AMENDMENT 1 TO RESAR-SP/90 PDA MODULE 3,

"INTRODUCTION AND SITE"

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INSTRUCTION SHEET

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440.1 It is not clear what philosophy was followed in determining the scope of the Nuclear Power Block (NPB) to be provided by Westinghouse and the balance of plant to be provided by the A/E. For example, Section 1.1.1.1 contains the following in the first paragraph:

"The scope of the WAPWR design includes all ... systems and components that are essential to the safe and proper operation of the nuclear power plant. Specifically excluded are ... the service water/cooling water structure ..."

This is consistent with Table 1.1-1, which shows the service water structure and dams to be outside the scope of the NPB and indicates that W will provide design criteria for the Service Water System (SWS). We do not understand the reasoning for not providing the SWS as an integral part of the NPB in light of its importance to safety.

Another example may be taken from Table 1.1-1, where the startup feedwater system is stated not to be safety related. Nevertheless, it has important safety implications, and can serve as a valuable backup to safety related systems as well as reducing demand placed upon safety related systems. We do not understand why this is not provided as part of the NPB in light of the safety implications and close coupling with NPB response.

Other examples where W has elected to provide items that are traditionally in the A/E scope may readily be provided, such as election to provide the emergency diesels as part of the NPB rather than relying upon the A/E and inclusion of the Component Cooling Water (CCW) System in the NPB.

We believe that the more systems that are provided within the NPB, the more standard will be the resulting plant.

In light of the above, please address the rationale for selection of what will and what will not be provided as part of the NPB. We would particularly like to be provided with information pertinent to safety related systems and systems that are important to safety, including the above examples.

RESPONSE

The WAPWR Nuclear Power Block (NPB) scope and configuration was established in accordance with the following criteria:

- Traditional NSSS scope of supply

  plus

- Other structures, systems, and components

  - o important to safe and proper operation
  - o important to mitigation of accidents
  - o important to power capability and load follow capability
  - o which minimize interfaces important to safety
  - o sufficient to a self standing one-step license application
  - o which otherwise makes economic sense

  plus

- Complete management control of all plant safety features

With respect to the service water system (SWS), Westinghouse recognizes its importance to safety. However, the SWS design may be significantly impacted by individual site considerations such that providing a bounding design would be impracticable. Westinghouse will provide detailed design and interface criteria for the SWS as part of RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

With respect to the startup feedwater system (SFWS), Westinghouse recognizes its importance to improved reliability of the emergency feedwater system (EFWS), although the SFWS is a control grade system. Therefore, detailed design criteria, which essentially stipulates the design of the system, are provided in Appendix 10A, Subsection 10A.4.9 of RESAR-SP/90 PDA Module 6/8, "Secondary Side Safeguards System/Steam and Power Conversion System".

440.2 Section 1.1.1.2 contains the following:

"The following power levels are assumed in the accident analyses for the WAPWR design:

- A core thermal power level of 3800 megawatts is assumed in the accident analyses for events which are departure from nucleate boiling (DNB) limited and initiated at full power.

- A core thermal power level of 3876 megawatts, which is 1.02 times the core thermal power level, is assumed in the accident analyses for events which are not DNB limited and initiated at full power.

These assumed power levels are in accordance with the recommendations of Regulatory Guide 1.49, Revision 1, 'Power Levels of Nuclear Power Plants,' for the licensed core power level of 3800 megawatts."

The referenced Regulatory Guide (RG) 1.49 does not mention DNB, and the regulatory position is stated as follows:

"1. The proposed licensed power level of all nuclear plants for which a construction permit application is filed pursuant to Section 50.34 of 10 CFR Part 50 should be limited to a reactor core power level of 3800 megawatts thermal or less until January 1, 1979, at the earliest.

2. Analyses and evaluation in support of the application should be made at an assumed core power level equal to 1.02 times the proposed licensed power level (with a maximum acceptable value of 1.02 times 3800, or 3876 megawatts thermal) for (a) normal operating conditions, (b) transient conditions anticipated during the life of the facility such as load changes, control rod malfunctions and loss of forced coolant flow, loss of load or turbine trip, loss of normal ac power, primary system depressurization, etc., and (c) accident conditions necessary to evaluate the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

3. Analyses of the possible offsite radiological consequences of postulated design basis accidents made to demonstrate acceptability of the site in accordance with 10 CFR Part 100 should be performed for an assumed core power level equal to 1.02 times the proposed licensed power level or may, at an applicant's discretion, be made at a high power level, not to exceed 4100 megawatts thermal.

Analyses made at an assumed power level greater than 1.02 times the proposed licensed power level should be regarded as supporting operation of the facility at a proposed licensed core power level no greater than 3800 megawatts thermal."

Explain how the power level is in accordance with the RG 1.49 position.

We also point out that Chapter 15 of NUREG-0800 also calls for transient and accident evaluations to be performed at a power level of 102%. Thus, Westinghouse does not appear to meet the SRP criteria.

RESPONSE

The difference in initial power levels assumed is due to the difference in methodology used for these transients.

For accidents which are DNB limited nominal values of initial conditions such as pressure, temperature, and power (3800 MWt) are assumed. The allowances on pressure, temperature, and power (2%) are determined on a statistical basis and are included in the limit DNBR as described in WCAP-8567. This procedure is known as the Improved Thermal Design Procedure (ITDP).

For accidents which are not DNB limited or when ITDP is not employed the initial conditions are obtained by adding the maximum steady state errors directly to the nominal values.

The error allowances are therefore considered on core power either in the accident analyses initial conditions or factored into the limit DNBR.

440.3 The following statement is provided in Section 1.2.3.4:

"The accumulators, core reflood tanks and heat exchangers are located inside the containment building. It is proposed that the four pumping modules be housed in Containment Pressure Pump Enclosures (CPPE's) in order to encompass all piping and components associated with any post accident recirculation of highly radioactive fluid within a containment boundary. This total containment encapsulation concept for the ISS eliminates the potential for post accident releases of highly radioactive liquid or gases into the auxiliary building and subsequently into the environment."

The wording "It is proposed that ..." leads the staff to believe a final decision has not been reached. This belief is emphasized by the figures provided with Module 3, which do not reflect the "proposed" part of the statement, although, with a slight modification, this would appear possible. The last sentence is inconsistent with Section 1.2.3.3.1.3, Annulus Air Cleanup System, which contains the statement:

"This system also collects and processes potential airborne contamination resulting from leakage in the ISS recirculation paths outside containment, ..."

the implication being that the CPPE is not incorporated into the W SP/90 design.

The proposed CPPE's appear to be a contribution to safety and the control of radioactive material under a number of conditions, including design basis events, other accidents including severe accidents, and post-accident control of conditions which may exist following an accident.

The staff cannot make decisions or accept a design based upon proposed concepts. If this is to be a design option, it is to be clearly identified as such, and sufficient information is to be provided for a staff evaluation of each option. In light of the above, W should clarify their position on the CPPE concept, provide consistency within the Module and with other Modules, and provide sufficient information for each design option that the staff can conduct the review.

RESPONSE

The current W position, with regard to the Containment Pressure Pump Enclosures (CPPE's), is that part but not all of the CPPE design goals will be incorporated into the W RESAR-SP/90 design. The four ISS pumping modules will be housed in individual filtered-vented pump compartments in order to encompass all piping and components associated with any post-accident recirculation of highly radioactive fluid within four dedicated, separated safeguard component areas.

Each safeguard component area is designed to minimize the potential for post-accident releases of highly radioactive liquid or gases into the auxiliary building and subsequently into the environment. All penetration and personnel/maintenance accesses from the auxiliary building to these compartments are to be water tight and are located at the 2nd floor level of each compartment to prevent liquid spills from flowing into or out of any safeguard component area. Radioactive steam releases into these compartments are individually ducted to a redundant annulus air cleaning system before being vented to the environment. Individual room coolers are also provided in each of the four safeguard component areas.

The two CPPE features that are not incorporated into the safeguard component area design are the containment design pressure requirement and the capability for the compartments to vent back to the containment via pressure relief devices in the event that a breach of the RHR pressure boundary occurred during any RHR operation. The current W position is based on an unfavorable cost/benefit analysis. Unless the perceived benefits can be quantified sufficient to justify the additional capital cost, these specific CPPE design features cannot be incorporated into the W RESAR-SP/90 design at this time. However, should future evaluations produce a more favorable cost/benefit ratio, these CPPE features may be reconsidered.

440.4 Section 1.2.3.4 contains:

"The four pumping modules are totally independent and identical to each other. The ISS concept recommends that two of the modules be located on the opposite side of the containment 180 degrees apart from the other two modules. This arrangement is shown in Figure 1.2 (Sheet 1)."

Use of "The ISS concept recommends ..." is not definitive. In this case, the independence is consistent with the figures contained in the module with respect to the physical location. However, we do not understand how true independence exists for a two diesel NPB. Please explain. In addition, please address common cause failures since all systems are identical.

RESPONSE

The referenced paragraph from Subsection 1.2.3.4 is specifically refering to four pumping modules as mechanical subsystems and make no claim as to the electrical independence. Please refer to the second paragraph of Subsection 1.2.3.4, which clearly states that; "The ISS consists of four identical and totally separated mechanical subsystems which (can be) powered from either two or four separate and redundant emergency electrical power trains..."

440.5 Section 1.2.3.4 addresses the four electrical train concept. Please explain the four electrical train statement in that none of the figures show this concept, nor is it immediately evident to us how the equipment could be fitted into the space available. Is this a serious consideration in the design process? When is a decision to be made? What information may we expect and when in regard to the four train concept?

RESPONSE

The current W position is to incorporate the two electrical train concept for the W RESAR-SP/90 design. This position is based on an initial unfavorable cost/benefit analysis for the four electrical train concept.

However, subsequent, independent, two versus four train cost evaluations have resulted in a more optimized plant arrangement for the four electrical train concept and a significant reduction in the associated cost penalty. Also a Westinghouse sponsored utility review team has indicated a unanimous preference for the adoption of a four train electrical design, provided this will result in relaxed Technical Specifications for the diesel-generator units.

It should be emphasized that the current W position is that the RESAR-SP/90 design should be licensable with a two train design. However, additional evaluations are planned, and as a result, this decision may be reconsidered in the future.

440.6 In Section 1.2.3.5, the statement is made that "The SFWS also serves to minimize the number of EFWS actuations required which enhances the reliability of the EFWS." We understand that the number of demands placed upon the EFWS may be diminished, but do not understand the stated impact on EFWS reliability. Please clarify the statement in light of our difficulty.

RESPONSE

There are two points that should be made in connection with the reliability impacts of the SFWS. The first is that in the implementation of automatically starting of the SFWS additional start signals were added to the EFWS. These start signals improve the reliability of the EFWS because actuation reliability was a limiting factor of the overall system's reliability.

The second point is that automatic start of the SFWS improves the reliability of the combination of SFWS and EFWS. This is not an improvement in the reliability of the EFWS, per se, but rates and improvement relative to the traditional auxiliary feedwater system function.

440.7 Please discuss the reasoning which led to a decision not to use the passive heat removal system which was contained in earlier W design concepts.

RESPONSE

There are several reasons why the passive steam condenser system (PSCS) was dropped. One reason is cost, both capital and developmental. Out detailed evaluations have shown that the PSCS costs more than an EFWS and in addition it would require extensive efforts to design and test the condenser. A second point is that our preliminary PRA work indicates that although the PSCS is more reliable than the EFWS it does not result in reduced core melt frequency because other events are dominating.

Also, the PSCS by itself does not significantly improve steam generator (SG) tube rupture mitigation (in particular overflow). Instead Westinghouse has incorporated a special SG overflow system (see RESAR-SP/90 PDA Modules 6 and 8, "Secondary Side Safeguards System/Steam and Power Conversion System") which is less costly and more effective than the PSCS. Another factor is the PSCS requires more high energy lines and requires them to be in areas of the plant that would not otherwise have them; i.e., the upper level of the REB, which contains HVAC equipment.

440.8 Section 1.2.3.5 states "The pumps are sized such that any two of the four pumps delivering to any two of the four steam generators provides the minimum emergency feedwater flow." What are the criteria applicable to sizing the pumps? What would be typical plant response if only one pump were available?

RESPONSE

The sizing of the EFW pumps is based on a feeds line break (condition IV event). For this event one EFW pump is assumed to spill and at least one of the two cross over isolation values is assumed to close. This leaves 3 pumps which are connected to 3 intact steam generators (SG). The worst single failure would result in one of the 3 pumps failing and the other two pumps delivering to 2 SG. This is demonstrated in the EFWS FMEA in RESAR-SP/90 PDA Modules 6 and 8, "Secondary Side Safeguards System/Steam and Power Conversion System." Note that in this condition the RCS remains subcooled even though this is a condition IV event.

W has determined by analysis that with one pump feeding one SG the RCS will remain subcooled under best estimate conditions. If SAR conditions are assumed in combination with this double failure there would be some RCS voiding however the core would not uncover and there would be no fuel damage.

440.9 The Figure 1.2-2 Elevation 72.0 identifies various sumps. Are the sumps to be interconnected in such a way that flooding in one cavity cannot flood other cavities via a path through the sump drain lines? What isolation is provided?

RESPONSE

The sumps shown in each compartment at elevation 72.0 meters (Fig. 1.2-2 Sheet 1 of 9) are individual floor drain sumps with individual sump pumps. These sumps are not interconnected therefore there will be no path for flooding in one compartment to flood other compartments.

440.10 What is the "EL 104.9M" notation in Figure 1.2-2, El 72.0?

RESPONSE

The floor of the ICIS tunnel and the reactor vessel cavity is EL 72.5 meters. The 104.9 meter notation is in error.

440.11 Is a sump provided at the bottom of the vessel cavity? If so, what is its configuration? If not, please explain how leakage or spillage is to be handled.

RESPONSE

Yes, a sump and sump pump would be located in the bottom of the reactor vessel cavity at el 72.5 meters.

440.12 Are any cooling provisions made for the air in compartments at this level? Please discuss.

RESPONSE

Yes, cooling provisions are made for the reactor vessel cavity. The cooling requirements for all the containment compartment (i.e. steam generator compartments, pressurizer compartments, reactor vessel cavity, RHR heat exchanger compartments, regenerative heat exchanger compartment and letdown heat exchanger compartments) are satisfied by redistribution fans which are not depicted in Figure 1.2-2. The general in-containment HVAC design is for the four containment
(a,c) recirculation fans, located on elevation [ ] meter, to draw containment air through the four containment recirculation cooling units, located on elevation 100 meters, and deliver the cooled air to the floor of the containment at elevation 84.4 meters. Redistribution fans are then used to force the required cooled air to the designated compartments.

440.13 The lower portion of the elevations in Figure 1.2-2 show a number of electrically driven pumps. Please discuss alternates to electrically driven SI pumps including the reasons these are not used as a diverse means of backup to the planned ISS or as part of the ISS.

RESPONSE

All of the integrated safeguards system (ISS) pumps use AC motor drivers because Westinghouse feels that this provides the most reliable/practical arrangement. Other solutions such as steam turbines or direct diesel drives would be less reliable and also would introduce design problems; for example, a steam turbine could not use steam generator (SG) steam because for LOCA; especially large LOCA's, the SGs do not produce much if any steam.

In addition our preliminary PRA work indicates that use of AC motor drives in the ISS will allow a very low core melt frequency even with common mode failures taken into account.

440.14 With respect to El 77.4 of Figure 1.2-2, can an accident in the PD charging pump or CVCS valve areas impact both these areas and the ones below, causing loss of all charging capability? Please discuss.

RESPONSE

The CVCS backup seal injection pump has been separated from the normal centrifugal charging pumps so that anticipated type leaks or other failures would not affect both. However, these components are not required to mitigate any design basis events; as such they are not required to be protected from hypothetical events such as severe fires, the complete severence of a high energy line, etc. In that case there is redundant safety grade equipment which is located in other areas of the plant which is protected from fires, HELB, floods, etc. that could occur in the CVCS area. For example the HHSI pumps are capable of providing RCS makeup and boration and the CCWS is capable of cooling the RCP seals.

440.15 In Figure 1.2-2, El 77.4, what is the seal capability in the openings to the ISS valve areas (from the corridors leading to the SR sluice pumps and to the Chem Dr Tk)?

RESPONSE

As stated in the response to Question 440.3, all penetrations and personnel/maintenance accesses from the auxiliary building to each of the safeguard component areas are to be water tight. The corridors adjacent to these compartments are main equipment access corridors and will be more clearly depicted as such on the next revision of these drawings

All compartment, such as the SR sluice pump compartments in the west wing of the RE/B and the chemical drain tank compartment in the east wing of the RE/B, will be provided with curbs and individual drains where necessary to prevent spill out to the main corridor.

440.16 What cooling is provided for the areas at the El 77.4 level of Figure 1.2-2?

RESPONSE

Module 13 will provide the descriptions for the RESAR-SP/90 HVAC systems. At elevation 77.4 the plant layout drawing reflects only a few specific HVAC components, such as the four individual safeguard component area cooling units. The Reactor External Building (RE/B) supply cooling unit and fans and a RE/B exhaust unit and fans are located at elevation 107.6 meters (Sheet 7 of 9, Figure 1.2-2). These units provide the cooling and ventilation requirements for the RE/B in general.

440.17 With respect to Figure 1.2-2, El 77.4, please discuss the reasoning which led to use of a DC motor driven pump, as contrasted to a turbine driven pump, as a means of providing diverse methods for RCP seal injection.

RESPONSE

The direct current motor drive arrangement for the WAPWR backup seal injection pump was selected on the following bases:

Reliability - The operation of this pump has an impact on the overall plant core melt frequency, considering non-design basis events such as a loss of all AC power or a loss of all auxiliary cooling. Also, this pump can improve plant availability by preventing the reactor coolant pump seals from being fouled or damaged for many events which do not impact plant safety, such as temporary loss of normal CVCS seal injection due to equipment failure or operator error. Therefore, it was desirable to use a highly reliable pump driver for the backup seal injection pump. A steam turbine driven pump was considered for this reason. However, historically, the reliability of such pumps is not exceptionally high; an engine / DC generator / DC motor / pump combination will have comparable reliability.

Divers:ty - In order to effectively address the loss of all AC power and loss of all auxiliary cooling scenarios mentioned above, the backup seal injection pump must be independent of the emergency diesel generators and of component cooling water, service water, and HVAC. A small, commercial type internal combustion engineer / DC generator / DC motor / pump arrangement meets these objectives. The engine / generator is considered to significantly diverse from the emergency diesels due to the great difference in size (75 HP vs. several thousand horsepower), manufacturing (commericial manufacturer vs. specialized equipment manufacturers), and cooling system (air cooled vs. water cooled). Also, the fuel for the engine could be selected to be different from the emergency diesels; a different grade of diesel fuel, or propane, or gasoline.

Cost - Since the backup seal injection system is not required for the mitigation of design basis accidents, and is assumed to be unavailable in the analysis of those accidents, it is not required to be a safety grade system, and minimization of capital cost is desirable. The commercial type equipment currently specified for the backup seal injection system is very inexpensive. In contrast, a steam turbine driven pump would require special equipment development, at high cost. Also, use of such a pump would require high energy steam line to be routed into the non-safeguards area of the auxiliary building; such practice would in*roduce high energy line break and steam line integrity concerns, as well as having a very high cost.

Simplicity - The use of commercially available equipment such as a DC generator and DC motor insures that no specialized maintenance procedures or spare parts will be required, and that the design can be easily understood by plant maintenance and operating personnel. Finally, the pump required for this application is very high head / very low flow; such turbine driven pumps are not readily available and would require special development, which would complicate maintenance and increase the capital cost.

440.18 Please discuss communication within containment to the EWST (applicable to all levels within containment).

RESPONSE

The communication within the containment to the EWST is shown at elevation 84.8 meters (Sheet 4 of 9, Figure 1.2-2) as six individual EWST spillways. These spillways are located outside the loop compartments and each spillway contains two 20 inch diameter vertical pipes, which extend through the concrete slab that serves as the floor of the containment as well as the roof of the EWST.

Two of the six spillways are shown immediately outside the two RHR HX compartments located between column lines H and N. The other four spillways are shown on either side of column line H between the stairs and the loop compartments. There are a total of twelve 20 inch diameter pipes that allow water to return to the EWST from the containment floor.

Each spillway provides a trash rack and rough screen that envelopes the two corresponding 20 inch diameter pipes. All twelve of the 20 inch diameter pipes extend into the EWST to a position several inches below the nominal EWST water level and they extend approximately 12 inches above the floor of the containment. In the event of a LOCA, the spilling coolant water would first flood into the loop compartments. After a predescribed water volume had accumulated on the loop compartment floor, the water elevation would exceed a curb height and the coolant would flood into the reactor vessel cavity and in-core instrument tunnel (ICIS). Only after all dead volume, below the EWST spillway elevation, were completely flooded would the coolant start to return to the EWST via the twelve 20 inch diameter pipes.

440.19 In El 84.8 of Figure 1.2-2, can a major leak (break) in the CCW HX "A" area result in flow into the "B" area? Can leaks (breaks) result in flow downward into the CCW pump areas below?

RESPONSE

The two CCW heat exchanger compartments will be designed to prevent a leak in one compartment from flooding into the adjacent compartment or into the CCW pump compartment at elevation 77.4 meters by utilizing curbs or water tight doors if required. Since the CCW heat exchanger compartments are located above grade, at elevation 84.8 meters, a device can be employed that would permit excess flooding in a CCW HX compartment to spill back into the service water pipe tunnel and return to the ultimate heat sink.

440.20 In El 92.2 of Figure 1.2-2, the location of the switchgear rooms and the diesels above implies the SI pumps located directly below (El 72.0) are powered by the diesels above them (i.e., SI pumps to the left in the El 72.0 drawing obtain power from the diesel generator B; and pumps to the right from diesel generator A). Is this a correct assumption?

RESPONSE

Yes, the ISS pumps located to the left of column line 7 would be connected to the safeguard bus in switchgear room "B", which would in turn be connected to the diesel generator "B". Likewise, the ISS pumps located to the right of column line 7 would be connected to safeguard bus "A" and therefore to diesel generator "A".

440.21 Please discuss the approach to avoiding cold overpressurization of the RCS. Include a comparison to existing plants and show how past accident experience has been integrated into the W SP/90 design.

RESPONSE

Subsection 5.2.2.10 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System", provides a detailed discussion of the Westinghouse approach to avoiding cold overpressurization of the RCS. The administrative controls, operating procedures and use of the pressurizer power operated relief valves planned for the WAPWR are essentially the same as those employed in a large number of existing Westinghouse PWR plants.

In addition, each of the four RHR pump suction lines from the RCS hot legs contains a self-actuated liquid relief valve to protect the RHR subsystem against overpressurization. Each of these valves can accommodate the flow resulting from operation of two centrifugal charging pumps, with the normal letdown path isolated. This capacity is consistent with credible mass input events. The availability of more than one of these valves provides a capability for limiting RCS pressure to the relief valves' setpoint that is well beyond that required by credible overpressurization scenarios.

As the design evolves and credible overpressurization mechanisms become better defined, more emphasis may be placed on these liquid relief valves as the primary means of cold overpressure mitigation in the pressure range where they are available, i.e. with the RHR pump suction line isolation valves open. Further evaluation and analysis at higher pressures (with the isolation valves closed) may show that adequate protection against cold overpressurization can be provided by alternate means, such as a pressurizer steam cushion. Should this be the case, use of the pressurizer PORVs to protect against cold overpressurization may be unnecessary.

The results of such further evaluations will be provided in the FDA version of RESAR-SP/90.

440.22 What steps have been taken to avoid LOCAs under shutdown conditions? Please contrast the SP/90 design features to existing plants and plant accident experience.

RESPONSE

Since this question specifically addresses LOCA's under shutdown conditions, this response will assume that the principal issue is a LOCA in a RHR recirculation loop after the RCS has been depressurized and refueling operations are in progress. During this mode of operation the residual heat load would have decreased sufficiently such that eventually only one of the four ISS RHR subsystems would be required to provide sufficient recirculation flow to remove the residual heat.

Due to the system configuration and the layout arrangement, the probability for a LOCA to occur during this operating mode is considered negligible. There are three highly improbable scenarios that could be postulated to occur during a shutdown phase that could result in a significant spill from one of the four RHR subsystems. These three events are:

(1) Administrative Errors -- An out-for maintenance RHR subsystem could conceivably be aligned to the RCS during refueling operation by an inadvertant opening of the two corresponding inner and outer RHR suction isolation valves. This error could result in a drain down of refueling water from the refueling canal to the corresponding ISS pump compartment, should a valve or pump be in the process of being disassembled at the time. Administrative procedures, such as a control board valve tagging, should preclude this type of accident; however, in the unlikely event that such an accident did occur, the four independent water-tight compartments

specified for the four WAPWR ISS subsystems would contain the spill and consequently limit the flooding to only one compartment. In addition, the 8-inch RHR letdown line, associated with each of the four RHR subsystems, would significantly reduce the flow rate into the corresponding pump compartment as compared to the potential flooding rates associated with the current plants. Operator action would be required to terminate this type of spill; however, sufficient time would be available for the operator to close the appropriate inner and outer RHR suction isolation valves before a significant inventory of water is lost from the refueling canal.

(2) Pipe Rupture -- Any size pipe break, from a 3/4-inch instrument line to a complete severance of the 8-inch RHR pump suction or discharge line, could be postulated for one of the operating RHR subsystems. However, pipe breaks are considered highly improbable during this operation phase since the system would not experience any unusual design transients and the system would be operating at a pressure significantly below the allowable design pressure. As discussed above, the WAPWR ISS configuration/layout arrangement does provide several features that would minimize the consequence of any LOCA outside containment in the unlikely event that one did occur.

The four independent water-tight components would contain the spill and limit the flooding to only one compartment. Since all outside-containment components, associated with each of the four ISS subsystems, are totally encapsulated in a dedicated compartment, the method for detecting and terminating a leak or major break is significantly improved over current plants. A dedicated sump in each compartment would collect any leakage associated with a given subsystem and redundant level instrumentation would provide an alarm in the control room. The operator would know immediately which subsystem was affected and could take immediate action to terminate the problem by isolating the system from the RCS.

(3) Valve Misalignment -- Two operator errors in conjunction with the failures of two specific interlocks would be required in order to effect a valve misalignment LOCA in an operating RHR subsystem. Interlocks are provided for essentially all motor operated valves associated with the RHR subsystems to ensure the proper valve positions prior to aligning the RHR subsystem to the RCS. There are two series motor operated valves or a motor operated valve and a series check valve in all interconnecting lines or branch lines associated with the RHR subsystems.

These interlocks require that all branch line motor operated valves be in their closed position prior to the opening of the corresponding inner and outer RHR isolation valves, which effectively aligns the RHR subsystem to the RCS. These interlocks also prevent any branch line motor operated valves from being reopened unless the corresponding inner and outer RHR isolation valves have been reclosed. In addition, power removal and restoration provisions are to be provided for several key valves, such as the two series motor operated system test valves and the two series motor operated spray header isolation valves. This feature is in addition to the interlocks and it not only ensures that power is removed from these valves before the corresponding RHR subsystem is aligned to the RCS, but it also ensures that power is not restored to these valves until the corresponding RHR inner and outer isolation valves are closed.

440.23 Leakage from the upper portion of the containment shell clearly is into the annulus. What is the leakage path from the lower portion of the containment shell, where the shell appears to be totally surrounded by concrete?

RESPONSE

Any post-LOCA leakage from either the upper or lower portions of the containment shell or from any containment penetration would be drawn into the annulus air cleanup system by two redundant centrifugal fans which would establish a negative pressure in all related compartments. The leakage would pass through redundant charcoal filtration units before being exhausted to the plant vent. The details of this HVAC system are discussed in RESAR-SP/90 Module 13, "Auxiliary Systems". However, a brief description of the containment leakage paths and their interface with the annulus air cleanup system is provided below.

The two potential containment leakage paths are the spherical steel containment plate seam welds and at any of the numerous mechanical or electrical containment penetrations. As shown in Section A-A (Sheet 8) and Section B-B (Sheet 9) of the WAPWR plant layout drawings, Figure 1.2-2, the spherical steel containment sits in a concrete
(a,c) cradle that extends up to elevation [ ] meters. Some of the containment plate seam welds and containment penetrations are located
(a,c) above elevation [ ] meters while others are located below elevation
(a,c) [ ] meters.

(a,c) Any potential containment leakage points located above elevation [ ] meters would obviously leak directly into the major annulus area. In general, the annulus area for the WAPWR is recognized as the area above the concrete cradle with an outside parameter defined by the containment shield building and the inside perimeter defined by the spherical steel containment. However, it should be noted that there
(a,c) are areas below elevation [ ] meters which are connected directly to the major annulus area, therefore, these areas are also considered part of the annulus volume.

The containment weld leak chase system, which is associated with that portion of the spherical containment in direct contact with the concrete cradle, is considered part of the annulus area. The containment weld leak chase system is a network of leak chase channels which provide a small annulus enclosure for all containment plate
(a,c) welds below elevation [ ] meters. This interconnected system of leak

chase channels is welded to the exterior surface of the spherical containment and it provides a means for collecting and directing any potential containment weld leakage back to the major annulus area. The volume of the containment weld leak chase system is considered an integral part of the overall annulus volume.

Approximately half of the piping penetration areas, located below elevation [ ] meters, are also considered an integral part of the overall annulus volume. As shown on plan elevation 84.8 meters, (Sheet 4 of Figure 1.2-2), there are eight separate piping penetration compartments. The four piping penetration compartments that straddle column line (7) are open directly to the major annulus area, therefore, they are considered an integral part of the overall annulus volume. The four piping penetrations compartments that straddle column line (H) are designed such that they can be completely isolated from the major annulus area if required. Each of these four penetration areas are designated as an integral part of a corresponding filtered/vented pump compartment which houses one of the four ISS mechanical subsystems. (a,c)

Of the four piping penetration compartments which are open directly to the annulus area, two of these piping penetration compartments are dedicated primarily to non-safety, radioactive piping. These two penetration compartments are located above the CVCS valve and charging pump compartments. The other two piping penetration compartments that are open to the annulus area are dedicated primarily to non-radioactive safety system related piping such as component cooling water and emergency feedwater system piping. These two penetration compartments are located above the emergency feedwater system pump and valve compartments. Any assumed leakage from the containment penetrations located in these four compartments should be considered as a leak directly into the annulus volume.

It should be noted that Sheet 4 of Figure 1.2-2 (plan elevation 84.8 meters) does not correctly reflect the personnel accesses into the four piping penetration areas discussed above. Each of these four piping penetration compartments would be provided with one personnel access and each personnel access would be designed with double air-tight doors which would permit the annulus air cleanup system to establish a slight negative pressure in these compartments and in the major annulus area, whenever the system is in operation. It should also be noted that all piping and electrical penetrations between the reactor external building (RE/B) and these four piping penetration compartments would be designed to ensure the air-tight integrity required for the annulus air cleanup system to establish the specific negative pressure with a specific air in leakage.

In conclusion, it should be emphasized that the four dedicated filtered/vented pump compartments (Safeguard Component Areas), which house the four ISS mechanical subsystems, are also connected to the annulus air cleanup system by individual ducts and isolation dampers. In the event of a LOCA, the annulus air cleanup system would first be automatically aligned to establish a slight negative pressure in the overall annulus area. Subsequently, the individual and redundant safeguard component area (SCA) isolation dampers would automatically open to permit a negative pressure to be established in each of the four SCA compartments.

The boundary of each SCA compartment includes one of the four ISS pump compartments located at elevation 72.0 meters, one of the four ISS valve compartments located at elevation 77.4 meters, and one of the four ISS piping penetration compartments located at elevation 84.8 meters. Therefore, by automatically aligning the two redundant annulus air cleanup systems to the four SCA compartments, any leakage, from any component (pumps, valves, piping, instrumentation, containment penetrations, etc.) located within these four compartments, is drawn into processed by the redundant annulus air cleanup charcoal filtration units.

440.24 What type of concrete will be specified for the lower portion of the structure, particularly that beneath the reactor vessel?

RESPONSE

The concrete aggregates used in civil structures, including the containment building, will be materials normally available within the locale of the sight.

440.25 In Figure 1.2-13, the PD pump is identified as a seal injection pump. Does this mean that seal injection is its only function and that it has neither the capacity nor the pipe connections to operate as a normal charging pump? The figure also shows no filter, valves, and heat exchanger. Do they exist? The right hand portion of the figure is too cursory to be meaningful. This is a typical example of an area where better drawings will be needed for further review. Will they be provided as a part of the modules yet to be submitted or will we have to request them separately?

RESPONSE

The design of the Westinghouse reactor coolant pump seals is such that the seal injection enters a seal cavity that is in direct communication with the reactor coolant system as well as the seals. A typical situation has about eight gallons per minute seal injection entering the cavity and about three gallons per minute leaving through seal leakoff, such that five gallons per minute enters the reactor coolant system. For the WAPWR, the total seal injection flow entering the reactor coolant system would thus be about twenty gallons per minute. As a result, the backup seal injection pump which is only connected to the reactor coolant pump seal injection connection will be able to provide sufficient charging flow to make up for minor RCS leakage and to control RCS chemistry in the short term. In addition, as shown on Figure 1.2-13, the flow from the backup seal injection pump to the seals is filtered. This filtration is identical to that provided for normal seal injection, so use of the backup seal injection pump would have no adverse effect on the seals even if this operation were to continue for a long period of time.

During normal plant operation, RCP seal injection is provided from the centrifugal charging pumps. The positive displacement pump will only be used in unusual situations, such as times when both centrifugal charging pumps are out of service.

A complete flow diagram and description of the backup seal injection sub-system of the CVCS Will be provided as part of RESAR-SP/90 PDA Module 13, "Auxiliary Systems."

440.26 The list of categories for external accidents in Section 2A.2 does not include items such as floods, earthquakes, plane crashes, et. al. Are these to be included?

RESPONSE

The category of external accidents listed in Section 2A.2 includes all accidents external to the plant. Plane crashes are a class of these external accidents. Floods and earthquakes are not considered as external accidents but are treated in the design as environmental conditions as described in Table 2A-2, and Sections 3.4 and 3.7 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

440.27 Section 2A.5.2 states: "It is anticipated that the selection of a WAPWR site will not result in higher seismic risks than the nuclear power plant sites currently in existence." Certain existing sites have been found to have high risk due to earthquakes, and the staff anticipates that an advanced plant will demonstrate significant improvements based upon what we have learned. Section 2A.5.2 also contains the statements:

"The selection of 0.3G for the SSE is based on review of seismic data for existing and future sites. Existing Westinghouse PWR's outside of California are designed for SSE magnitudes equal to or less than 0.3G."

Please discuss the SP/90 plant design with respect to the above W statement and the staff response. W should prepare a plan to address seismic hazard on a generic rather than site specific basis, and identify where in the review process they intend to address this issue.

RESPONSE

The Nuclear Power Block is designed for the seismic hazard on a generic basis as described in Table 2A.2, and Chapter 3.7 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". This will provide an improvement in seismic margin above that incorporated in a plant which is designed specifically for a given site.

440.28 We have two overview comments in regard to Module 3, the work we have accomplished with other modules, and our impression based upon the preliminary review:

We have encountered many references to the Regulations and to Regulatory Guides in our review of Module 3 and our initial work with other modules. We did not find these references to be sufficiently described in many cases, and we found that a significant review time was needed to carefully study the wording of the referenced material and of the submittal. W should provide a complete and precise definition of what they mean in referencing such material, including the following words and phases:

a. fully complies

b. complies

c. meets

d. meets the intent of

e. guidance ... was followed

f. complies followed by a discussion which only covers a portion of the reference requirements

g. no statement with respect to the reference, but with a discussion of the subject that may not be complete

h. meet ... with the following clarifications

i. will demonstrate the level of conformance ... to

j. (particular equipment) ... will be incorporated in ... the design (but will it be in complete compliance with the reference requirement?)

k. ... design will include ... (see above item)

l. ... will be considered ... (see above)

m. ... demonstrate that ... to meet the requirements

n. provide ... that can

o. W will perform (will whatever is to be performed be in complete compliance with the reference?)

p. careful consideration will be given

RESPONSE

During the review of RESAR-SP/90, the NRC must remain cognizant of the fact that this document is currently in the PSAR stage. As such, certain statements in RESAR-SP/90 are written in the future tense and certain commitments are made which will be fulfilled later in the licensing process. This is consistent with the guidance of Regulatory Guide 1.70, Revision 3 which states:

> "If certain information identified in the Standard Format is not yet available at the time of submission of a PSAR because the design has not progressed sufficiently at the time of writing, the PSAR should provide the criteria and bases being used to develop the required information, the concepts and alternatives under consideration, and the schedule for completion of the design and submission of the missing information."

440.29 A comparison of the SP/90 design, as described in Module 3, and existing nuclear power plants clearly shows a response to recognized design weaknesses in existing plants. Many of the design improvements are not required by existing regulations, yet appear to provide significant improvements. Typical of these improvements are:

a. The "pump house" concept of enclosing the SI pumps and associated equipment which reduces the probability of a LOCA outside of containment and provides improvement in control of release of radioactive material following an accident.

b. Provision of a DC powered charging pump for reactor coolant pump seal injection with the attendant reduction in the probability of a seal LOCA.

c. Addition of a startup feedwater system which should reduce demand placed on the emergency feedwater system and can be used as a backup system in case of problems with the emergency feedwater system.

d. The Section 1.2.3.8.1 discussion covering use of the CVCS to diminish reliance on safety grade equipment or as a backup for such equipment, which is an excellent example of broad thinking during the design process.

We encourage W to continue with this approach of positive response to past experience and flexible thinking in the design process.

RESPONSE

(No response necessary).