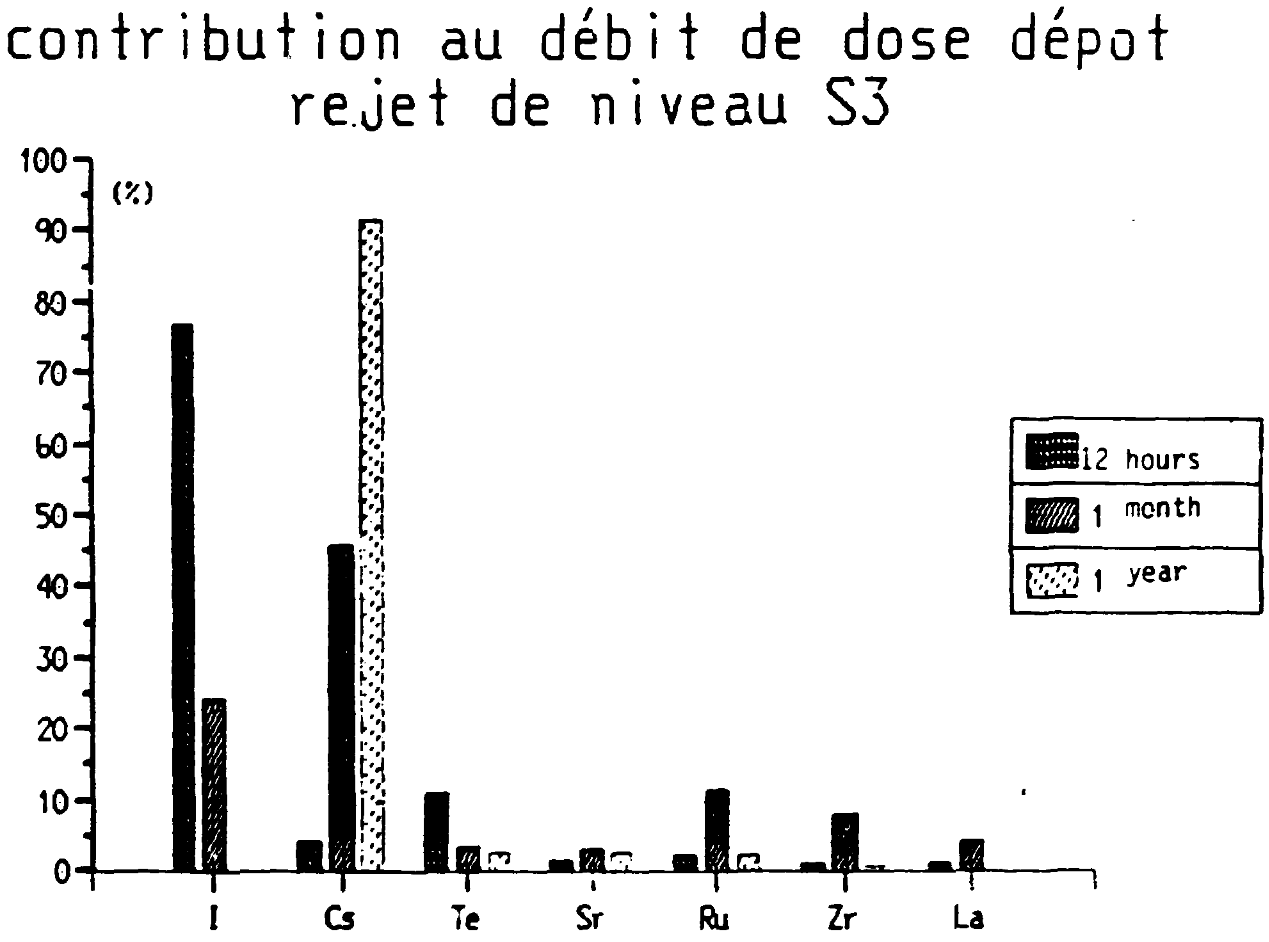
Fig. 2



AEROSOLS CODES

FIGURE 3

Contributors to the dose rate from groundshine after a S3-level release.



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Appendix 1

Venting/Filtering System of French PWR Containments

The venting-filtering system to be installed on all French PWR units has been conceived to mitigate the consequences of severe accidents, by :

- reducing the internal pressure of the containment to the design value,
- decreasing significantly the release of some radioactive products to the environment,
- directing the filtered gases towards the stack, where their radioactivity is counted before dispersion into the environment.

The corresponding ultimate procedure to be applied is termed the U5 procedure.

- Filtering System description (Fig. 1)

The device adopted includes a discharge line, normally isolated from the containment by two valves, a let-down orifice, the filter tank and the exhaust line to the stack, equipped with a gamma-spectrometrer for monitoring radioactivity. A conditioning line, isolated when U5 is implemented, prevents the moistening and possible resulting degradation of the filtering medium. The discharge gas, which has been depressurized to about 1 bar through the let-down orifice and slowed down by baffles along the piping, penetrates into the upper cavity of the filtering tank.

This tank (Fig. 2) is a stainless steel cylinder, 7.3 m in diameter and 3.5 m high. Flow distribution is homogenized at the entry of the sand bed by deflectors and a sifter. The filtering medium is a 80 cm thick layer of sand of specified grade, supported by a 20 cm thick layer of expanded clay, housing a network of stainless steel strainers for collecting filtered gases, the details of which have been patented by the utility. The actual filter efficiency for aerosols is expected always to be higher than the minimum required (a factor ten), even in the early stages of operation, when some vapor condensation will occur due to the initial ambient temperature of the sand.

The characterization of the sand bed to obtain such a level of filtration efficiency derives from the PITEAS filtration R and D support program. The exhaust line is an independent pipe in the stack cavity, dimensioned to provide a gas velocity high enough to sweep away any condensation droplets. The system is not designed to withstand major seismic loading, except the discharge line between the containment and the outlet of the isolation valves ; conversely, it has been checked that the addition of such a system does not alter the seismic response of the buildings and safety-classed systems as built.

- System installation

The U5 filtration system is installed in two different ways, according to the PWR series : one system can be associated to twin 900 MWe units, whereas one system is needed for each of the 1300 MWe units. The filtering tank is located on the roof either of the nuclear auxiliary building (twin 900 MWe units) or of the fuel building (1300 MWe units). At the end of 1987, the following units were equipped : Chinon B1 and B2, Chinon B3 and B4, Paluel 1, Cattenom 1, Cattenom 2, Belleville 1, Nogent 1. All units in operation should be backfitted by mid-1989. The U5 filtration system is included in the N4 standard design (1400 MWe PWRs).

- Operating procedure

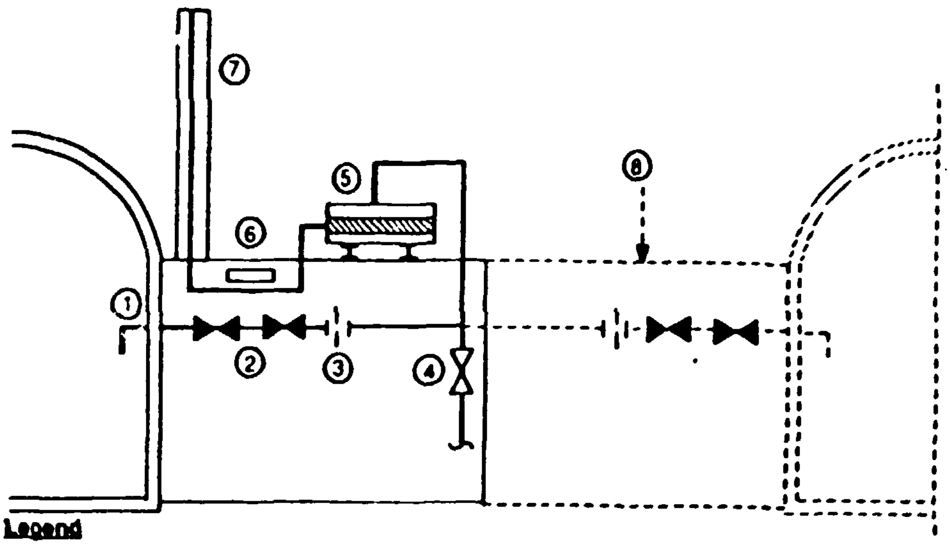
Currently, U5 procedure may be actuated when the containment pressure exceeds the design pressure, which is significantly lower than the containment failure pressure. Such a situation is not expected to occur before one day into the accident, according to our severe accident assessments ; this permits extensive discussions between the plant management, the local and national emergency teams and the civilian authorities, before the decision of implementing U5 is finalized. In this case, the two isolation valves are opened manually by an operator, who is protected by a wall, and the radioactivity on the exhaust line monitored.

The possible extension of the U5 procedure to the prevention and the mitigation of the consequences of other modes of containment failure than the δ mode is currently under investigation. As an example, a competitive leakpath through the sand bed filter could be envisaged in the case of an uncontrolled containment leakage (β - mode of failure).

FIGURE 1

DEPRESSURIZATION - FILTRATION SYSTEM OF THE CONTAINMENT OF FRENCH PWRs

SCHEMATIC DIAGRAM

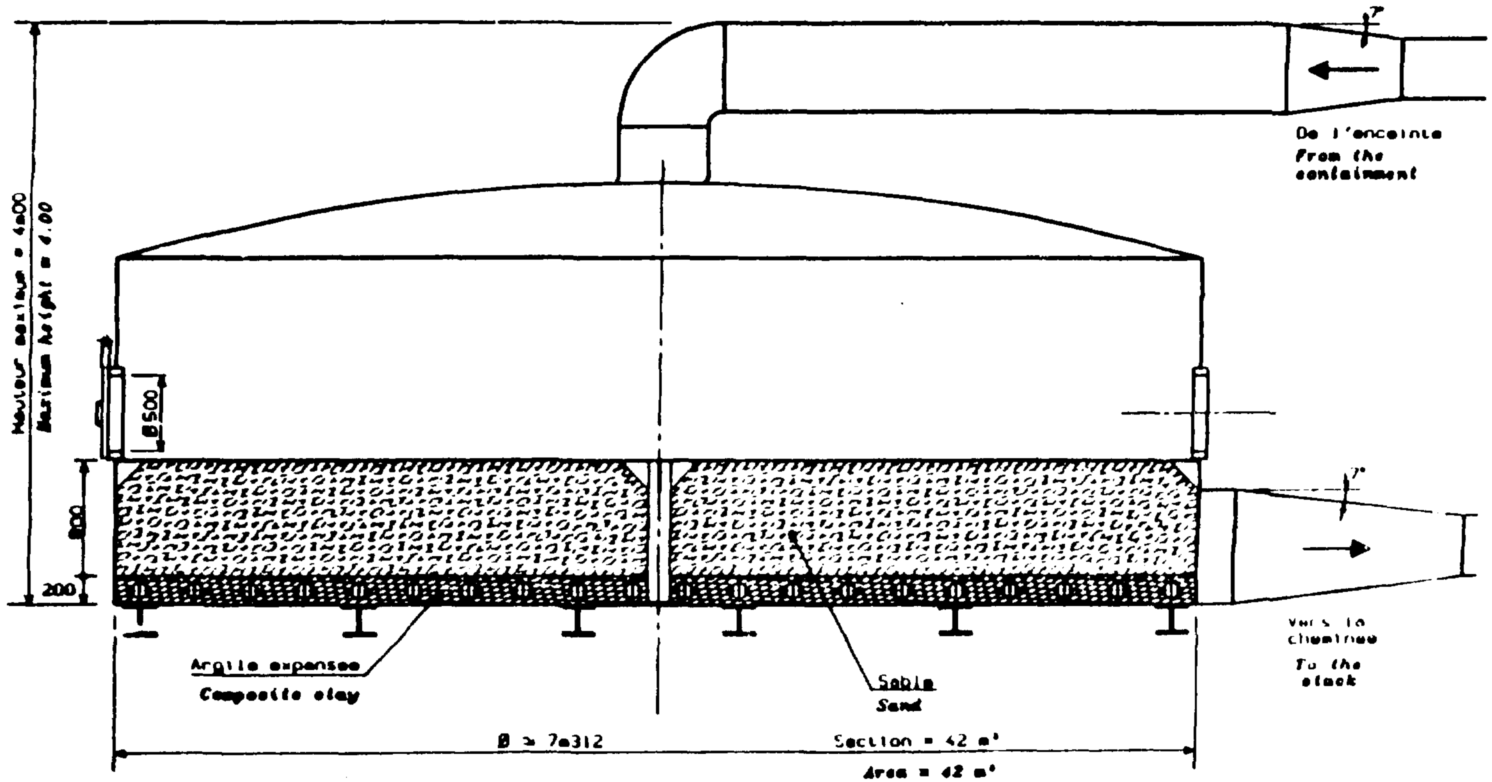


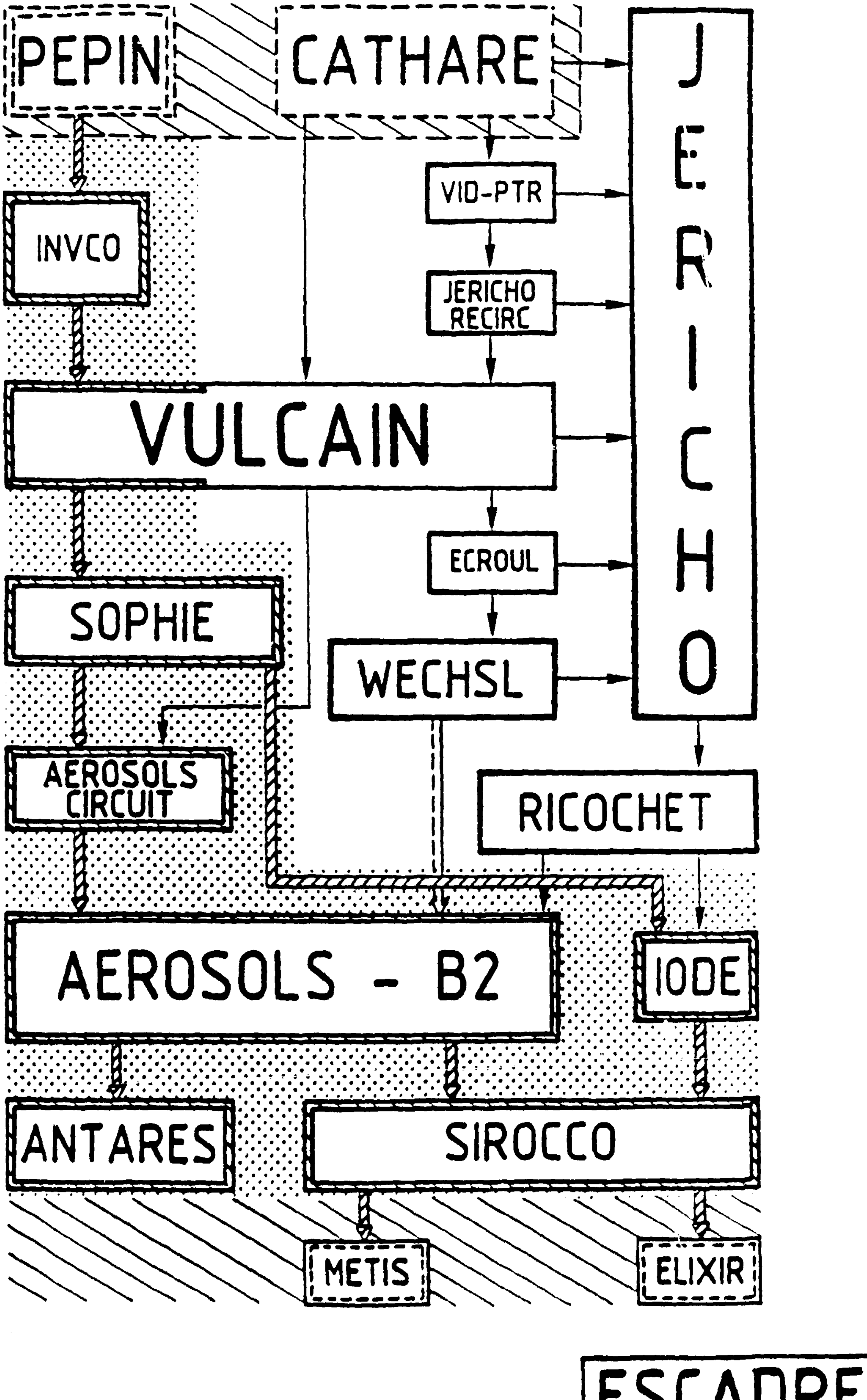
Legend

1. Existing penetration, 300 mm diameter for 1300 MWe plants, 250 mm diameter for 900 MWe plants.
2. Manual valves, operated by reach rods from behind shielding.
3. Pressure letdown orifice.
4. Filtered dry air supply during normal operation.
5. Sand filter.
6. Radiation monitor.
7. Plant stack, with small vent duct inside
8. Arrangement for twin units.

FIGURE 2

U5 - FILTRE A SABLE
U5 SAND BED FILTER




PEPIN
CATHARE
JERICHO
VID-PTR
INVCO
JERICHO RECIRC
VULCAIN
ECROUL
SOPHIE
WECHSL
AEROSOLS CIRCUIT
RICOCHET
AEROSOLS - B2
IODE
ANTARES
SIROCCO
METIS
ELIXIR
ESCADRE

DESCRIPTION OF CODES

THERMALHYDRAULICS (APPENDIX 1)

VID-PTR	:	*RCS THERMALHYDRAULICS BEFORE RECIRCULATION*
JERICHO-RECIRC	:	*RCS THERMALHYDRAULICS DURING RECIRCULATION*
VULCAIN	:	*RCS THERMALHYDRAULICS BEFORE CORE SLUMP*
ECROUL	:	*CORE SLUMP AND VESSEL FAILURE*
WECHSL	:	*CORIUM-CONCRETE INTERACTION*
JERICHO	:	*CONTAINMENT THERMALHYDRAULICS*
RICOCHET	:	*THERMALHYDRAULICS POST-PROCESSOR*

F.P. TRANSPORT (APPENDIX 2)

INVCO	:	*F.P. CORE INVENTORY*
VULCAIN	:	*F.P. RELEASE DURING CORE DEGRADATION*
SOPHIE	:	*VAPOR F.P. BEHAVIOUR IN RCS*
AEROSOLS-CIRC	:	*AEROSOL F.P. BEHAVIOUR IN RCS*
IODE	:	*IODINE BEHAVIOUR IN THE CONTAINMENT*
AEROSOLS/B2	:	*AEROSOL F.P. BEHAVIOUR IN THE CONTAINMENT*
ANTARES	:	*DOSE RATE DUE TO F.P. IN THE CONTAINMENT*
SIROCCO	:	*DOSE RATE DUE TO F.P. IN THE ENVIRONMENT*
METIS	:	*DEPOSITED F.P. HYDROGEOLOGICAL TRANSPORT*
ELIXIR	:	*DEPOSITED F.P. HYDROLOGICAL TRANSPORT*

Novembre 1988

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

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Conseil Général des Mines : M. DE TORQUAT
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Monsieur le Président du G.P.d. : M. GUILLAUMONT
Direction Générale de l'Energie et des Matières Premières : Mlle TISSIER
FRAMATOME : M. le Directeur Général
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TECHNICATOME : M. le Directeur Général
TECHNICATOME : Service Documentation
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M. GOURIEVIDIS (Conseiller Nucléaire près de l'Ambassade de France en RFA)