AMENDMENT 1a TO RESAR-SP/90 PDA MODULE 3
INTRODUCTION AND SITE

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INSTRUCTION SHEET

Replace current pages 440-9 thru 440-12 with revised pages 440-9 thru 440-12a.

Replace current pages 440-17 thru 440-22 with revised pages 440-17 thru 440-22.

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

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

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 1a 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 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 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.

(a,c)

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

1a

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. 1a 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

The responses to 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.

1a

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 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. 1a

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.

1a

1a

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

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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.

Any potential containment leakage points located above elevation
(a,c) [ ] 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 paramater defined by the containment shield building and the inside perimeter defined by the spherical steel containment. However, it should be noted that
(a,c) there 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 welds
(a,c) below elevation [ ] meters. This interconnected system of leak