AMENDMENT 3 TO RESAR-SP/90 PDA MODULE 3
INTRODUCTION AND SITE

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AMENDMENT 3 TO RESAR-SP/90 PDA MODULE 3
INTRODUCTION AND SITE

INSTRUCTION SHEET

Replace current page 1.2-3 with revised page 1.2-3.

Replace current page 1.9-5 (Table 1.9-3) with revised page 1.9-5.

Place pages 440-1 through 440-8 (Questions/Responses) after Amendment 2 (Page 240-5) in the Questions/Answers section to Module 3.

Replace current page 440-9/440-10 of Amendment 1 with revised page 440-9/440-10 in the Questions/Answers section to Module 3.

Replace current page 440-19/440-20 of Amendment 1 with revised page 440-19/440-20 in the Questions/Answers section to Module 3.

would take direct suction from the EWST and provide the required flow to the RCS and the containment spray headers. Only after all the lower elevations within the containment are flooded, would water return to the EWST via the [ ] inch diameter spillways shown on plan elevation [ ] meters (a,c) (sheet 4 of Figure 1.2-2).

The reactor external building (RE/B) essentially contains all the NPB scope systems and components not located inside the SSCV. The RE/B is located on the [ ] meters common basemat and it extends 360° around the secondary (a,c) containment (shield building). The equipment located in the RE/B has been arranged to: 1) separate the non-safety equipment from the safety related equipment; 2) separate the Train A components from the Train B components; and 3) separate the radioactive (dirty) components from the non-radioactive 3 (clean) components.

The RE/B general arrangement drawings (Figure 1.2-2, sheets 1 thru 9), show the safety related equipment generally located between building column line (A) and (H). The non-safety related equipment is generally located from column line (H) to column line (Q). For RE/B electrical train separation, Train A equipment has generally been located to the right of the RE/B centerline and Train B equipment is located to the left of the RE/B centerline. The majority of non-safety related component areas are located in radioactive control areas and the majority of safety-related component areas are located in non-radioactive control areas. The only safety-related component areas that are classified as dirty areas are the four ISS safeguard component areas (SCA) located in the shadow area beneath the sphere, between elevation [ ] meters and elevation [ ] meters. (a,c)

It should be noted that the RE/B boundary does include the building volume commonly referred to as the shadow area beneath the sphere. This building volume below elevation [ ] meters and between the primary containment (a,c) (SSCV) and the secondary containment (shield building) is subdivided into seven dedicated and totally separated zones. One of these seven zones is dedicated to the non-safety related chemical and volume control system (CVCS) pumps, valves, and piping. Two of the zones are dedicated to the two emergency

TABLE 1.9-3
SITE INTERFACE PARAMETERS

Consideration	Parameter	PDA Module	
1. Operating Basis Wind	50 yr. fastest mile wind speed $\leq$ 110 mph	3	2
2. Tornado Wind Speed	$\leq$ 320 mph	3	2
3. Tornado Missiles	$\leq$ ANSI/ANS 2.3 - 1983 Standard Design Missile Spectrum for Wind Velocity of 320 mph	3	2
4. Safe Shutdown Earthquake	$\leq$ 0.3G Horizontal ZPA with Reg. Guide 1.60 Spectra	3	
5. Operating Basis Earthquake	$\leq$ 0.1G Horizontal ZPA with Reg. Guide 1.60 Spectra	3	
6. Soil Shear Wave Velocity (V)	$\geq$ 1000 ft/sec	3	2
7. Soil Bearing Strength	Must be capable of supporting NPB (8 KSF static bearing pressure) under all specified conditions	3	
8. Flood Level	$\leq$ Finished Grade	3	
9. Safety Related Cooling Water	Max. temperature at intakes $\leq$ 95°F. Flow (later)	13	
10. Air Temperature - (Outside)	Minimum $\geq$ (-25° F) Maximum $\leq$ (100°F) Extreme Maximum $\leq$ (110°F)	7 13	2
11. Probable Max. Precipitation in a five-minute period	$\leq$ 6 inches/hr	13	3
12. Snow Load	$\leq$ 80 psf	7	
13. Accidents External to Plant	Any accident for which the consequences exceed Part 100 guidelines must have a low probability of occurrence.	3	
14. Population Distribution	Population distribution must be within the bounds used in the PRA analysis	16	

REQUEST FOR ADDITIONAL INFORMATION

WESTINGHOUSE ADVANCED PRESSURIZED WATER REACTOR (RESAR SP-90)

DOCKET NO. 50-601

The following Questions/Responses were formally transmitted in Addendum 3 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3338, dated May 13, 1988.

440.254 (Module 3, Section 1.1.1.2) You stated that the WAPWR design includes a NSSS with a thermal rating of 3816 megawatts, which includes a core thermal power of 3800 megawatts plus 16 megawatts from the reactor coolant pump heat. Are the primary coolant heat losses included in calculating the NSSS thermal rating? If not, why not?

RESPONSE:

The primary coolant system heat losses are not included in calculation of the NSSS thermal power rating, which is consistent with standard practice. However, primary coolant system heat losses are considered in calculating the heat transferred between the primary and secondary sides, and in determining the plant's electrical rating (by heat balance). Typically, primary coolant system steady state heat losses are on the order of 0.1 percent of the NSSS thermal power rating.

440-262 Our review has identified several areas in which unique aspects of the SP/90 design do not appear to have been exploited to achieve the maximum reasonable safety. These include:

(a) The diesel start and loading time requirements of a few seconds do not appear necessary with the SP/90 ECCS design. The staff believes that longer start times will enhance safety by reduction of stress and wear to the diesels. Please discuss why such short loading time are necessary.

(b) The four train primary side safeguards system was originally conceived, with one option, as having one diesel with each system. What are the quantitative difference in plant cost and safety when this is changed to the present two diesel design. Please also address the possibility that a four diesel approach may offer a diverse diesel design possibility that has not been included in the two diesel concept.

(c) Please address the use of four diesels of diverse design and with relaxed start and load time requirements with respect to the fraction of severe accidents associated with loss of all ac power.

(d) Early conceptual design of the RCS included large diameter connections which could be used for rapid depressurization. Why was this capability removed and what is the impact of the change on accident mitigation and upon risk?

(e) The containment design may allow cooling via a few nozzles which direct water onto the outside containment surface. Was consideration given to such a system of pre-installed piping and nozzles with a connection which could be used, for example, by a fire truck as a source of pumped water? If not, what would be the cost and impact upon safety if such a system were installed?

(f) Early versions of the SP/90 design included a non-safety related "pump-house" for each of the primary side safeguards systems. This appeared to offer many advantages over the present design under severe accident conditions and for control of release outside containment under a wide range of conditions. What is the cost differential (details please) and impact upon both safety and releases between the early concept and the present design?

RESPONSE:

(a) The observation that short diesel start times do not appear necessary in case of the SP/90 is correct; at the FD stages, diesel start time will be revised to 20 seconds or more.

(b) All mechanical systems of the SP/90 are compatible with either two or four emergency diesel-generators. The additional cost for four diesel generators relative to the SP/90 design has been estimated at [ ] this assumes that the present 2 way separation is maintained. With regard to the question on diverse diesel-generators, these have not been evaluated.

(c) RESAR-SP/90 PDA Moduel 16, "Probabilistic Safety Study," evaluates the effect of 4 diesel-generators on core melt frequency. Assuming an improvement of a factor of 10 in the reliability of the on-site emergency power supply (which is probably the maximum achievable) leads to a reduction in core melt frequency due to internal events from a base of 1.5 E-6 per year to a value of 0.9 E-6 per year.

(d) To our knowledge, rapid depressurization capability was never included in the SP/90 design, even at the conceptual stage. Incorporating such capability would not significantly change any of the accident sequences evaluated in the RESAR SP/90 PDA Probabilistic Safety Study.

(e) The concept of external cooling of the containment shell using pre-installed piping and nozzles coupled with an improvised water source has been evaluated early in the design stage. Two main issues were identified:

- o Large steam venting capability from the containment annulus would be needed; this could require significant changes to the design and could possibly compromise the integrity of the secondary containment.

- o Potential would exist for flooding of safety related equipment that could be useful during recovery operations (e.g. RHR pumps)

Based on the above considerations, it was decided not to include this capability.

(f) The primary objective of the so-called "pump-house" was the mitigation of interfacing LOCA's outside containment. Detailed evaluations showed that the mitigation of an RHR suction valve opening at power and subsequent pipe rupture outside containment was impractical because of the very large mass and energy releases involved. For this reason the "pump-house" concept was not adopted.

Instead, the following design-features were adopted to minimize the probability of a LOCA outside containment.

- o The design pressure of the RHR system was increased
- o The check valves in the RHR/CS pumps EWST suction lines were eliminated to allow vent back to the EWST.

Note that the latter change was made after Module 1 had been submitted and is therefore not reflected in the Integrated Safeguards System flow diagram; however, credit has been taken for this feature in Module 16, "Probabilistic Safety Study."

The following Questions/Responses were formally transmitted as part of Addendum 5 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3338, dated May 13, 1988 and were the result of Staff's review of the following NRC questions/W responses:

1) Amendment 1 (dated May 1986) to RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System."

2) Amendment 1 (dated December 1984) to RESAR-SP/90 PDA Module 3, "Introduction and Site."

3) Preliminary W responses to NRC questions, 440.242 through 440.262, submitted by staff on March 2 and March 15 of 1988.

The issues identified in the questions that follow were addressed in a W/NRC meeting held in NRC's Rockville, Maryland office on April 21, 1988. Resolution of the issues, as agreed upon by staff and Westinghouse, resulted in modifications to original Westinghouse responses in items 1, 2, and 3 above. Additionally, text changes were made in RESAR-SP/90 PDA Module 4, "Reactor Coolant System" as part of this review.

Module 3, "Introduction and Site"

440.2 How is the Improved Thermal Design Procedure (ITDP) factored in the 2% power as well as the allowances on pressure and temperature?

RESPONSE:

The Improved Thermal Design Procedure (ITDP) was used for most DNB related transients. Consistent with the methodology presented in WCAP-8567, Reference 3 in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System," allowances for power, pressure, temperature and flow are included. These uncertainties were calculated specifically for the APWR design.

440.8 How is the EFW system designed to ensure that any two EFW pumps feed to any two steam generators?

RESPONSE:

The following sentence has been added to the original Westinghouse response to 440.8 of Amendment 1 to Module 3, "Introduction and Site":

"The EFW pumps are sized such that two of four EFW pumps feeding two or more of the four steam generators provide sufficient feedwater flow and RCS heat removal."

440.13 Under what size LOCA will the steam supply to the turbine driven EFW pump not be available? Discuss the consequences under these accident conditions.

RESPONSE:

The original response to 440.13 of Amendment 1 of Module 3, "Introduction and Site," has been modified as shown for purposes of clarification:

"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 (even small LOCA's) the steam generators do not produce much if any steam."

"For example, the Chapter 15 LOCA analyses show that even for a small (3-inch) LOCA the RCS pressure is reduced below the SG saturation temperature in ~10 minutes. This pressure response shows that sufficient break flow exists to remove core decay heat. For very small breaks that cannot remove all the core decay heat, the SG steam is made available to the turbine driven EFW pumps to assure SG heat removal is maintained."

440.21 Why does the credible mass input events only include the operation of two centrifugal charging pumps, with the normal letdown isolated?

RESPONSE:

Our original response to 440.21 (Module 3) was unclear and inconsistent. To provide clarification to this response, our "draft" response to 440.256 has been revised and Subsection 5.2.2.10 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System" has been modified. The original response to 440.21 will be revised as follows: "The responses to staff questions 440.255 and 440.256 provide a discussion of the current SP/90 cold overpressure protection method, which utilizes two of four of the ISS RHR suction relief valves during all low temperature operations."

440.22 A LOCA during RHR mode may not be limited to a LOCA in a RHR recirculation loop. Discuss the consequences of a LOCA at an RCS loop during RHR mode.

RESPONSE:

Our original response to 440.22 of Amendment 1 to Module 3, "Introduction and Site," did not address the intent of the original staff question. Therefore, we have replaced our original response with the following:

"If a LOCA occurs during the RHR portion of cooldown operations or during shutdown, and if it is assumed that all four RHR pumps are damaged by running dry, the four HHSI pumps can be made available for injection. Operator action would be required to open the HHSI discharge valve and start the pumps, i.e. restore the normal ECCS alignment. Unlike many conventional PWR's, the HHSI pumps will have an uninterrupted source of water from the in-containment emergency water storage tank (they do not depend on the RHR pumps for suction flow from the containment). Since, this event is postulated to occur at least 4 hours after reactor shutdown; clearly, only one of the four HHSI pumps would provide sufficient water to maintain the water level in the reactor vessel above the fuel."

"If the LOCA is postulated to occur in one of the RHR recirculation loops outside containment, the operator would be alerted of the leakage by redundant high sump water level alarms on the MCP from the affected RHR pump compartment. The operator would take immediate action to terminate the LOCA by isolating this subsystem from the RCS."

In addition to the above, the SP/90 will of course consider/apply the results of the on-going Westinghouse Owners Group study on loss of RHR capability.

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 an 1a
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

1a developmental. Our 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

1b 1a The EFW pumps are sized such that two of four EFW pumps feeding two or more of the four steam generators provide sufficient feedwater flow and RCS heat removal.

1a Sizing of the EFW pumps is based on a feed 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 valves 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

1a

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

If a LOCA occurs during the RHR portion of cooldown operations, or during shutdown, and if it is assumed that all four RHR pumps are damaged by running dry, the four HHSI pumps can be made available for injection. Operator action would be required to open the HHSI discharge valve and start the pumps, i.e. restore the normal ECCS alignment. Unlike many conventional PWR's the HHSI pumps will have an uninterrupted source of water from the in-containment emergency water storage tank (they do not depend on the RHR pumps for suction flow from the containment). Since this event is postulated to occur at least 4 hours after reactor shutdown; clearly, only one of the four HHSI pumps would provide sufficient water to maintain the water level in the reactor vessel above the fuel. 1b

1a

If the LOCA is postulated to occur in one of the RHR recirculation loops outside containment, the operator would be alerted of the leakage by redundant high sump water level alarms on the MCP from the affected RHR pump compartment. The operator would take immediate action to terminate the LOCA by isolating this subsystem from the RCS.

In addition to the above, the SP/90 will of course consider/apply the results of the on-going Westinghouse Owners Group study on loss of RHR capability.

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1a