REPUBLIQUE FRANÇAISE

AUTORITE
DE SURETE
NUCLEAIRE

Direction de la sûreté des installations nucléaires

French Nuclear Safety Authority

DSIN-FAR/SD6/N° 01-60 119

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Fontenay-aux-Roses, March 26th, 2001

Dear Mrs Dunn-Lee,

Following a DSIN's request, the Institute for Nuclear Safety and Protection has initiated a comparative technical evaluation of several advanced nuclear power plant projects with the objective to notably identify relevant advantages with regard to safety.

The USNRC has kindly provided DSIN with documents related to the 80+ and AP 600 projects that have been reviewed by IPSN. The review raised a number of technical issues which are addressed in the questionnaire attached to my letter.

Answers to those issues and questions would be extremely valuable for the completion of the IPSN technical evaluation which will be then submitted to the Standing group of experts on reactors for advice to my Directorate.

Therefore, I would be very grateful if the USNRC could assist IPSN in clarifying the issues, at its best convenience.

When completed, I will be very glad to provide the USNRC with the conclusion of the review.

Many thanks in advance for your help in this matter.

Sincerely yours

A-C. LACOSTE
Director

Mrs. J. Dunn-Lee
Director, Office of International Programs
US Nuclear Regulatory Commission
Washington, DC 20555

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ANNEX

1 SYSTEM 80+

The Institute for Nuclear Safety and Protection (IPSN) constitutes the technical support of the French Nuclear Regulatory Body (DSIN). Within the Safety Evaluation Department of IPSN, the Risk and System Evaluation Branch (SERS) has initiated a comparative analysis between design solutions proposed for several advanced nuclear power plant projects.

In addition to the System 80+, the concepts selected include passive concept AP-600 developed by Westinghouse, the common French-German project EPR, the Russian advanced reactors WWER/1000/V-392 and WWER-640/V-407.

The analysis focuses on the assumptions taken, the criteria retained and the requirements applied in relation to several safety key issues.

One of the objectives of the study is to identify relevant advantages of selected concepts with respect to safety, operating simplicity and reliability.

At this time, with respect to the System-80+, our assessment is based on the review of the Standard Safety Analysis Report (SSAR) as well as the Final Safety Evaluation Report (FSER) you already provided to us.

The review of this material has raised some preliminary remarks and comments. For each selected key issue, the following paragraphs present our questions in order to clarify technical items.

At your convenience, this questionnaire might be used as a guide for a technical exchange to be developed at a future technical meeting.

1.1 General Safety Approach and associated objectives

Question 1.1
What is the approach followed to set up the radiological releases criteria for fault situations ? What are the results obtained ?

Question 1.2
Could you explicit the various probabilistic objectives associated with a controlled core melt :

- reactor states considered (including shutdown states),
- external events taken into account,
- low pressure / high pressure core damage states,
- core melt with containment bypass ?

1.2 Classification of equipment and components

Question 2.1
Could you be more explicit about the content of requirements applied to the various safety classes,

especially with respect to :

- implementation of the single failure criterion;
- application of redundancy and diversity on the functions and systems levels;
- physical separation of redundant trains;
- emergency power supply by the emergency diesel generators;
- periodic testing ?

1.3 Treatment of the guillotine primary break

It seems that you have taken various assumptions with respect to the postulation of large primary breaks. To complete our understanding of this approach, some complementary information is needed, as well as additional explanation concerning the justification of the different break sizes postulated.

Question 3.1
How is defined the boundary break for the reactor internals ?

Question 3.2
What is the level of conservatism for the assumptions taken for the containment and the Safety Injection System design, as well as for the equipment qualification and for the radiological consequences ?

Question 3.3
What is the classification of the equipment needed for the detection ? How the "low probability" of the break is demonstrated using the "deterministic" mechanical fracture approach ?

Question 3.4
The stratification phenomena as well as the frequency of temperature cycles (fatigue) of the surge line are difficult to describe; how its break can be excluded ?

1.4 Containment design

Radiological consequences

Question 4.1
How the design leakage rate of 0.50% volume per day for the first 24 hours is met during all the plant's life?

Question 4.2
Is the design leak rate of 0.50% volume per day applied to the steel containment, to the shield building or both?

Question 4.3
Why no credit has been taken for the carbon absorbers in analyzing the consequences of a design basis accidents?

Containment structure

Question 4.4
How is ensured the mechanical strength of the steel containment vessel over the 60-year life of the plant principally in peculiar zones (penetrations...)? How is taken into account the fact that no failure will occur?

Question 4.5
How do you proceed to control the leaktightness of all containment electrical penetration assemblies? Are they pressurized and under which pressure?

Question 4.6
How will be performed the different tests of the electric penetration assemblies (in accordance with IEEE-317-1983)?

Design-basis accident

Question 4.7
Does the System 80+ design take into account the steel containment movement following a design basis accident (LOCA...)?

Question 4.8
The steel containment vessel is designed to withstand the conditions of pressure and temperature following the most restrictive design basis LOCA. Has a containment design for beyond-design basis accident been analyzed for the System 80+?

Annulus Ventilation System (AVS)

Question 4.9
The AVS is designed to maintain a negative pressure of -0.5 in. water gauge during a LOCA.

- What's your basis for this value?
- What kind of atmospheric condition is taken into account?

Question 4.10
How has the exhaust flowrate of 16000 cfm after a LOCA been chosen?

Question 4.11
Each filter train consists of a moisture eliminator, electrical heater, prefilter, an absolute filter, a carbon filter, and a post filter.

- What is the useful of the post-filter?
- What is the technology of the moisture eliminator?

Question 4.12

What the AVS strategy during severe accidents?

Question 4.13
Concerning the severe accident management, the containment is vented after 24 hours the onset of core damage. Why is this venting not filtered?

1.5 Main safety systems design required for heat removal

Shutdown Cooling System (SCS)

Question 5.1

How the containment bypass risks are taken into account? Could you be more specific about design provisions, detection and mitigation means?

Question 5.2

We have noticed that mid-loop operation cannot be excluded on system 80+ as operating state for steam generator inspection. It seems on the contrary that reduced inventory operations are avoided for nitrogen sweeping and steam generator tubes draining. Could you explain to us how and when do you perform these functions ?

Safety Injection System (SIS)

Question 5.3

Concerning the Safety Injection tanks (SITs),

- Would you like to confirm that no failure of the SITs is assumed ?
- In case of failure of the isolation of SIS accumulators, how do you consider the impact of the cold nitrogen on the reactor vessel and the connected lines, as well as the influence on the clad temperature and on the heat transfer via the steam generators ?
- Concerning the small break LOCA, we have noticed that the SITs are vented or isolated in establishing Shutdown Cooling conditions for Small Break Long Term Cooling procedure. Could you be more specific about the need of such an operation ?

Question 5.4

As regards the large-break LOCA, it was drawn that the maximum containment spray flows is the most penalizing case. For that reason, all active part of the SIS and spray systems were considered as effective in the frame of the accident analysis. So could you explain how you apply the single failure criterion to the safety injection system ?

Question 5.5

Following specific initiators like small primary breaks, the heat may be removed via the steam generators in the reflux condenser mode. The possibility of recriticality should be considered. Do you consider this risk of an inherent heterogeneous dilution ?

Question 5.6

As the system 80+ do not include low pressure safety injection system, we have understood that the primary make-up function in case of LOCA is carried out by throttling at the output of the high pressure safety injection pumps. This solution constitutes a particular case in comparison with other advanced PWRs. Could you be more explicit about : the accidental procedure followed and specific devices implemented (including corresponding I&C, experience feedback and reliability parameters) ?

Emergency FeedWater System (EFWS)

Question 5.7

In the safety analysis, is a passive failure applied in the long-term period (24 hours after the occurrence of the accident) ?

Question 5.8

We have noticed that the motor driven emergency feedwater pump motors are cooled by a portion of the CCWS; so the EFWS will be dependent on the CCWS. Could you confirm?

- With regard to the diesel generator engine, it receive makeup water from the Demineralized Water System and uses as it's sink the Component Cooling Water System. Could you confirm that the Diesel Generators depend on the CCWS?

Question 5.9

Consequently of a feedwater-line break assumed with a loss-of-offsite power, have you considered the formation of a steam blanket in the ruptured SG tube and the stop of the depressurization of the RCS? As a result, are the size of the EFWS tanks designed due to this phenomenon?

Component Cooling Water System (CCWS)

Question 5.10

The spent fuel pool cooling heat exchangers are isolated automatically on a safety injection actuation signal. This automatically isolation is likely to lead to a transient in the spent fuel pool with a possibility for boiling. How is this situation dealt with whereas the reactor is already experiencing an incident or accident?

Question 5.11

Could you explain the reasons why the CCWS major portion is constructed with carbon steel and how is this consistent with the use of the Station Service Water System as a water makeup following an accident?

Station Service Water System (SSWS)

Question 5.12

Usually, heat sink loss initiating events taken into account in PSA studies include the loss of the ultimate heat sink (to be assessed by expert judgement and depending on configuration site) and loss of flow in the cooling systems (mainly due to failure of the pump sets or electrical power supply, leakage or fouling of the lines). Could you be more specific about the characteristics of the loss of the ultimate heat sink (water intake of the pumping station) considered for your PSA calculations :

- annual probability of occurrence,
- mean repair or recovery time associated to it,
- causes (external hazard, fouling, ...) ?

Containment Spray System (CSS)

Question 5.13

24 hours after the onset of a severe accident, a non-safety emergency containment spray system (ECSBS) can be used to inject water to the CSS by the way of an external backup source. Could you explain design and operation of this system? Besides, could you confirm that this specific system is totally independent on the CCWS/SSWS chain ?

1.6 Single Passive failure criteria

Question 6

Could you provide us with some complementary explanation concerning the following items :

- the complete list of systems designed according to the passive single failure,
- the justification of the non-consideration of short term passive failures,
- the determination of a technically appropriate value for the passive failure flowrate,
- with respect to the detection and isolation, the requirements applied to the equipment needed to cope with postulated passive failures ?

1.7 Electrical power supplies, distribution and response to station blackout

Question 7.1

The section 9.3.4.2 of the CESSAR specifies that the CVCS pumps and the Dedicated Seal Injection System (DSIS) pump can be powered by the EDGs. How this alignment is possible, considering that the charging pumps and the DSIS are connected to the X and Y Permanent Non-Safety Buses ?

Question 7.2

It is difficult for us to understand completely the load shedding and sequencing. Could you give more pieces of information about it ?

Question 7.3

According to various parts of the CESSAR, it is not clear whether the activation and the connection of the AAC is performed manually or automatically.

Station Blackout

Several issues still need to be clarified. It concerns mainly :

Question 7.4 the definition of the SBO coping time,

Question 7.5 the need and conditions for resupplying the emergency feedwater tanks (use of alternate sources to meet the postulated coping time),

Question 7.6 the requirements applied to the AAC power source :

- design basis (DBA+SBO more severe than 10CFR52 request),
- minimal delay for making it available (2 or 10 min after LOOP or SBO ?),
- mode of connection (manual, automatic from the MCR ?),
- consumers supplied,
- reliability assurance and operability at shutdown conditions,

Question 7.7 the behavior of the RCP seals under black-out conditions,

Question 7.8 the assumptions related to the pressurizer safety valves (opening/closing),

Question 7.9 the management of station blackout under shutdown conditions, especially for long term unavailability of offsite power with the reactor coolant system opened,

Question 7.10 the PSA results and some important study assumptions :

- period of calculation,
- maximal recovery time for the external electric power (consistent with the fuel and feedwater capacities ?),
- reliability data used for the Emergency Diesel Generators,
- behavior of the primary safety valves,
- delay available before core melt.

1.8 External and internal hazards

Protection approach for internal hazards

Question 8.1
What are the main objectives of the design against internal hazards ? More precisely :

- how the principle of non-aggravation of an initial event is applied (e.g. if an initiating event belong to category of « incidents » it has not deteriorate in the category of « accidents » due to the effects, as for example, due to the whipping of the initially broken pipe which could impact a neighboring pipes) ?
- are there some radiological design targets to be met within the internal or external hazards analysis ?

How the consequential hazards are taking into account?

- Internal hazard as a consequence of an External hazard (e.g. fire induced by an airplane crash)
- Internal hazard as result of other Internal hazard (fire induced by explosion and vice versa)

Question 8.2
Is an Internal hazard considered in the accident analysis as a single failure?

Question 8.3
Demonstration of non-propagation and of non-aggravation:

- what is classification of systems considered in the analysis of internal hazards ?

Question 8.4
Due to their impact on the plant, are different internal hazard analyzed in distinguished way (independent of accidents, leading to events and as a consequence of design basis events or Beyond Design Basis Events) ?

Question 8.5
We understood that if an internal hazard is postulated in one safety division, no single failure is applied on the redundant safety train. Is there some probabilistic justification of it ?

Question 8.6
How a functional analysis for internal hazards (notably in the building where the separation is not ensured, e.g. Reactor Building) is realized ?.

Question 8.7
What is the scope of buildings checked by an analysis of internal hazards ?

Question 8.8
What are the initial states where an internal hazard can be postulated (only power operation state or the shutdown states too) ?

Question 8.9
For an independent internal hazard (which does not lead to the plant initial state deterioration) :

- Is a loss of more than one safety train (due to the same event) considered acceptable or not (knowing that they are not needed for the situation because the hazard does not lead to an accident or incident)?
- How is done the functional analysis?
- How is a heavy load drop dealing with?
- What is the criteria to be met for the demonstration of the acceptability of an independent internal hazard consequences. (e.g. if an internal hazard affects a tank with radioactivity substance, but does not lead to an accident linked with the reactor core, what are the radiological limits to be verified : those for the normal operation or those that correspond to an accident) ?

Failure of pressure retaining components

Question 8.10
Is the break preclusion applied for other circuits than Main Coolant Linbes (e.g. seldom operation of circuits under high energy,..) ?

Question 8.11
Completeness of considered effects of pipe failures :

- how the impact of whipping pipe on a neighboring equipment and constructions is considered (what equipment considered to be destroyed completely ?, partially damaged ?, will resist to impact ?,...) ?
- is the effect of local mass considered (indeed, if the whipping pipe has a valve or a diaphragm the kinetic energy will be increased), ?
- Postulated leaks and breaks in pipes for high and low-energy piping :What are the assumptions for
  - the size of leaks,
  - the locations of initial failure ?

- Are the longitudinal breaks assumed (what is their size) ?
- What is the size of a consequential leak (induced by the whip of initially broken pipe) ?

Question 8.12
How the qualification of the isolating organ to be able to close under accident condition is demonstrated (e.g. under condition of a ΔP corresponding to the break downstream of the organ) ?

Internal flooding

Question 8.13
How is assessed the risk of internal flooding (basic assumptions for analysis...)?

Question 8.14
How the released volume is calculated (sum of initial and induced leaks and breaks) ?

Question 8.15
What is the approach of leakage duration estimation (grace period for the operator) ? Is the time for isolation of all breaks and leaks (initial and induced ones) accounted ?

Question 8.16
Specific questions :

- Could be a flooding considered as a result of an erroneous alignment?
- Do you take into account the water arrival from neighboring buildingst?
- Are the Fire Protection System and tanks overfilling considered as a possible initiator?
- Are failures of detecting or isolating devices considered as single failure? What is their classification ?
- How the leak (or break) on the SIS outside the containment considered (under LOCA conditions or when the SIS is used in the RHR mode) ?

Fire

Question 8.17
Is fire considered during post-accidental conditions ?

Question 8.18
Are the fire detectors seismic classified ? what is the principle for their redundancy and diversity ?

Question 8.19
We understood that the fire suppression system for Reactor Building and Nuclear Annex is classified to earthquake ; is this system separated from the non-seismic part of the plant fire suppression system (and how) ? What is the water capacity of the seismic classified pipe (volume, duration of spray,..) ?

Question 8.20
Are there some design requirements (e.g. stability in case of seismic vibration) applied to the equipment failure of which can trigger a fire (e.g. oil containing, electrical cabinets,...) ?

Internal explosions

Question 8.21
How this hazard is dealt with ?

- What are the facilities to prevent an internal explosion?
- How an explosion in the non-classified buildings (e.g. turbine hall) is assumed ?

Heavy loads drop

Question 8.22
What are the requirements for the handling engines ?

Question 8.23
What are precautions for the fuel handling process ?

Question 8.24
What are the assumptions regarding a simultaneous (or single) drop of non-classified object(s) with risk to damage more than one safety redundant train ?

Seismic design

Question 8.25
Is it foreseen for the qualification of the equipment important for safety (e.g. valves and sensors in the containment or the Injection pumps,...) to take into account a SSE + LOCA conditions ?

Question 8.26
How to deal with the multiple long-term loss of many normal operating equipment which are non-classified for SSE (electrical grid, turbine hall,...) ? What is the assumption related to the restoring time ?

Other external hazards

Question 8.27
Is it foreseen to apply an « event » approach for certain hazards (freeze,..) ?

1.9 Containment bypass

The questions listed below concern the containment bypasses occurring in normal operation of the reactor for leaks and breaks in systems connected to the Reactor Coolant System.

1-Rupture risk of the Safety Injection System (SIS) following a leakage at the seat of a component in the SIS lines

1. Leak prevention

Question 9.1
What are the design conditions (pressure and temperature) for all piping parts of the Safety Injection System, Containment Spray System and Shutdown Cooling System) particularly sections outside containment?

2. Leak detection

In the SIS, provisions are made for the detection, containment, and isolation of the maximum expected leakage from a moderate energy pipe rupture.

Question 9.2
What are the leak rate values that lead to exceed design parameters?

Question 9.3
Could you state precisely what kind of detection instrumentation (minimum flow meter, pressure sensor outside containment, level sensor in SIS tanks, alarms...) are used to detect piping leak or check valve leak and the maximum expected leakage taken into account?

Question 9.4
What is the detection threshold of the leak flow rate metering?

Question 9.5
Could you be more specific about the qualification of the related measurements (sump level, minimum flow rate detector,...)?

Question 9.6
Is the evaporation of the coolant fluid in case of reactor coolant leak taken into account in the SIS design (risk of the piping break in two-phase flow, safety valves performance with two-phases flow)?

Question 9.7
What are the available time allowed to isolate the leak according to the leak rate?

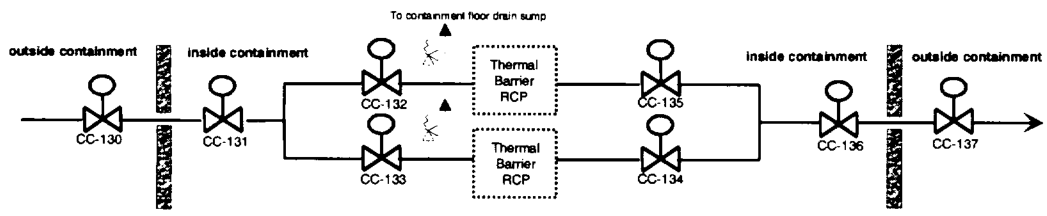
Cold leg and hot leg injection lines of the SIS :

Question 9.8
How is detected a leakage outside the containment from the three non-return valves in the direct vessel injection line? From which limit is the inverse flow rate identified?

Question 9.9
Does verification exist of the isolating valves capacity to close in case of unusual conditions (two-phase fluid, high temperature, inverse flow rate,...)?

2-Rupture risk of the Component Cooling Water System (CCWS) outside containment due to thermal barrier RCP rupture

CCWS sections that cooling the reactor coolant pump seal :



Question 9.10
How is designed the piping section between valves CC-132-135 and CC-133-134? To the reactor coolant pressure and temperature? Which criteria is used to close those valves (pressure, flow rate...)? How are designed the flow meter in those sections?

Question 9.11
The containment isolation valves (CC-130, 131, 136, 137) are automatically closed on low-low CCW surge tank. In case of overpressure and non-functionality of the previous relief valves (guillotine break), are the containment valves closed in the Control Room to assure isolation?

Question 9.12
How do you assure that the leak flow meters keep their functionality when the primary coolant has two phases?

The questions listed below concern the containment bypasses occurring in accidental conditions of the reactor (in case of safety injection actuation).

IRWST/SIS/CSS non-isolable containment extension

Question 9.13
What are the design conditions (pressure and temperature) of the In-containment Water Storage Tank and the non-isolate section between the IRWST and the first isolation valve included in the SIS/CSS suction line?

Question 9.14
If this section is not designed for reactor coolant parameters, how is this piping section prevented against leak risk when the SIS is required?

1.10 Low-temperature overpressure protection

Question 10.1
What is the reactor vessel chemical composition (percent of Ni, Cu...)?

Question 10.2
How is assured the reactor vessel integrity in low-temperature overpressure modes when the Shutdown

Cooling System lines are isolated spuriously?

Question 10.3

How is calculated the RT_{NDT} at the end of reactor life?

Question 10.4

The choice has been made to install relief valves for LTOP in shutdown cooling system suction lines; could you justify this solution instead of protection based on pressurizer relief valves?

Question 10.5

How will be taken into account the risk of isolation of the SCS lines during outage (spurious one due to operator failure or another) ?

2 AP600

The Institute for Nuclear Safety and Protection (IPSN) constitutes the technical support of the French Nuclear Regulatory Body (DSIN). Within the Safety Evaluation Department of IPSN, the Risk and System Evaluation Branch (SERS) has initiated a comparative analysis between design solutions proposed for several advanced nuclear power plant projects.

In addition to the AP-600 reactor, the concepts selected include the advanced System 80+ developed by ABB-Combustion Engineering, the common French-German project EPR, the Russian advanced reactors VVER-1000/V-392 and VVER-640/V-407.

The analysis focuses on the assumptions taken, the criteria retained and the requirements applied in relation to several safety key issues.

One of the objective of the study is to identify relevant advantages of selected concepts with respect to safety, operating simplicity and reliability.

At this time, with respect to the AP-600, our assessment is based on the review of the Standard Safety Analysis Report (SSAR) as well as the Final Safety Evaluation Report (FSER) you already provided us.

The review of this material has raised some preliminary remarks and comments. For each selected key issue, the following paragraphs present our questions in order to clarify technical items.

At your convenience, this questionnaire might be used as a guide for a technical exchange to be developed at a future technical meeting.

2.1 General safety approach and associated objectives

Question 1.1
Could you explicit the various probabilistic objectives associated with a controlled core melt :

- reactor states considered (including shutdown states),
- external events taken into account,
- low pressure / high pressure core damage states,
- core melt with containment bypass ?

Question 1.2
What is the approach followed to set up the radiological releases criteria for fault situations ? What are the results obtained ?

Question 1.3
Could you provide some pieces of information regarding the means you intend to implement in order to achieve the objective of total level radioactive waste volume less than 1750 ft^3/year ?

2.2 Classification of equipment and components

Question 2

Could you be more explicit about the content of requirements applied to the various safety classes, especially with respect to :

* implementation of the single failure criterion,
* application of redundancy and diversity on the functions and systems levels,
* physical separation of redundant trains,
* emergency power supply by the emergency diesel generators,
* periodic testing ?

2.3 Treatment of the guillotine primary break

It seems that you have taken various assumptions with respect to the postulation of large primary breaks. To complete our understanding of this approach, some complementary information is needed, as well as additional explanation concerning the justification of the different break sizes postulated.

Question 3.1

How is defined the boundary break for the reactor internals ?

Question 3.2

What is the level of conservatism for the assumptions taken for the containment and the Safety Injection System design, as well as for the equipment qualification and for the radiological consequences ?

Question 3.3

What is the classification of the equipment needed for the detection ? How the "low probability" of the break is demonstrated using the "deterministic" mechanical fracture approach ?

Question 3.4

The stratification phenomena and frequency of temperature cycles (fatigue) of the surge line are difficult to describe; how its break can be excluded ?

2.4 Containment design

Question 4.1

Some complementary information is needed to understand the containment air cooling system as the air flowrate, the efficiency of the system, the modes of operation and the use of this system in accidental and radiological studies.

2.5 Main safety systems design required for heat removal

Normal Residual Heat removal system (RNS)

Question 5.1

Could you transmit us your shutdown risk evaluation related to reduced level or loss of normal heat removal system at mid-loop operation ? Could you be more specific about design provisions, detection and mitigation means for water make-up and decay heat removal functions?

Question 5.2
We have noticed that mid-loop operation cannot be excluded on AP-600 as operating state for steam generator draining and maintenance activities. It seems on the contrary that reduced inventory operations are avoided for nitrogen sweeping. Could you explain to us how and when do you perform this function ? Besides, could you be more specific about the design measures taken to avoid air-drive into the RNS pump suction during mid-loop operations ?

Passive Core Cooling System (PXS)

Question 5.3
The passive residual heat exchanger and the Core Melt Tanks constitute a significant extension of the reactor coolant system. These kind of elements will have a difficult construction and need regular in-service inspections of tubes which will be difficult to attain, to inspect and to repair. How do you foreseen to lessen the impact of these problems?

Question 5.4
How do you intend to modify or stop an accidental event when passive systems are in operation? How do you ensure that all events are covered?

Component Cooling Water System (CCS)

Question 5.5
In case of leakage into the CCS from a high pressure source, which is the initiator (number of tube ruptures, flowrate...) used to design the alarm flow measurement in the cooling water outlet line? What is the qualification of the isolation valves?

Question 5.6
In case of loss of CCS, how is cooled the Spent Fuel Pool Cooling System heat exchangers?

Question 5.7
Some complementary information is necessary about the qualification of the isolation valves and the design flow rate.

Question 5.8
What are the set pressure and the design flowrate of the relief valve downstream the reactor coolant pump? Will these valves keep their functionality in case of two phase flow?

Question 5.9
In case of leakage into the CCS, what is the pressure increase into the CCS pipes?

Passive Containment Cooling System (PCS)

Question 5.10
What are the transfer heat modes through the steel containment vessel during an accident to remove heat ? What is the dominant mode ?

2.6 Single Passive failure criteria

Question 6

Some detailed information is missing at this time, regarding mainly :

- the complete list of systems designed according to the passive single failure,
- the justification of the exclusion from the passive single failure criterion for the line blockage,
- the justification of the non-consideration of short term passive failures,
- the determination of a technically appropriate value for the passive failure flowrate,
- the preventive and mitigative measures implemented to cope with the postulated passive failure applied on each safety system, including provisions taken for detection and isolation as well as for water exhaust.

2.7 Electrical power supplies, distribution and response to station blackout

After this first review of the documentation available, several issues still need to be clarified. It concerns mainly :

Question 7.1

the definition of the SBO coping time,

Question 7.2

the justification of the batteries capacity for a 72 hours duration,

Question 7.3

the detail of means used and measures taken to ensure the reactor coolant primary integrity and the reactor coolant system tightness (especially in the primary pumps area) under black-out conditions,

Question 7.4

the operation of safety systems without electric power until 72 hours after the event,

Question 7.5

the management of station blackout under shutdown conditions,

Question 7.6

the PSA results and some important study assumptions :

- period of calculation,
- maximal recovery time for the external electric power,
- behaviour of the primary safety valves,
- delay available before core melt.

2.8 Seismic design and internal hazards

Seismic design

Question 8.1
How do you take into account the combination of a Safe Shutdown Earthquake and an accident condition?

Fire protection system

Could you provide us your design provisions for the protection against :

Question 8.2
- a fire induced by explosions or aircraft crash;

Question 8.3
- explosions induced by fire;

Question 8.4
- fire which induces a transient or accident;

Question 8.5
- fire which leads to the BDB conditions (spurious opening of several safety valves, or steam dump to atmosphere,...);

Question 8.6
- consideration of Single Failure Criteria in the Fuel Handling Area;

Question 8.7
- the « 3-hours resistance » ; what is the time-temperature curve ? what is the verifying criteria (keep of integrity or temperature outside the barrier or other) ?

2.9 Containment bypass

Question 9.1
What are design means to ensure prevention of the containment bypass, detection and isolation of the leak as well as mitigation of the consequences for each kind of containment bypass?

Question 9.2
Could you provide us with main findings coming from your probabilistic assessment of containment bypass?

Question 9.3
How is taken into account the risk of rupture in the Component Cooling Water System (CCS)?

Question 9.4
How is taken into account the risk of rupture of the thermal barrier of the reactor coolant pump can lead to the rupture of the low pressure Component Cooling System (CCS) outside the containment?

Question 9.5
How is taken into account the leakage through a heat exchanger between the CCS and a system connected to the reactor coolant system?

2.10 Low-temperature overpressure protection

Question 10.1
What is the reactor vessel chemical composition (percent of Ni, Cu...)?

Question 10.2
How is assured the reactor vessel integrity in low-temperature overpressure modes when the Normal Residual Heat Removal System (RNS) lines are isolated spuriously?

Question 10.3
How is calculated the RT_{NDT} at the end of reactor life?

Question 10.4
The choice has been made to install relief valves for LTOP in RNS suction lines; could you justify the choice of this technical solution instead the use of pressurizer relief valves?

Question 10.5
How will be taken into account the risk of isolation of the SCS lines during outage (spurious one due to operator failure or another) ?