WESTINGHOUSE CLASS 3

AMENDMENT 2 TO RESAR-SP/90 PDA MODULE 4
REACTOR COOLANT SYSTEM

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AMENDMENT 2 TO RESAR-SP/90 PDA MODULE 4
REACTOR COOLANT SYSTEM

INSTRUCTION SHEET

Replace current page 1.6-3 with revised page 1.6-3.

Replace current pages 1.8-22 through 1.8-25 with revised pages 1.8-22 through 1.8-25.

Replace current pages 3.2-1 through 3.2-3 with revised pages 3.2-1 through 3.2-3.

Replace current page 17.0-1 with revised page 17.0-1

Insert pages A2-1 through A2-6 in Question/Answer section.

TABLE 1.6-1 (cont)

MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status	
WCAP-8301(P) WCAP-8305	LOCA-IV Program: Loss-of-Coolant Transient Analysis	Rev 0	15.0, 15.6	7/12/74	AE	
WCAP-8302(P) WCAP-8306	SATAN-IV Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant	Rev 0	15.0, 15.6	7/12/74	AE	
WCAP-8324-A	Control of Delta Ferrite in Austenitic Stainless Steel Weldments	Rev 0	5.2	6/23/75	A	
WCAP-8370	Westinghouse ESBU/NFBU Quality Assurance Plan	Rev 11	1.9, 17B	10/06/88	U	2
WCAP-8424	Evaluation of Loss-of-Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs	Rev 1	15.3	5/30/75	U	
WCAP-8510	Method for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients	Rev 0	5.3	7/76	U	
WCAP-8567-P(P) WCAP-8568	Improved Thermal Design Procedure	Rev 0	15.0	7/75	A	
WCAP-8693	Delta Ferrite in Production Austenitic Stainless Steel Weldments	Rev 0	5.2	3/16/76	B	
WCAP-8768	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors Program Summaries - Winter 1977 through Summer 1978	Rev 2	5.4	10.78	B	

REGULATORY GUIDE 1.124, REVISION 1, JANUARY 1978, SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS

The WAPWR SP/90 design will meet the intent of this regulatory guide. However, the SP/90 plant will be designed to conform to the rules of ASME III, Subsection NF, "Component Supports," 1986 Edition or to the latest code of record. 2

2

REGULATORY GUIDE 1.133, SEPTEMBER 1977, LOOSE-PART DETECTION PROGRAM FOR THE PRIMARY SYSTEM OF LIGHT-WATER-COOLED REACTORS

Westinghouse has taken a position which takes exception to any need for regulatory guidance relative to loose parts monitoring. This position is

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the RCS are important to safety because they:

a. Assure the integrity of the reactor coolant pressure boundary.

b. Assure the capability to shut down the reactor and maintain it in a safe condition.

c. Assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

d. Contain or may contain radioactive material.

The purpose of this section is to classify structures, systems, and components according to the importance of the item in order to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Table 3.2-1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design", delineates each of the items in the plant which fall under the above-mentioned categories and the respective associated classification that the NRC, ANS, and industrial codes committees have developed. Each of the classification categories in Table 3.2-1 is addressed in the following sections. 2

The classification of specific piping runs and valves in these runs is provided in the RCS flow diagrams contained in this module. Instrumentation and electrical equipment required to shut down the plant or mitigate an accident which is associated with the RCS will be classified as 1E (or Safety Class 3 per ANS 51.1) and identified in the appropriate module. 2

3.2.1 Seismic Classification

Seismic classification criteria are set forth in 10 CFR 100 and supplemented by Regulatory Guide 1.29.

TABLE 3.2-1

CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS
FOR THE REACTOR COOLANT SYSTEM

System/Component	Location	ANSI Safety Class	Quality Assurance	Code Class	Principal Construction Codes and Standards	Seismic Category
Reactor Vessel						
Steam Generators-Primary Secondary						
Pressurizer	(See Table 3.2-1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design")					
Reactor Coolant Pumps						
Pressurizer Relief Tank						
Piping						2
Valves						

17.0 QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

The Westinghouse Energy Systems Business Unit/Nuclear Fuel Business Unit Quality Assurance Program is described in Reference 1. 2

17.1.1 References

1. "Westinghouse Energy Systems Business Unit/Nuclear Fuel Business Unit Quality Assurance Plan," WCAP-8370, Revision 11, October 1988. 2

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 2 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3304, dated January 7, 1988.

252.10 What is the basis (date, experiments, experience, etc.) for the use of Incoloy 800 as tube support plate material? (5.4.2.1, Module 4)

Response:

The steam generator tube support plate materials is Type 405 stainless steel and not Alloy 800. At the time of submittal of RESAR-SP/90 PDA Module 4, "Reactor Coolant System," a development program was underway which included examining the use of Alloy 800 for the steam generator tube supports. The design was shown not to be feasible and the materials now will be Type 405 stainless. This change in design will be addressed in our FDA submittal.

252.11 What steps have been taken to avoid corrosion/erosion of J tubes attached to the feedwater ring? (5.4.2.1, Module 4)

Response:

The feedring material is carbon steel with a specified minimum chromium content of 0.08%. The J-tubes are Alloy 600 material. These selections are to reduce the susceptibility to erosion/corrosion compared to that shown for carbon steels with lower chromium content.

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.

440.255 (Module 4, Section 5.2.2) Section 5.2.2 on page 5.2-3 states that the liquid relief valves of the residual heat removal system (RHRS) are used to protect RCS at low temperatures when the RHRS is in operation. Section 5.2.2.10 states that the pressurizer PORVs will be used for the low temperature overpressure protection (LTOP) function. Please clarify the LTOP design for the WAPWR.

RESPONSE:

Low temperature overpressure protection for the SP/90 will be provided by the RHR suction relief valves. Please see response to 440.256 and modifications to Subsection 5.2.2.10 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System."

440.256 (Module 4, Section 5.2.2) Expand this section to address the assumptions used for a mass addition event relative to the LTOP system design.

RESPONSE:

The following has been included in modifications of Subsection 5.2.2.10.2 of RESAR-SP/90 PDA Module 4, "Reactor Coolant Systems."

Westinghouse has performed evaluations to identify the relief requirements for LTOP events and verify the acceptability of using the RHR relief valves to protect the RCS. These evaluations included:

- o A preliminary determination of the RCS Appendix G limiting pressure vs temperature o Determination of the individual LTOP event mass/heat inputs and required relief valve relieving rates. Analyzed events include:

a. [ ] a,c
b.
c.
d.

The bases for these analyses included:

1) Appendix G Limit - A low Appendix G pressure limit for the RCS was selected ([ ]) on which to base a conservatively low set pressure for the RHR suction line relief valves. The [ ] psig pressure limit is judged to be lower than the actual Appendix G limit that would be calculated with reactor vessel material containing a maximum of [ ] copper. a,c a,c a,c

2) The setpoint for the RHR relief valves was established in accordance with Section III of the ASME Code, Part NC-7513. The required capacity is based on the maximum RCS expansion rate determined as described in 3) and 4) below.

3) The effects of flashing flow through the relief valve and/or choked flow in the valve discharge line were evaluated assuming the RCS temperature was at 350°F, the maximum anticipated temperature when RHR provides LTOP protection.

4) RCS expansion rates for the above two (a and b) mass input events were determined at the nominal RHR relief valve set pressure. In the calculations the maximum allowed, as manufactured, pump head/flow delivery curves were used.

5) RCS expansion rates for the above two (c and d) heat input events were conservatively determined as follows:

   o The LOFTRAN code was used to analyze the heat input and RCS expansion due to the inadvertent start of a reactor coolant pump during water solid operation.

   o The RCS expansion due to inadvertent pressurizer heater operation considered the maximum rate at which water could be displaced from the pressurizer by steam formation.

The results of this analysis show that using relief valves with the capacity of the current standard RHR suction relief valve, two of the four ISS RHR subsystems aligned to the RCS provide acceptable LTOP protection.

440.257 (Module 4, page 5.2-11) Item A states that to preclude inadvertent ECCS actuation during heatup and cooldown, blockage of the safety injection signal actuation logic below 1975 psia is required. Discuss the impact of this design relative to a LOCA during modes 3 and 4.

RESPONSE:

The initiation of a LOCA in modes 3 and 4 and the blockage of the safety injection signal actuation logic to prevent inadvertent ECCS initiation is currently being investigated generically for Westinghouse designed plants. Upon completion of this generic investigation, the impact of the SP/90 design and the applicability of the generic conclusions relative to a LOCA in modes 3 or 4 will be applied to the SP/90 design in the FDA application. (Also see response to 440.256).

The following Questions/Responses were formally transmitted in Addendum 5 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3338, dated May 13, 1988.

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 has been 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."

440.238 What is the design criteria used for sizing of the rupture disc on the pressurizer relief tank? Is the rupture disc sized to accommodate all safety and PORVs lifting per the SRP? If not, provide justification.

RESPONSE:

Design criteria O.) has been added to the design bases contained in Section 5.1.1 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System." "The pressurizer relief tank rupture discs are designed to provide sufficient relief area to be consistent with the combined relief capacity of both the pressurizer PORV's and safety valves consistent with SRP requirements."

The following Questions/Responses were formally transmitted in Addendum 6 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3354, dated July 7, 1988.

210.27 The staff's comments in Q210.25 and 210.35 also apply to portions of Table 1.8-2, Section 3.2.2, Section 3.2.3 and Section 5.2.2.6 of Module 4. These sections should be revised to agree with the response to Q210.35.

RESPONSE:

Please refer to our original response to Staff Q210.1. Westinghouse believes that the initiative taken to design the SP/90 plant to the latest industry codes and standards, including ANSI/ANS 51.1, provides additional assurance that this plant design will operate more safely and with better reliability than current nuclear power plant designs. If this issue is not settled prior to final design submittal, Westinghouse will reexamine the manner in which safety classifications are assigned for systems, components, and structures for the SP/90 plant.

210.28 The information in Table 1.8-2 which discusses exceptions taken to Regulatory Guides 1.124, Revision 1 and 1.130, Revision 1 does not completely conform the current staff positions relative to design criteria for ASME Class 1 component supports. To be acceptable, this information should be revised to provide a commitment to construct all Class 1 component supports in accordance with the rules of ASME III, Subsection NF, "Component Supports," 1986 Edition or to the Code of Record which will be applicable to the final WAPWR plant. (Reference Questions 210.60 and 210.65.)

RESPONSE:

Positions on Regulatory Guides 1.124 and 1.1[illegible] will be revised to state that the intent of the Regulatory Guides will be met. The final Westinghouse SP/90 design will conform to the rules of ASME III, Subsection NF, "Component Supports," 1986 Edition or to the latest Code of Record.