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ACCIDENT RESPONSE IN FRANCE

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1. INTRODUCTION

French PWR power plant design relies basically on a deterministic approach. In such an approach, the consequences of a limited number of conventional situations are assessed, as well as the relevant frequencies of occurrence, which are classed in frequency categories expressed in terms of orders of magnitude. The conventional situations retained are such that their consequences are larger than those of all other situations of the same frequency category. These situations are called the design basis situations.

A probabilistic approach was introduced in France in the early seventies to define safety provisions against external impacts (aircraft crashes and risks related to an industrial environment). In 1977 an overall safety objective - more a target than a mandatory value - was issued by the safety authority in terms of an upper probability limit for having unacceptable consequences. Following that ruling, the utility was required to prove that the loss of the redundant safety-related systems would meet the safety objective. As it did not, additional measures were taken (the "H" operating procedures) to complement the automatic systems normally provided by the initial design, so as to satisfy the safety objective.

The TMI-2 accident enhanced the interest in confused situations in which possible multiple equipment failure and/or unappropriate previous actions of the operators impede the implementation of any of the existing event-oriented procedures. In such situations, the objective becomes to avoid core-melt by any means available : this is the goal of the U1 symptom-oriented procedure.

Whenever a core-melt occurs, the radioactive releases into the environment must be compatible with the feasibility of the off-site emergency plans ; that means that for some hypothetical, but still conceivable scenarios, provisions have to be made to delay and limit the consequences of the loss of the containment : the U2, U4 and U5 ultimate procedures - the latter providing a venting of the containment through a filtration system - have been elaborated for that purpose.

Emergency management procedures, whether event- or symptom-oriented, need a significant R and D effort in order to be elaborated and checked such an effort resulted in the achievement of the physical code CATHARE and the integral test facility BETHSY, which are both used for adjusting the procedures. The U5 filtration system also derived from the PITEAS filtration research program. In addition, the PHEBUS Severe Fuel Damage program was initiated to provide further data on core degradation events and kinetics ; such knowledge is a prerequisite to establishing a strategy for the optimal use of any means recoved for limiting core degradation. Ultimately, although the probability of a severe accident involving unacceptable consequences is remote, an important research program on the recovery of contaminated soils and water bodies was initiated for providing an adequate support to the optimization of post-accident actions.

For the case of an emergency, a nationwide organization has been set up to provide the plant operator with a redundant technical expertise, to help him save his plant or mitigate the radiological consequences of a core-melt. Besides, such an organization makes prognoses of possible radioactivity transfers to the environment, aimed at supporting the government representative in charge of protecting the public.

2. FRENCH SAFETY RATIONALE

2.1 Design bases of French PWRs for radioactivity retention in plausible situations

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 the 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 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 1300 MWe reactor series (P4 then P'4) and then the 1400 MWe reactor series (N4).

Accident prevention relies essentially on a deterministic approach, the objective of which is to demonstrate that, in the situations considered as 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.
The radioactive substances are confined by means of "barriers" placed between them and the plant staff or the general public. In French-built PWRs, three barriers 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 susbstances can only be released if all three barriers fail.

The "defense-in-depth" concept used to define the situations considered as plausible 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 system ve to be devised 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 unacceptable consequences. In this way, the total failure of the onsite and offsite power supplies can lead to a LOCA (leakage at primary pump seals) which the safeguard systems, having no power, will 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 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, Electricité de France has proposed the following limits, which have been accepted by the Safety Authorities 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 Hypothet. 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 1300 MWe nuclear units.

2.2 Complementary probabilistic approach

2.2.1 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 tonne 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 resulted in deleting some names on the list of candidates for becoming a PWR site.

As early as 1977, an examination of the general technical options for the 1300 MWe PWR series resulted, on the recommendation of the Standing Group ("Groupe Permanent") in charge of the nuclear reactors, in the setting forth of 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 had been met ; nevertheless, this objective was considered as 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 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 made 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 as a "cut-off" value, above which design provisions for the occasion 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 groups 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 as realistic values 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.

2.2.2 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 for several plant design improvements as regards 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 operating conditions. Upon the proposal of the Standing Group in charge of nuclear reactors, the 1977 letter of the Ministry of Industry to Electricité de France requested thorough examination of the probabilities and the consequences of :

a. anticipated transients without scram
b. total loss of the ultimate heat sink
c. 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 is required. Such studies have shown the importance 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.

3. REACTOR OPERATION AND SEVERE ACCIDENT MANAGEMENT

The first domain of application of operating procedures on French PWRs under incident/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 addressing 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.

3.1 The "I" and "A" operating procedures for conventional incident/accident events

The "I" and "A" operating procedures are essentially related to design basis events ; such accidental events 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/safeguard systems.

The rules for treating these events, 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.

3.2 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 and safety-related systems was higher than the safety goal, 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 taking into account all normal operating conditions. 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 "H" is for "hors dimensionnement", that is "beyond design" : actually, the designation "at the limit of the design" would be more appropriate.

3.2.1 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 induces a small break and leads 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 ($T \lesssim 180°C$, primary pressure $\lesssim 45$ 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 generator, 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 of 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. Residual 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 1300 MW plant sites.

3.2.2 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.

This event leads to the opening of the pressurizer relief valves. 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 depressurisation, 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 1300 MWe sites.

3.2.3 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 lead to the damage of the primary pump seals due to the loss of cooling and water injection on these seals. This situation results in a small break and leads to core melt due to the unavailability of the safety injection system.

The major objective of the 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 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 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 1300 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	$1.1.10^{-5}$	$1.3.10^{-7}$
1300 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 1300 MWe plants and have been decided upon for the 900 MWe plants, for which their complete implementation is expected shortly.

3.2.4 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 assured 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 studies showed it was possible, in case of total loss of the containment spray system (CSS) pump, by using connexion sleeves between the low pressure injection system (LPSI) and the CSS, to use LPSI pumps to assure the functions of the two systems and vice versa.

For the 1300 MWe plant, 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 plant, the decision to undertake this modification has been made and it will be completed in about one year.

For the N4 plant, the safety authorities asked EdF to demonstrate that the probability of core melt in case of a LOCA followed by a loss of the safety injection system or containment spray system is coherent with the safety goal ($\leq 10^{-7}$/reactor/year).

3.2.5 Protection of the sites along rivers against floods exceeding the millennial flood

This procedure allows for a flood 15 % higher than the millennial flood. Such an event will result in the loss of external power sources and of the heat sink during for 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.

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

The objective of the measures described in chapter 3.2 is to attempt to fulfill the overall safety goal in the particular case of the loss of redundant systems. However, all the design measures taken at the conception level may be inadequate due either to multiple equipment failures or to the operator's inappropriate previous actions.

In order to attempt to stop the development of potentially serious situations which could lead to core degradation, 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. Implementation of such an approach, which necessitates a water level measurement in the vessel, already installed on all 1300 MWe plants, is foreseen in 1989 at the start-up of the Penly plant and will be achieved progressively on all other 1300 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 1300 MWe plants where the U1 procedure is used by the safety engineer in all incidental and 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.

This procedure will be on line very soon ; computer aids are developped and will be integrated into the safety panel display system.

3.4 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 developped to make the fission product releases compatible with emergency plans.

3.4.1 Reference source terms vs 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 line 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 1, there are three source terms 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 ; such catastrophic scenarios are difficult to picture physically and thus are currently considered as a part of the residual risk, that is, not requiring, a priori, any specific arrangement ;

- S2, which corresponds to a large and direct release of radioactivity to the atmosphere one day after the beginning of the accident (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 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 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.

3.4.2 U2, U4 and U5 procedures for consequence mitigation

3.4.2.1 U2 procedure

This procedure addresses the search for and processing 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 an unit, reinjection of liquid waste inside the reactor building).

This having been accepted by the Safety Authority, the operating rules are now being written for each PWR standard.

3.4.2.2 U4 procedure (ϵ mode)

During the studies devoted to the analysis of the consequences of the basemat melt-through by the corium, it appeared that, in the 900 and 1300 MWe standard basemats, direct pathways to the atmosphere of early releases, not filtered by the ground (basemat auscultation, draining systems), were found.

For the N4 standard, these pathways were eliminated at the design stage. For the 900 ad 1300 MWe files of reactors, various arrangements are under study, covered by the general term of U4 procedure, aiming to suppress or to mitigate the presence of these pathways.

3.4.2.3 U5 procedure (δ mode)

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

- to reduce the internal pressure of the containment to the design value,
- to decrease significantly the release of some radioactive products to 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 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-spectrometer 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. 4) 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, which will be described shortly in § 4.3. 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 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 of 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).

4. R AND D AIMED AT DEVELOPPING ACCIDENT MANAGEMENT PROCEDURES

4.1 Development of the 1-D, 2-Fluid CATHARE best-estimate code (CEA-EDF program, joined by FRAMATOME

The first version of the code has been operational since October 1985, following an extensive, in-depth validation based on a large set of separate-effect experiments. Such a code permits a realistic description of the accident physics and kinetics. This knowledge is essential to defining the criteria for initiating the actions prescribed in the procedures, particularly in the symptom-oriented procedures currently developed. In this type of approach the operator actions are indeed defined at each point in time on the basis of the actual course of events affecting the NSSS, rather than 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. In addition, based on the CATHARE code models, the SIPA software is developed as a simulator for studies.

4.2 - Construction and operation of the Integral Test Facility BETHSY (Figure 5)

The Integral Test Facility BETHSY has been designed for the analysis of PWR accident situations controlled by automatic circuits and/or operator actions. It is a joint research action of CEA, the utility EdF and the vendor FRAMATOME. The main technical objectives are :

- the validation of the physical assumptions made for the definition of operating procedures, whether event-orientend or symptom-oriented ;
- the global validation of the CATHARE code.

The reference reactor for BETHSY is a three-loop, 2775 MWth FRAMATOME PWR with 17 x 17 fuel rod assemblies. The scaling factor of the facility is one in height and 1/100 in volume and power ; 428 full-length, electrically heated fuel rods provide a 3 MW core power, with a cosine axial shape, which represents 10 % of the nominal power. The various PWR circuits and systems are modeled, which will provide proper 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 available for intervention, which can be adjusted. Such an option eliminates bias that an interface and various operating crews implementing the actions could generate. Beside the above objectives, the BETHSY facility should provide elements for assessing the post-accident reactor operation.

The general test matrix can be divided into four main categories :

- Characterization tests of the facility. The purpose of these tests is to permit an accurate quantification of some of the characteristics of the facility (pressure drop along the different parts of the circuits, heat losses, behaviour of certain components). They are essential for a good preparation and a reliable interpretation of the tests and must be performed at the beginning of the test matrix.

- Tests under successive steady state conditions : "physical states" in the BETHSY facility. Steady state tests are performed with different values assigned to certain parameters (primary coolant mass inventory, secondary coolant mass inventory, pressure, core power, number of primary pumps in service, amount of non-condensable gases). By means of these tests it is possible to characterize different physical states, to check their stability and the possibilities of transitions between them. In principle, all these physical states are encountered during accident sequences and the steady state nature of these tests will allow a good physical analysis.

- Transient tests : accident sequences. A number of typical accident sequences have been defined and will be performed on BETHSY with their associated Emergency Operating Procedures (EOP), deduced from reactor procedures after computer code transposition. This will make it possible to confirm the relevance of the EOPs.

- Counterpart tests with nuclear plants and with other integral test facilities. Some comparative tests with reactor transients are planned. Moreover, among transient tests it is planned, in the scope of existing or projected international agreements, that some comparative tests will be performed with other integral test facilities presently in operation. These tests must be as comparable as possible to reveal only geometry and scaling effects, in order to check the ability of the computer codes to take them into account and to extrapolate to full reactor scale.

Steady state and transient tests have been classified according to general physical criteria ; the whole test matrix is the following :

- Characterization tests of the facility
- Counterpart tests with reactor transient
- Primary circuit in single phase flow
- Natural convection in two phase flow
- Secondary side heat transfer degradation
- Energy removal by primary circuit
- Presence of non-condensable gases
- Two-phase flow forced convection
- Combination of previous situations, with multiple failures.

Up to now the following tests have been completed :

- three characterization tests of the facility :
  . measurement of heat losses (test 1.2.a) ;
  . adjustment of electrical heat tracing for different states (test 1.2.b) ;
  . characterization of the mixer of the downcomer (test 1.3.c) ;
- two counterpart tests with reactor transients (piston effect in the pressurizer - test 2.4 - and spray efficiency in the pressurizer - test 2.5) in September 1987 ;
- one test with the primary circuit in single phase flow (natural convection - test 3.1) in November 1987 ;
- two tests on natural convection in two-phase flow (non-compensated 2" break - test 4.2.a carried out in December 1987 and in January 1988).

The next test is scheduled for April 1988 and will be devoted to the study of various modes of natural convection (test 4.1.a).

4.3 Characterization of the U5 filtration system

The PITEAS filtration program is a joint CEA/IPSN - EDF R and D effort, aimed at examining the feasibility of and ultimately characterizing the filtration device to be used for containment venting under U5 procedure.

The characteristics of the aerosol-laden fluid to be filtered, after it has been depressurized to about 1.1 bar abs., are given in Table 2. Such characteristics derive from scenario calculations made at the end of the seventies and have been used for the design of the filter, along with the double requirement of a filtration efficiency of ten at least for aerosols, and a pressure drop limited to 0.1 bar through the filter.

A first series of laboratory tests was carried out at the Cadarache nuclear research center, using the facility described in Fig. 6, in which air was substituted for the non-condensables in the reactor case. These tests, limited to one-hour duration, were conducted under steady-state conditions (pre-heated filter) for the bulk of them, some being achieved under transient thermal conditions, with the filter at ambient temperature at the test onset, so as to confirm the results obtained at equilibrium. The variations of the measured filtration coefficient, as a function of the aerosol size, the sand particle size, and the gas velocity through the filter are depicted in Fig. 7 ; no filter clogging nor noticeable aerosol reentrainment was observed.

A second series of tests was achieved on a pilot loop (Fig. 8) at the Cadarache nuclear research center, to confirm the laboratory-scale results and to further investigate the transient thermal behavior and retention efficiency of the filter when actuated at ambient temperature : some drop in efficiency was observed until the sand dried out, but in all cases the minimum requirement of a factor of ten efficiency for aerosols was satisfied.

Ultimately, full-scale demonstration test are scheduled in 1989. The facility used (Fig. 9) is aimed at reproducing all the story of the discharged fluid from the containment to the stack ; some molecular iodine is to be added at the filter inlet to help in scrutinizing the relevant filter efficiency under realistic transient conditions.

4.4. Severe core degradation phenomenology and mitigation

An EdF-CEA/IPSN joint brainstorming group has been constituted to examine the impact on the course of events of using still available or recovered means during a core-melt sequence : such an analysis is a prerequisite to establishing a strategy for the optimal use of these means. Data on core degradation events are provided by the PHEBUS Severe Fuel Damage Program and by the TMI-2 Accident Evaluation Program.

4.4.1. Objectives of the PHEBUS Severe Fuel Damage program (CEA/IPSN program, joined by EDF)

The PHEBUS Severe Fuel Damage (SFD ; CSD is the French acronym) program aims at providing :

- a better understanding of the physical phenomena governing the degradation of uncovering fuel assemblies in a PWR core and of the impact of possible corrective actions, for typical accident situations ;
- a set of adequate experimental data for the validation of models and codes, such as the ICARE code, devoted to PWR core severe accident assessment.

In this perspective, and due to the limited dimensions of the test train (a 21-rod bundle, 80 cm in height), the purpose of each PHEBUS SFD experiment is to investigate the actual behavior of a specific part of a PWR core ("space window") during a time lapse ("time window") beginning when the fuel temperature exceeds 1000°C ; the corresponding upper temperature limitation, beyond which cooling is initiated, depends upon the test ; stage 3 tests are stopped before the zircaloy melting point, whereas stage 4 trials go beyond, possibly resulting in some fuel relocation. Various cooling kinetics will be tested to assess their impact on embrittled, but still standing fuel rods.

Four typical PWR accident sequences are currently considered :

- a large break LOCA with safety injection failure (Scenario A) ;
- a small break LOCA with safety injection failure (Scenario B) ;
- a prolonged loss of all water sources to the steam generators (Scenario C)
- a prolonged core uncovery a few days after reactor shut-down (Scenario D).

Besides, nine positions have been identified in the core as candidates for a PHEBUS test (Figure 10). Hence, each test selected will address one core position during a given "time window" of the scenario considered. The alleged boundary conditions for the PWR core element of concern during the time window are provided by the VULCAIN assessment code ; they constitute a guide for setting the boundary conditions of the test train.

4.4.2. PHEBUS Experimental Facility

The tests are performed in the PHEBUS test reactor, comprising three main parts (Figure 11) :

. the driver core, which supplies nuclear power to the test fuel ;

. the SFD loop, which produces the thermohydraulic environment corresponding to a PWR core degradation phase, around the test fuel ;

. the pressurized water loop, which forms a buffer circuit between the SFD loop and the driver core.

4.4.3 PHEBUS SFD test matrix

Seven tests have been selected for implementation before mid-1989. They all use fresh fuel, and the corresponding PWR residual power will be simulated by fission power in the bundle. Three of them address the same "space window" (number 9 on Figure 10) of a PWR core during a small break LOCA with safety injection failure, assuming that cladding oxidation will largely dominate liquified fuel formation, which is the other major core degradation process.

. Test B9 was performed on December 3, 1986 ; its main features are a progressive fuel heat-up to 1800°C, with a large steam flow rate, then a fission power drop and helium cooling aimed at preserving the end state of the bundle for PIE. Some preliminary results are given in the next paragraph.

. Test B9R is scheduled for next March . Test conditions are similar to those of B9 but the cooling, which will be effected by a strong steam flow rate, possibly followed by a water droplet injection below a 1000°C fuel temperature, with the fission power continuing, in order to investigate the oxidized bundle behavior under some PWR typical cooling conditions.

. Test B9+ is expected to take place early in 89. Test conditions are similar to those of B9, but the fuel heat-up will be extended to 2200°C and more, resulting in the melting of unoxidized zircaloy. Bundle cooling will be ensured by a moderate steam flow, still with the fission power in progress.

Two of the seven tests will address "space windows" of a PWR core and scenarios where the formation of liquefied fuel is expected to be dominant vs cladding oxidation.

. Test C3 (scenario C, central volume on Figure 10) was carried out on October 30, 1987 ; its main design features were the following : a relatively rapid fuel heat-up under pressure with a continuous non-oxidizing flow rate (hydrogen), a small amount of steam injected over a short lapse of time to obtain a limited axial gradient of oxidized zircaloy, and a progressive cooling down from 1800°C by fission power drop and a helium flow. Actually, a pressure decrease occurred below 1100°C, which was compensated by an increased hydrogen flow rate ; although this event is still under investigation, the main objectives of the test are thought to have been met.

. Test B5+ is scheduled for June 1988. Test conditions up to 1800°C will differ only slightly from those of C3, but the system pressure, which will be low enough to prevent the collapse of the cladding on the pellets, resulting in an early formation of liquefied fuel. The experiment will be prolonged beyond the zircaloy melting point until there is an indication of significant fuel relocation or the instrumentation fails.

Two other tests are expected to have been completed by mid-89 ; currently they are not fully specified and will depend, to some extent, of the lessons learned from the previous tests of the series.

4.4.4 First results of the PHEBUS SFD B9 test

The test objective was to obtain a dominant cladding oxidation between 1000°C and 1800°C. Typical time histories obtained for the cladding temperature, system pressure, steam flow rate and linear power in the bundle are represented on Figure 12. The unexpected temperature plateau around 1500°C is under investigation ; there are some indications that it could result from hydrogen-enhanced radial heat losses through the zirconia thermal shield. Oxidation model predictions for the actual test boundary conditions are currently compared with PIE results. Figure 13 displays the state of the bundle 18 cm below the top of the test train : extensive cladding oxidation and inconel molten grid relocation are visible. Figure 14 is a micrography of the highly oxidized clad of fuel rod no. 2 of the outer row at bundle mid-height.

4.4.5 TMI-2 accident evaluation program

Within the framework of OECD/CSNI, CEA contributes to the DOE TMI-2 accident evaluation program ; core samples, including core bores, are currently examined at the CEA Saclay hot cells, while CATHARE calculations are performed to reconstitute the early stages of the accident ; further ICARE calculations are envisaged.

4.5 Off-site post-accident management

Although the probability of a severe accident involving unacceptable consequences is remote, an important program on the recovery of contaminated soils and water bodies has been initiated for returning to normal living conditions, which does not necessarily mean to living conditions prior to the contamination. This issue has been highlighted by the Chernobyl accident and is currently considered in national exercices, where a severe nuclear accident is simulated, in order to test emergency operating procedures on the NSSS as well as on- and off-site emergency plans and subsequent recovery actions.

The overall objective of the RESSAC program (Réhabilitation des Sols et des Surfaces après un Accident - radiological reclamation of contaminated land after an accident) is to provide technical elements for an improved assessment of the possibilities of land recovery after a radioactive contamination resulting from a severe PWR accident ; only natural soil whether cultivated or not, is considered at the present stage.

Such a program comprises three parts :

- A survey aimed at defining the scope of the studies : what are the type of land, the flora, the weather conditions, the variations of level of the water table, the radionuclides and the decontamination materials and techniques to be examined ?

- Outdoor experiments to test the capabilities of standard equipments and techniques used in agriculture or road construction, for removing plant materials and contaminated soil surface scraping ; minor technical modifications are sought, which could result in an improved decontamination efficiency.

- Indoor experiments aimed at investigating radioactive matter transfers in the soil itself (water table contamination problem) and from the soil to the plant : the parameters examined are : the radioactive substance (Cs, Sr or Ru), the soil, the weather conditions, the level of the water table, the flora, the cleaning methods and the time. Low cost trials in small containers are currently carried out ; from 1990 on, integral tests in lysimeters, four m2 surface area, 1.5 m in depth, are contemplated, in which fission products of concern will be used.

Other issues could be included in the RESSAC program in the future : among them is the reclamation of contaminated, constructed areas and the management of the solid waste generated by decontamination.

5. 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. 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") on the utility side, the particular (off-site) emergency plan (PPI, for "plan particulier d'intervention") on the side of the government representative at the "département" level - and a centralised organization at the national level.

5.1 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 "équipe 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 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 (PCM for "PC Moyens") is in charge of activating on-site intervention means and evacuating the plant staff when appropriate ; the PCM can be withdrawn into the BdS if necessary.

- The emergency management team for measurements (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 a technical support of SCSIN ; continuous connections between the three technical assessment teams permits the analyses to be compared and synthesized (Fig. 15).

5.2 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, coordinated by the "Secrétariat Général du Comité Interministériel de la Sécurité Nucléaire" - SGSN for short- (secretariat of the inter-department committee for nuclear security, at the prime minister level). This organization includes the SCSIN and its technical support, the CEA/IPSN, the SCPRI and the "Direction de la Protection Civile" (civilian protection branch of the department of the interior).

5.2.1 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 at Paris.

- The emergency management team (national-level PCD)

Such 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. 4).

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

Its role is to supplement the information of the above PCD and 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.

5.2.2 The emergency organization of SCSIN

Three teams are constituted in case of an emergency

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

- The emergency technical team is located on CEA/IPSN premises at Fontenay-aux-Roses, near Paris ; it is chaired by the director of IPSN

- 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 in 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.

About a dozen 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.

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.

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 Inorganic	Organic	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
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.

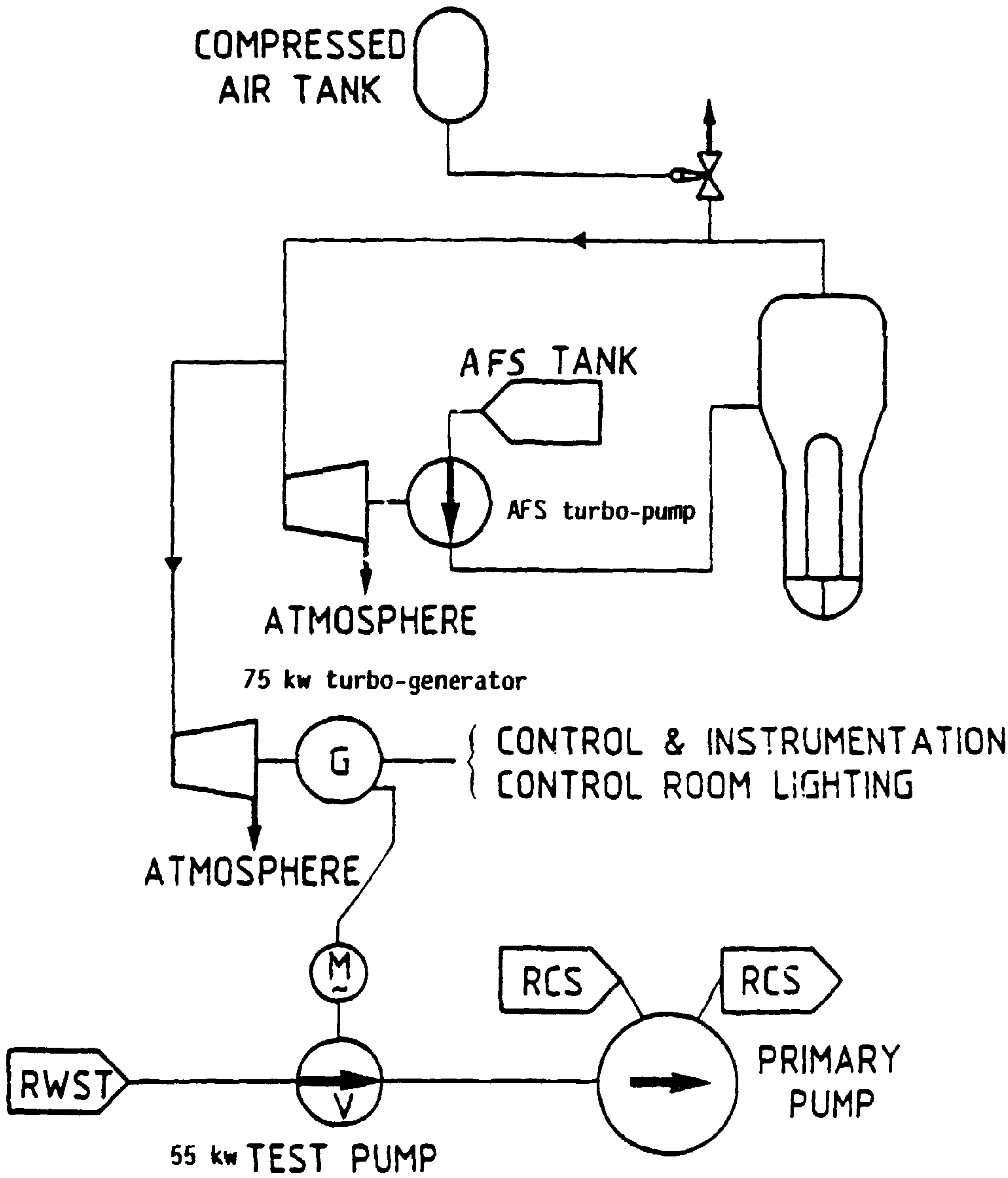
Table 2

characteristics of the fluid to be filtered

– maximum mass flow:	3.5 kg/s
– composition by weight:	air 33% CO_2 33% H_2O vapour 29% CO 5%
– temperature:	140°C
– pressure:	close to atmospheric pressure
– aerosol content: (of whatever origin)	maximum concentration 0.1 g/m^3 median diameter in terms of mass 1 μm total quantity 5 kg

FIGURE 1

H3 PROCEDURE



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

FIGURE 2

U1 PROCEDURE

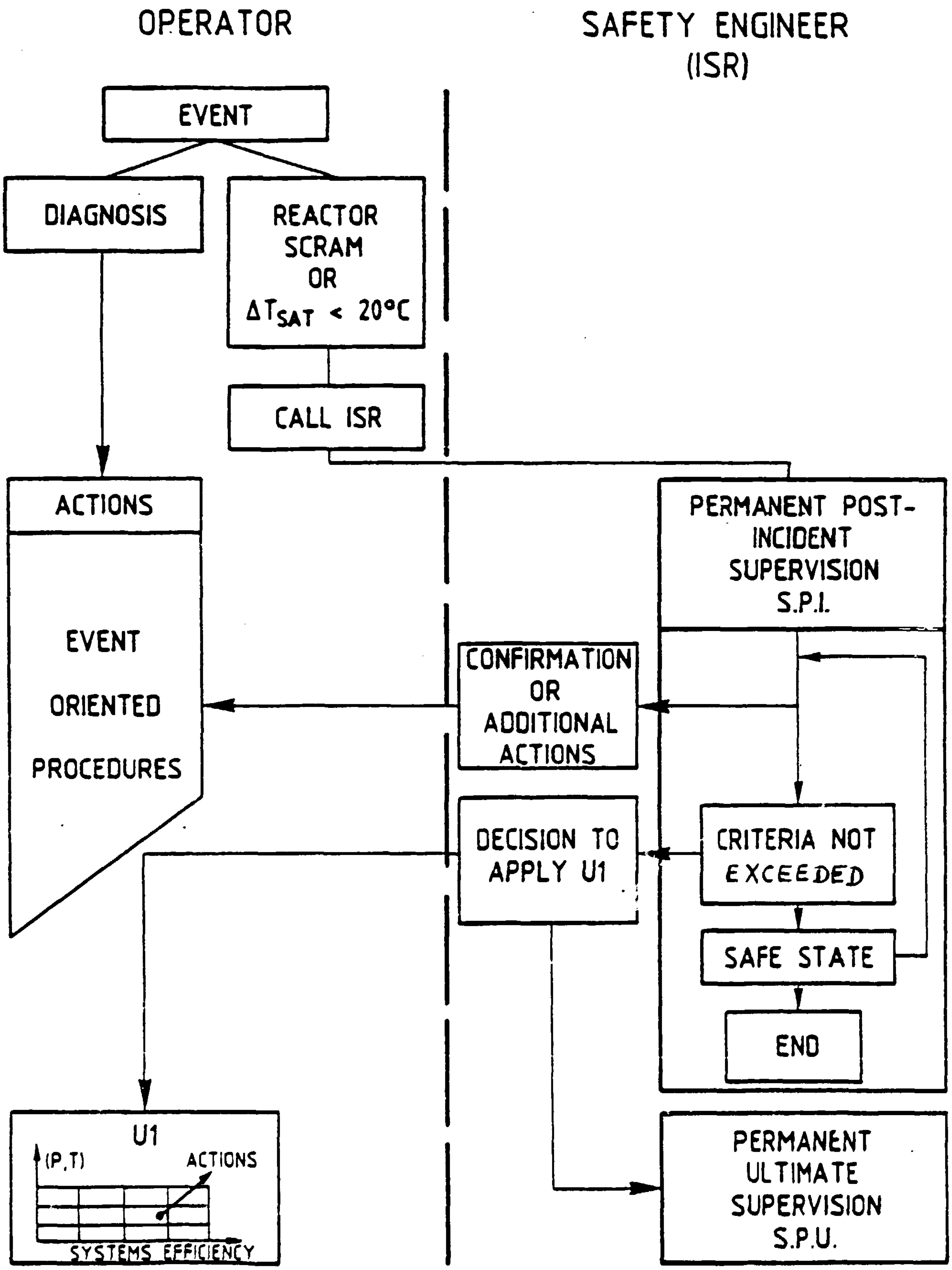
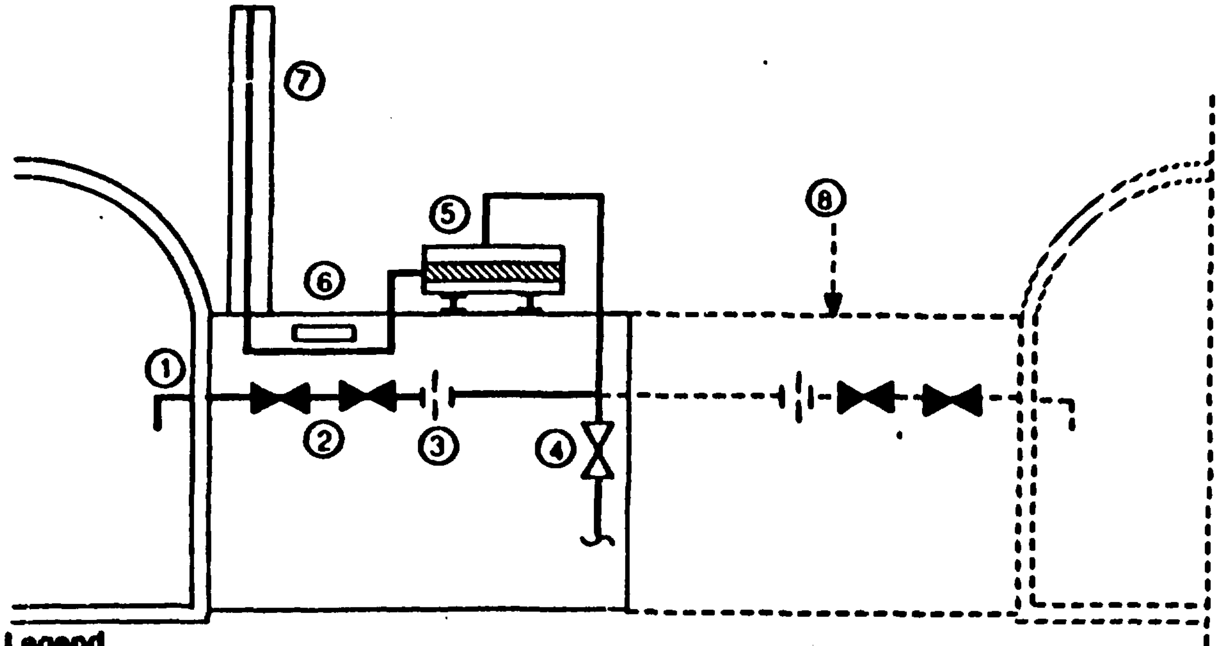


figure 3

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 o ration.
5. Sand filter.
6. Radiation monitor.
7. Plant stack, with small vent duct inside
8. Arrangement for twin units.

U5 - FILTRE A SABLE

U5 SAND BED FILTER

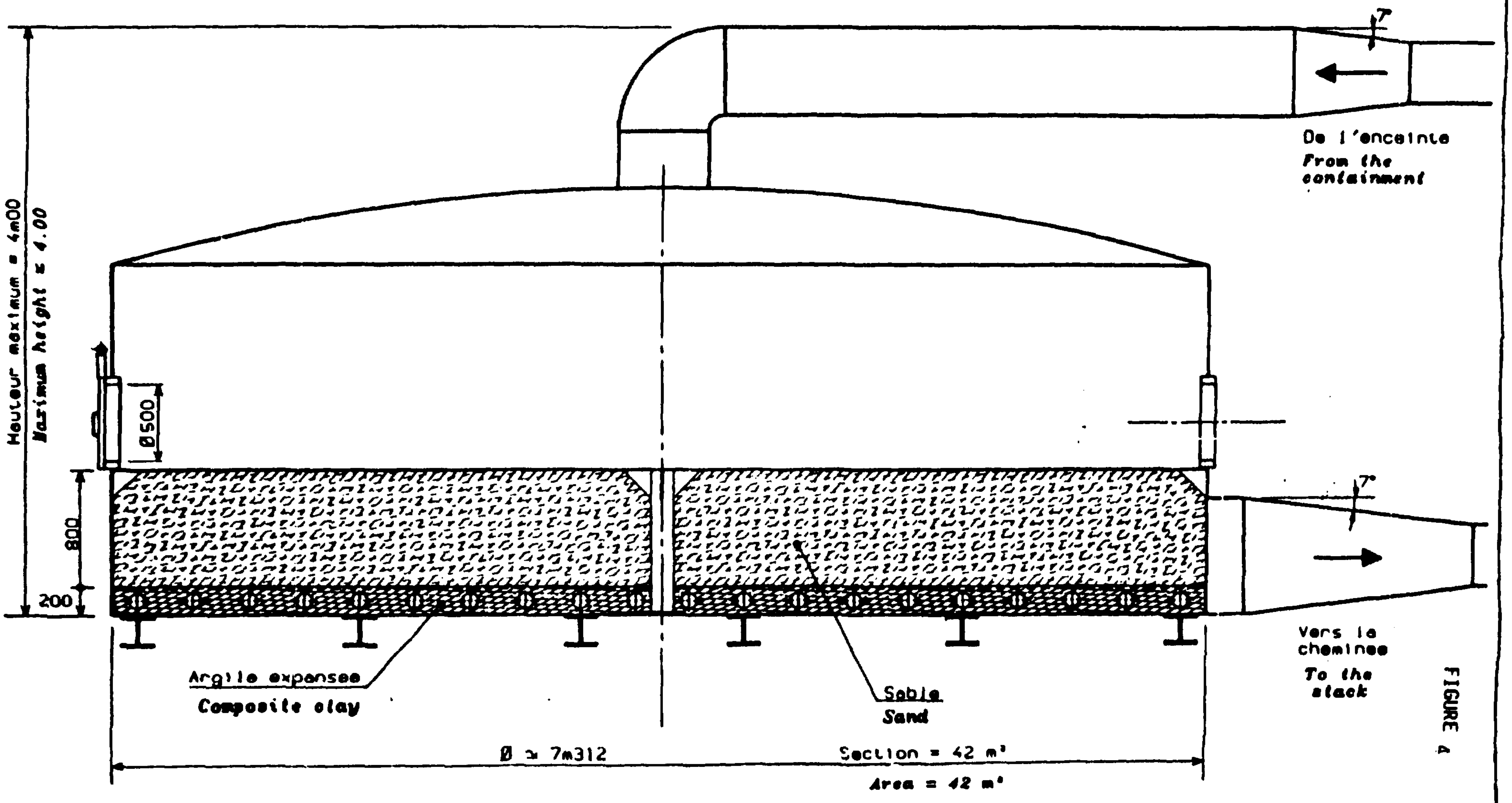


FIGURE 4

FIG. 5

Bethsy Integral Test Facility

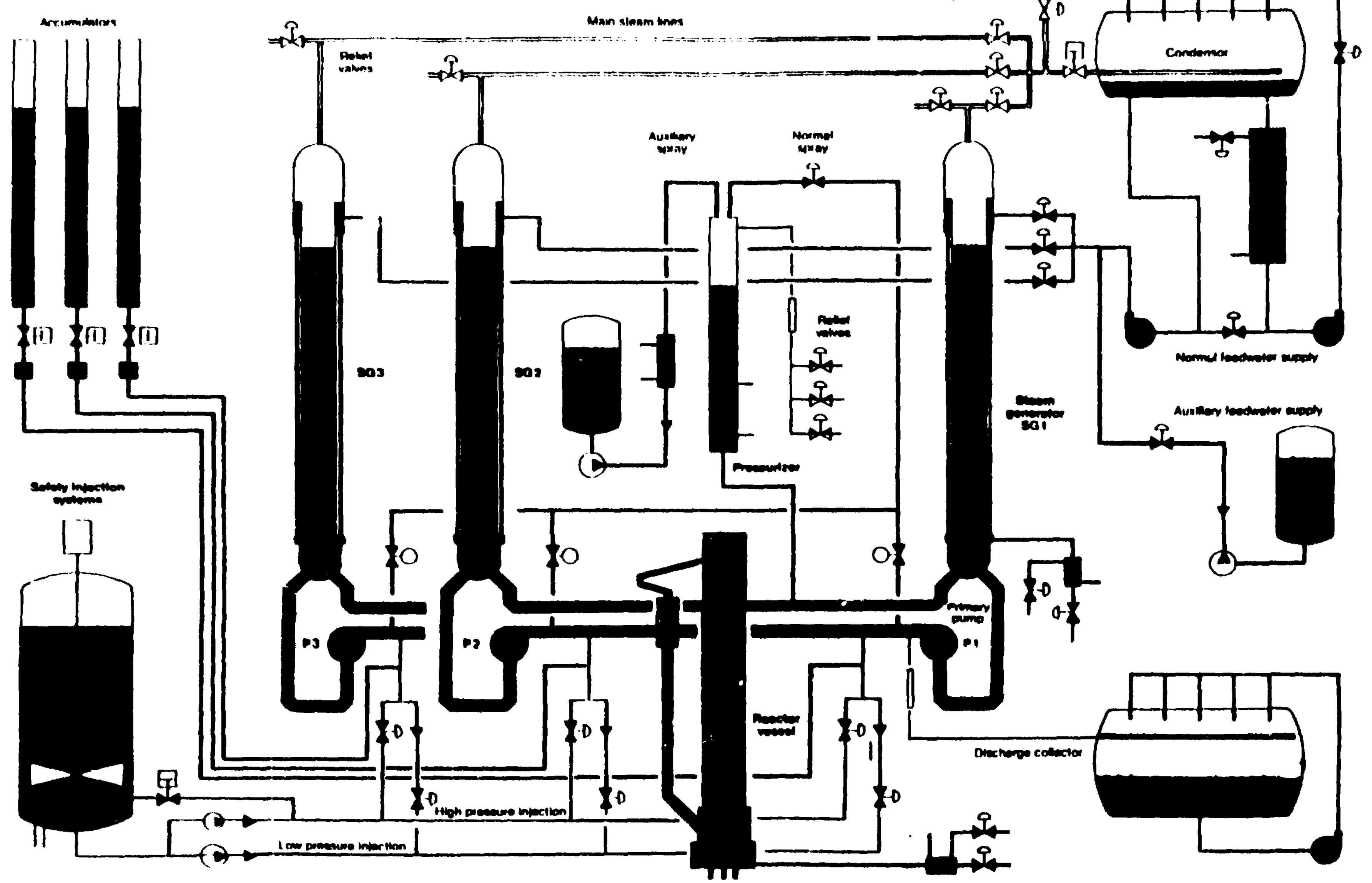
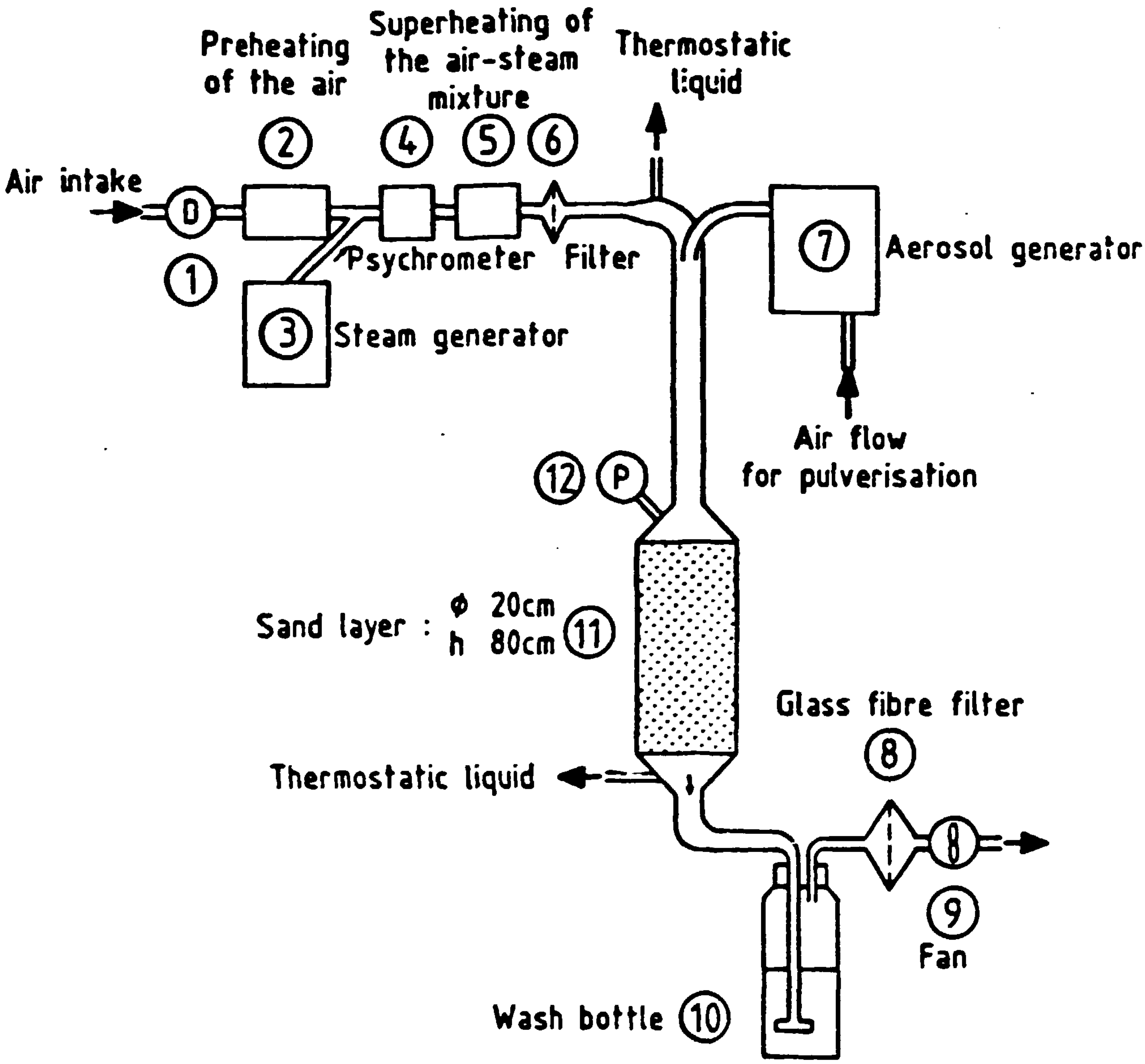
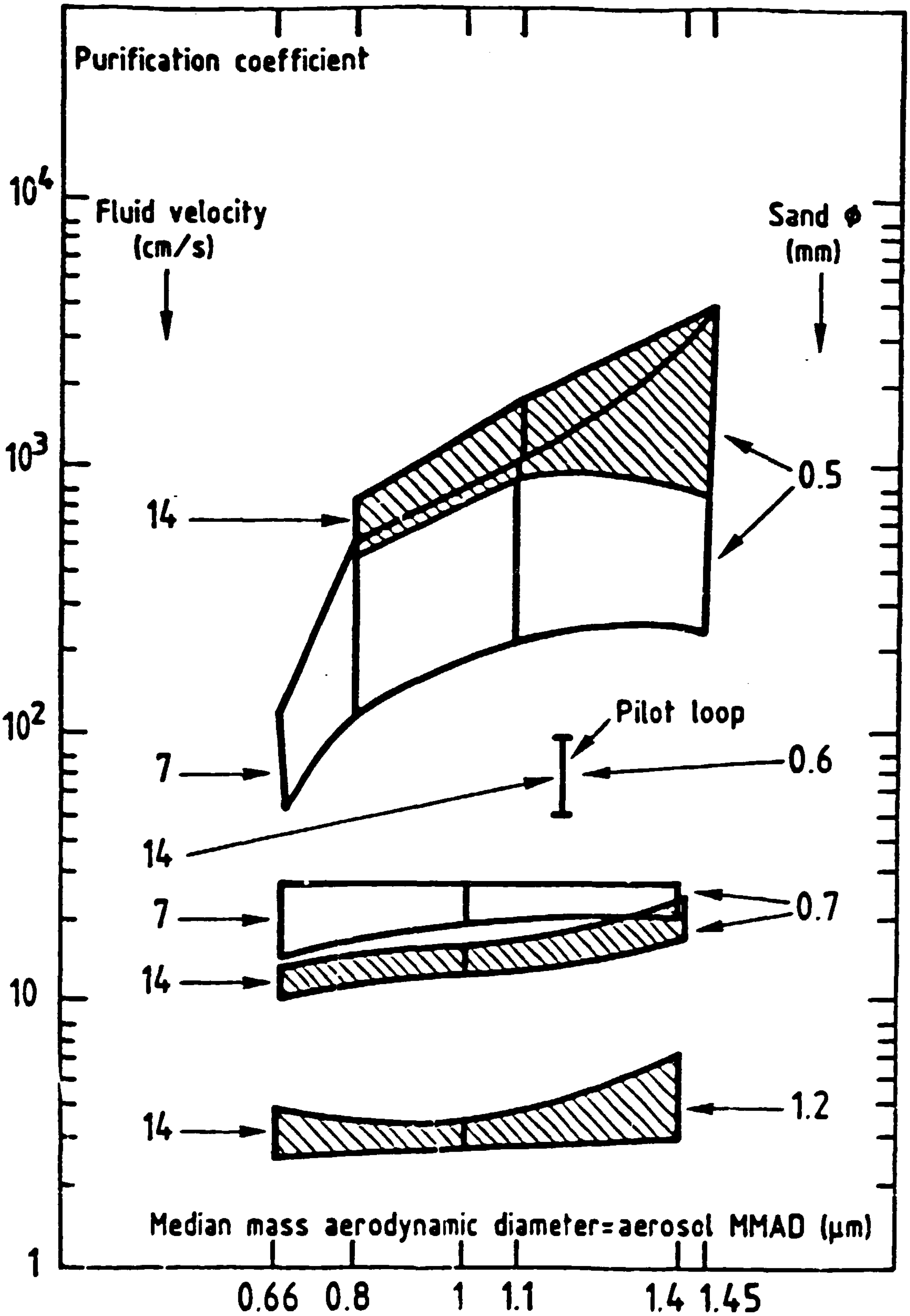


figure 6



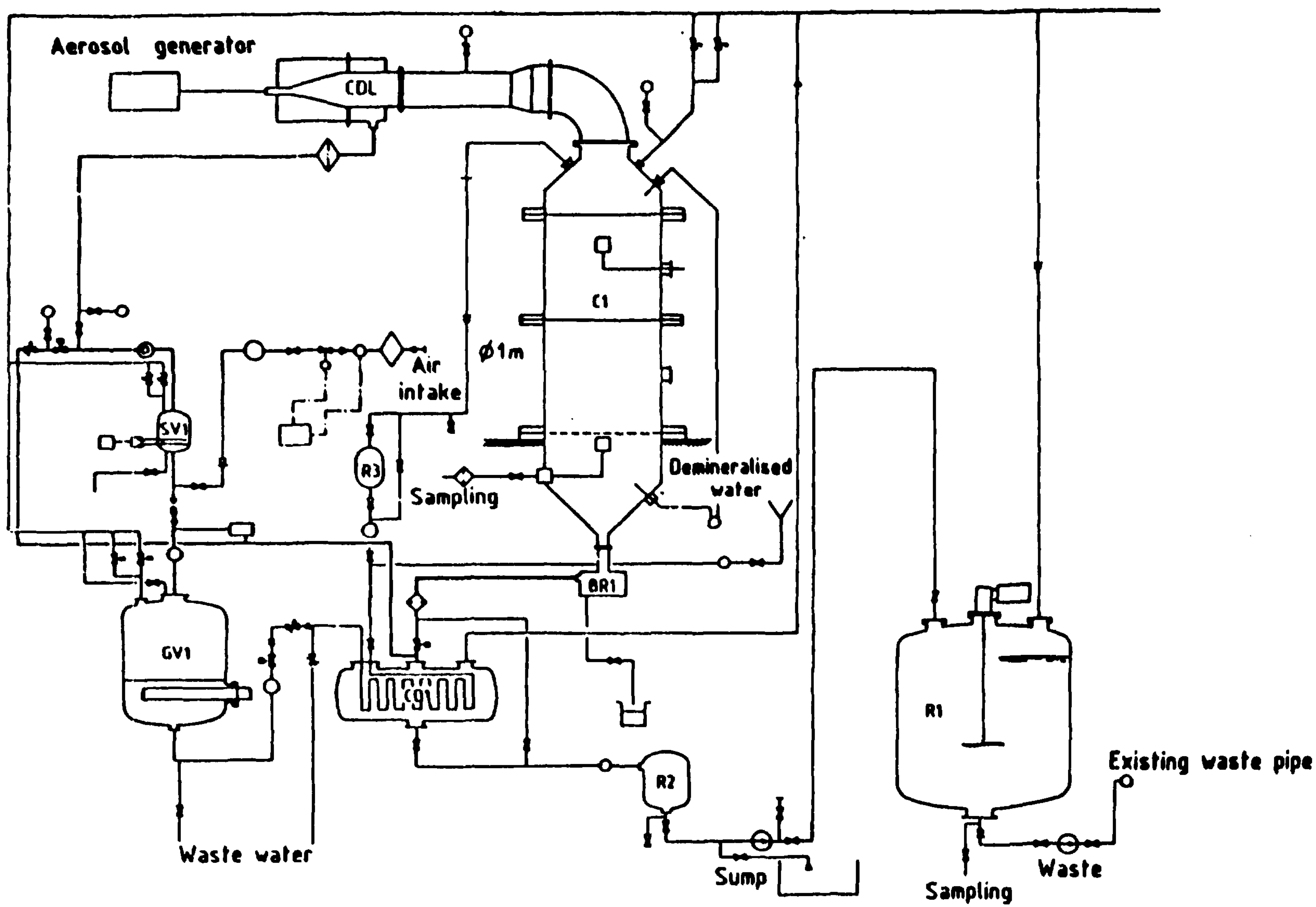
- Layout diagram for laboratory filtration (glassware)

figure 7



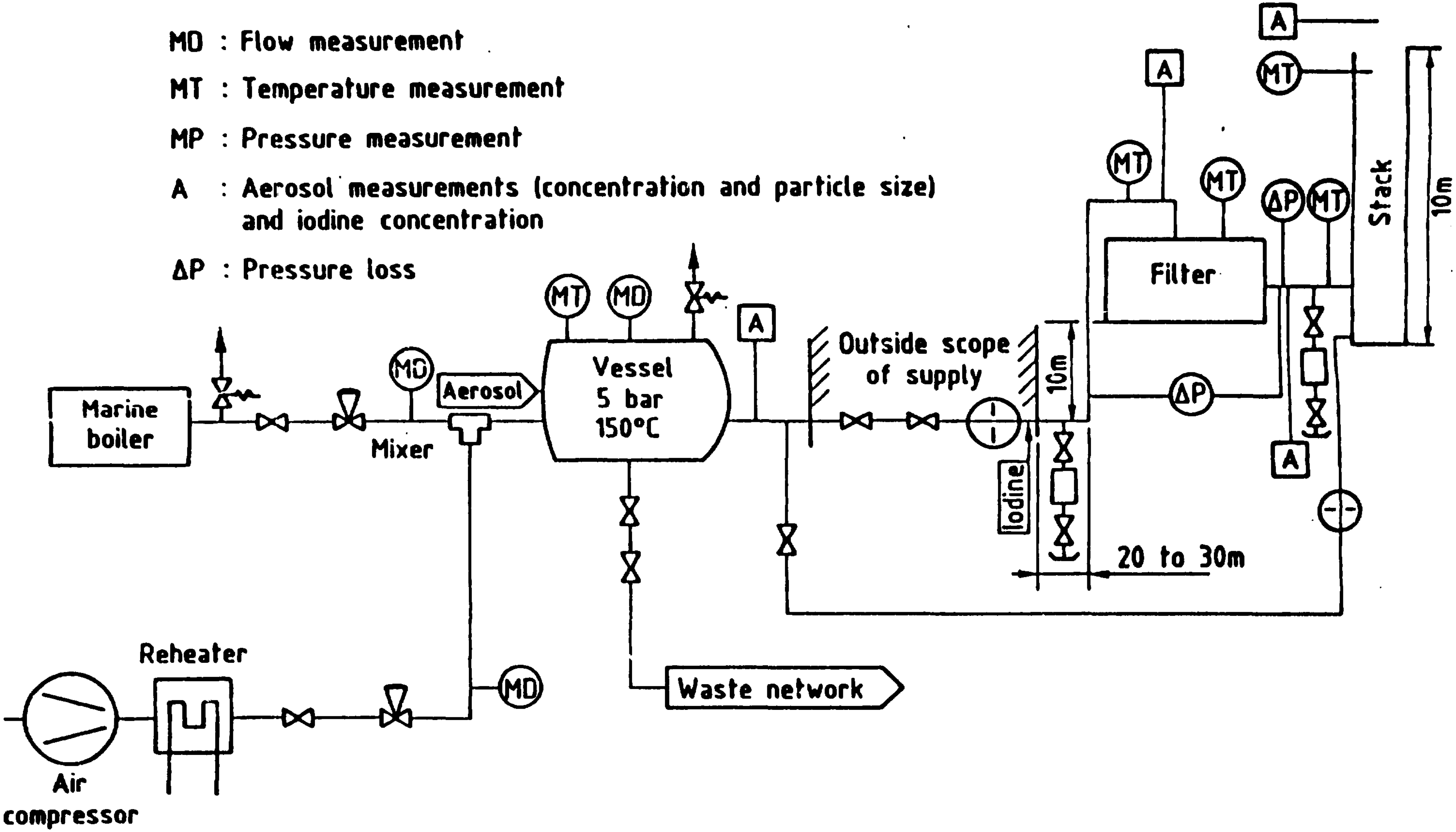
- Purification coefficient - Test under steady-state operating conditions

figure 8



- Diagram of pilot test loop

figure 9

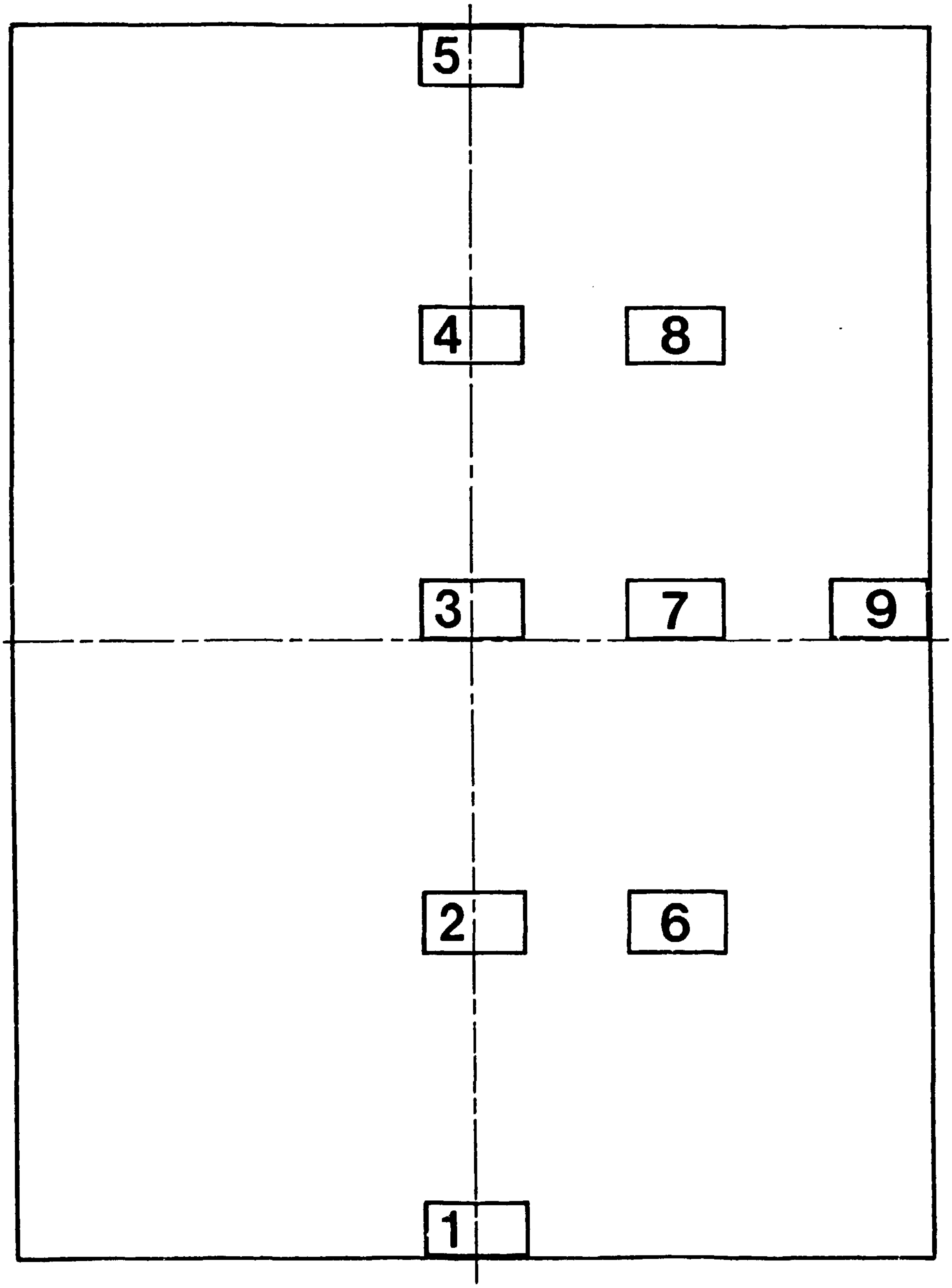


- *Schematic diagram for test installation (industrial scale)*

CEA

Fig. 10

LOCATION in the PWR CORE of the NUMBERED REGIONS

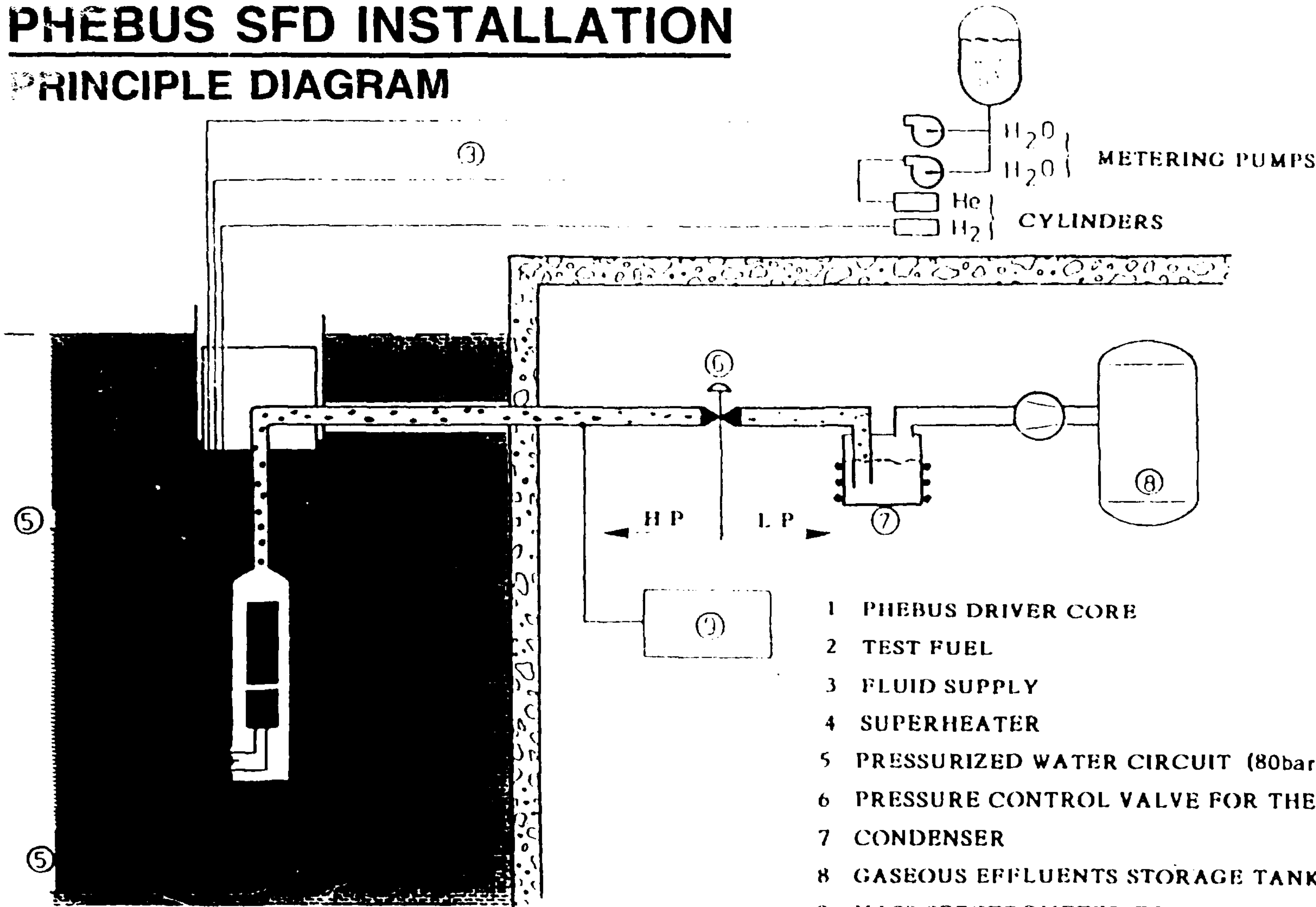


CEA

PHEBUS SFD INSTALLATION

PRINCIPLE DIAGRAM

Fig. 11

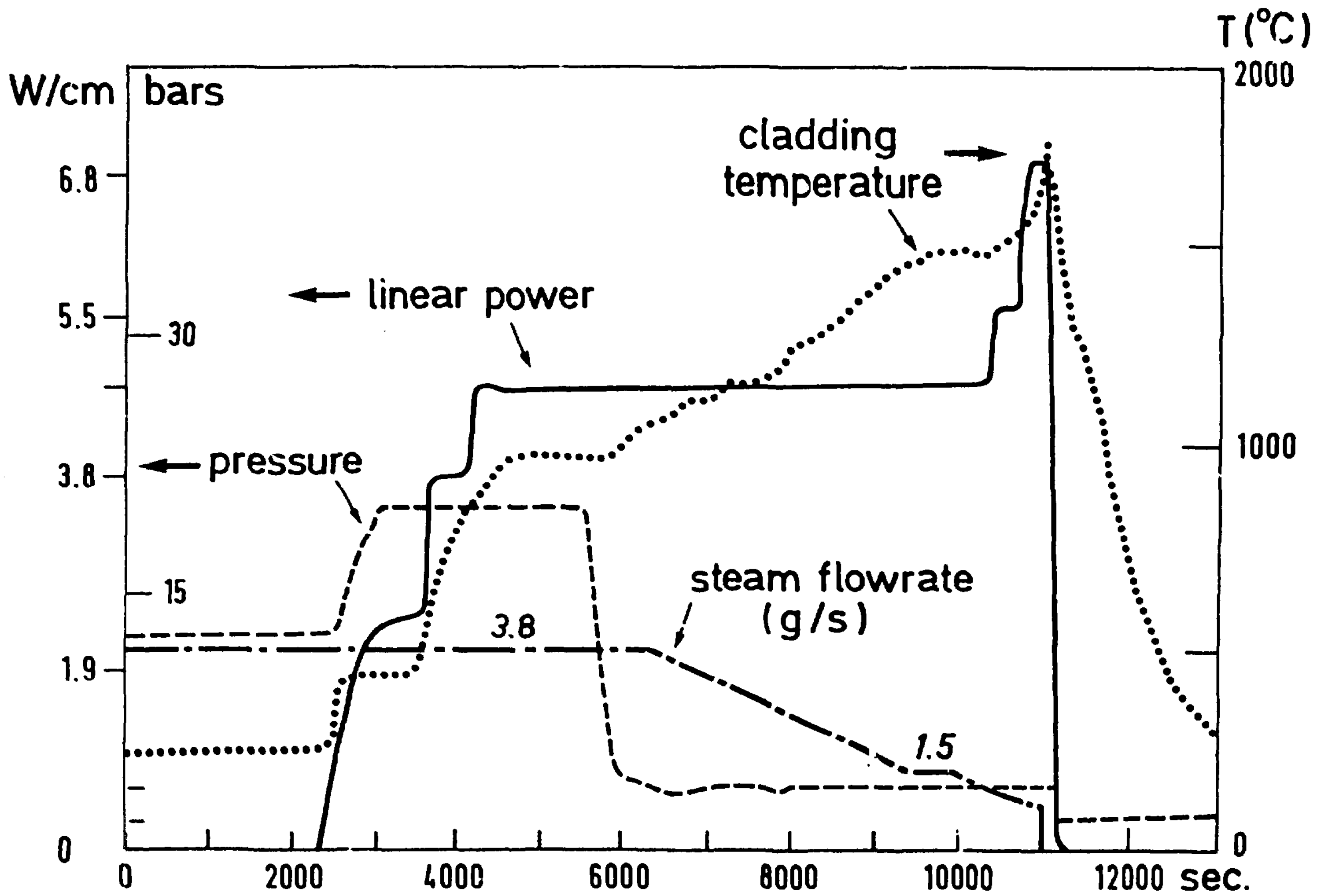


1 PHEBUS DRIVER CORE
2 TEST FUEL
3 FLUID SUPPLY
4 SUPERHEATER
5 PRESSURIZED WATER CIRCUIT (80bars - 280°C)
6 PRESSURE CONTROL VALVE FOR THE SFD LOOP
7 CONDENSER
8 GASEOUS EFFLUENTS STORAGE TANK
9 MASS SPECTROMETER FOR CONTINUOUS H_2 ANALYSIS

CEA

Fig. 12

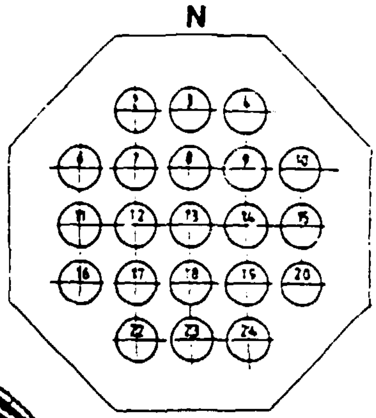
PARAMETER EVOLUTION for TEST B9

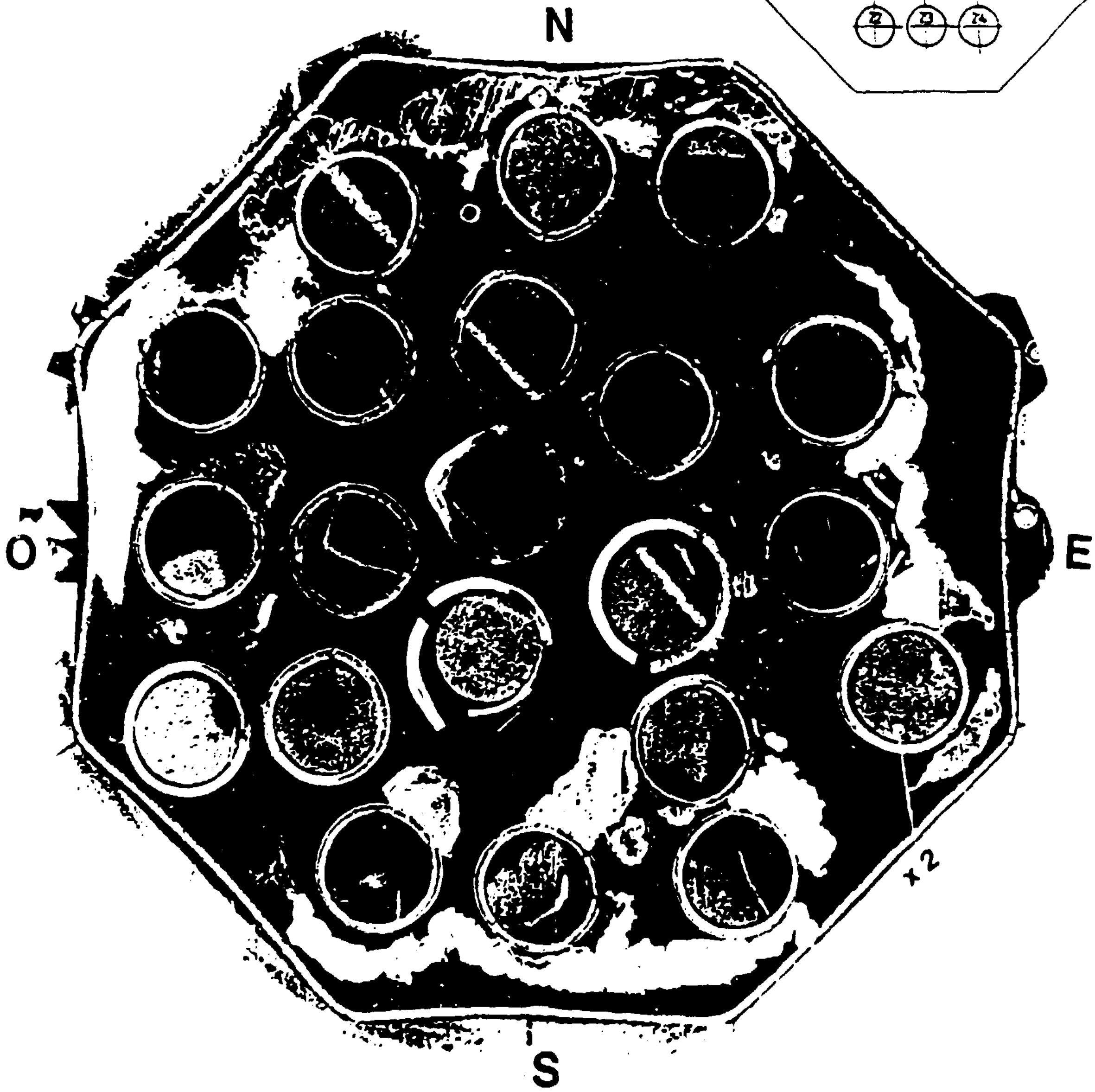


CEA

Fig. 13

PHEBUS SFD
PROGRAM TEST. B9





Macrography at elevation – 7281

Fuel rod elevations { top : - 7100 mm
bottom : - 7900 mm

PHEBUS SFD PROGRAM TEST B-9

ROD # 2 MICROGRAPHY -7460 mm (73 %) × 100

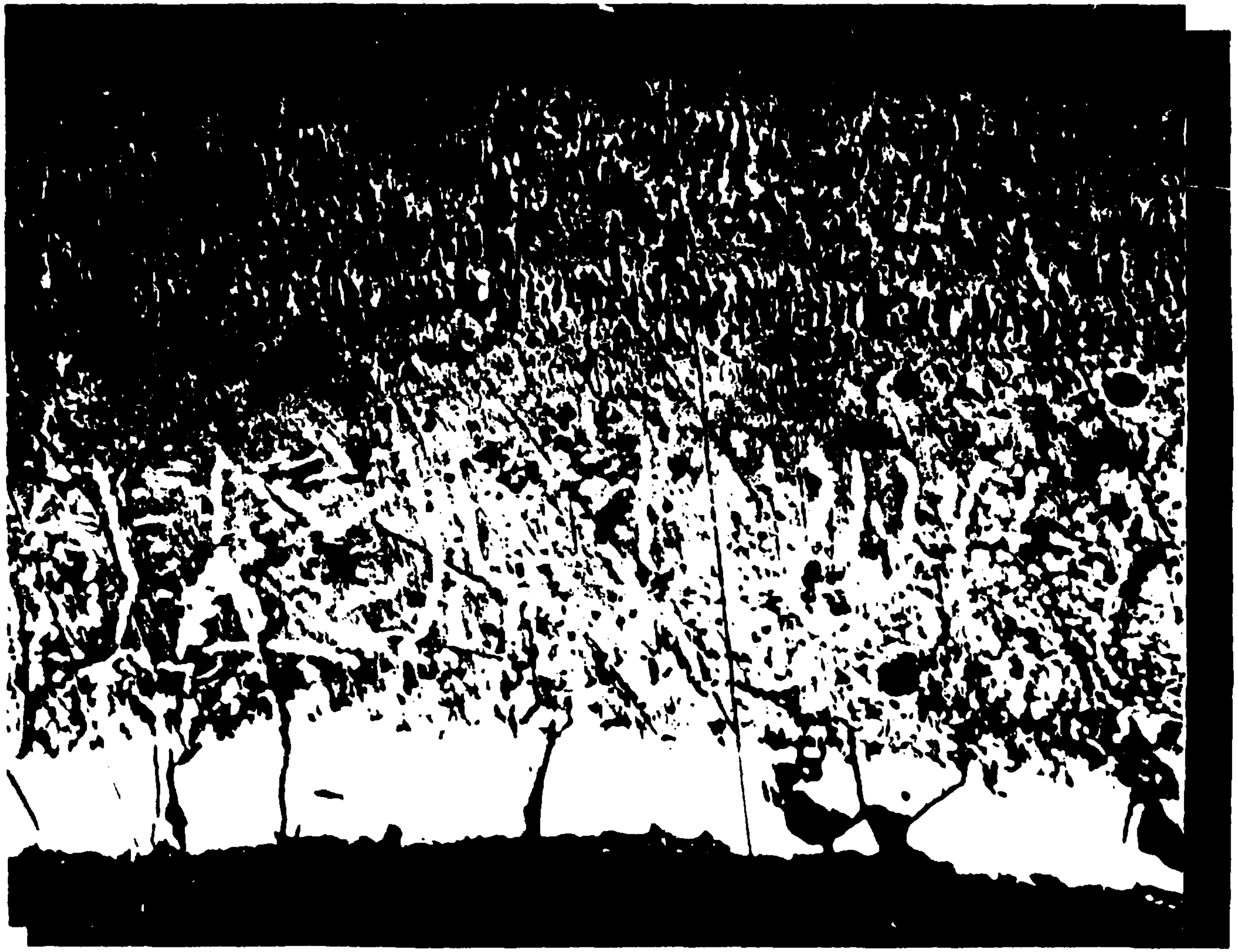


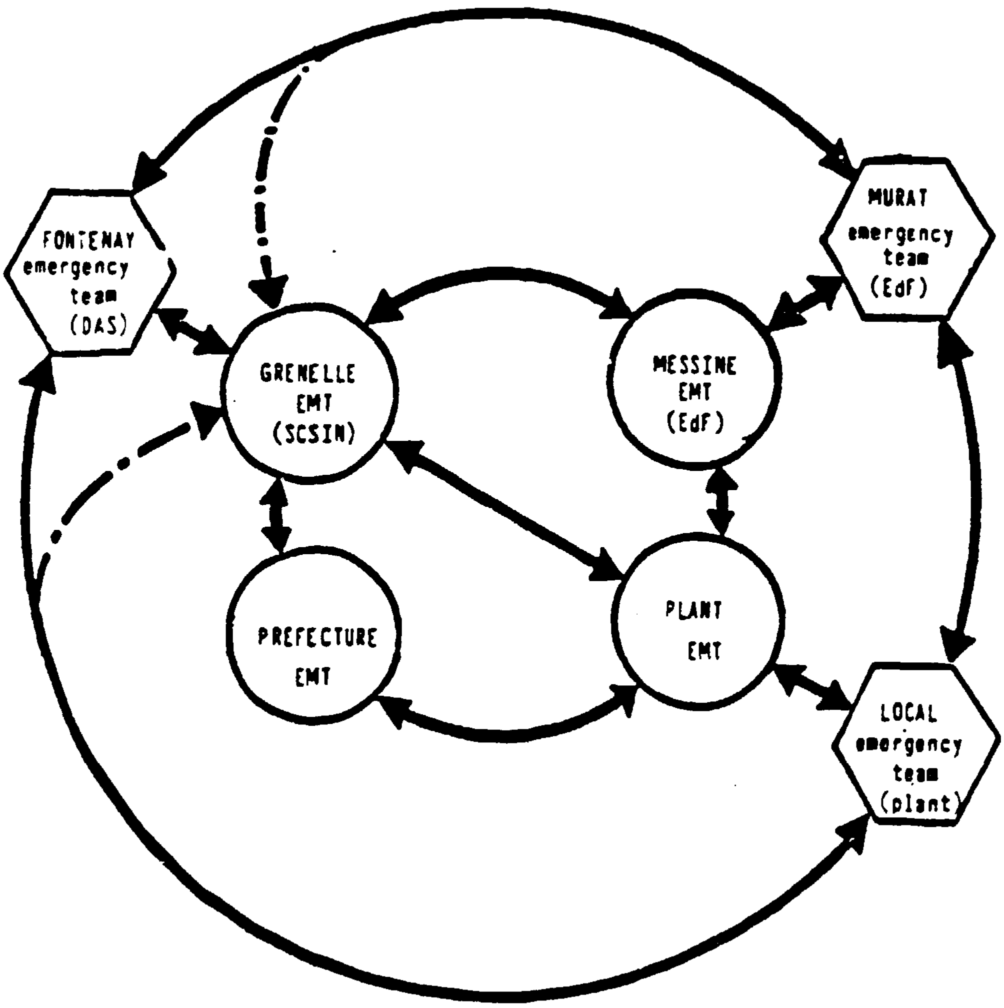
FIG. 14

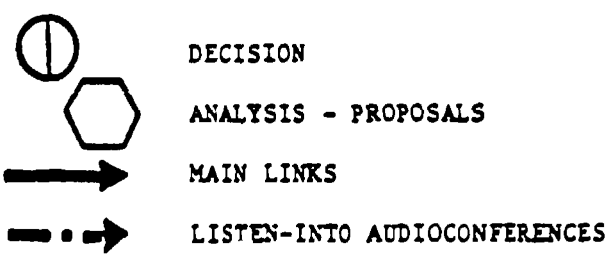
cea

FIGURE 15

ORGANIZATION DURING AN ACCIDENT

IN AN ELECTRICITE DE FRANCE REACTOR





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