AMENDMENT 4 TO RESAR-SP/90 PDA MODULE 1
PRIMARY SIDE SAFEGUARDS SYSTEMS

WEC
P.O. Box 355
Pittsburgh, PA 15230

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INSTRUCTION SHEET

R. place current page iii with revised page iii.

Replace current page xiv with revised page xiv.

Replace current page 1.7-1 with revised page 1.7-1.

Insert pages A4-1 through A4-17 after Amendment 2 (page 281-2) in Question/Answer section.

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

LIST OF TABLES

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-1 contains a list of each piping and instrumentation diagram and the corresponding ISS figure number. Figure 1.7-2 illustrates and defines symbols and abbreviations used in the diagrams. 4

REQUEST FOR ADDITIONAL INFORMATION

WESTINGHOUSE ADVANCED PRESSURIZED WATER REACTOR (RESAR SP-90)

DOCKET NO. 50-601

The following Question/Response was formally transmitted in Addendum 1 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3304, dated January 7, 1988.

252.12 What are the fracture toughness requirements for ferritic steel pressure boundary components in ESF systems? (6.1.1, Module 1, 10)

Response:

The fracture toughness requirements are specified in the ASME Code requirements for the component of interest. These requirements are contained in paragraph NB 2300 of Section III, and no additional requirements are imposed. These requirements are for Charpy tests of the material, with minimum specified energy value or lateral expansion at a given temperature. Fracture toughness in terms of K_{IC} or K_{IR} is not a requirement.

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

440.252 (Module 1, Section 6.3.2) You stated that the ECCS pumps are protected against low flow or no flow-operation by the miniflow path. It is not clear that how pump protection could be achieved by the miniflow lines under no flow or low suction pressure conditions. It is the staff's position that the ECCS pump protection should be provided by a safety grade low flow alarm system and to assure that the pump could withstand those operating conditions during the time delay for operator actions to manually trip the pump in response to the alarms.

Response:

Each ECCS pump is protected from low low flow due to high discharge line pressure or discharge line isolation by its normally open miniflow path. The RHR/CS pump miniflow contains no valves and therefore cannot be isolated by operator action. The HHSI pump miniflow containment isolation valve is not automatically closed, is normally de-energized and if closed would be immediately alarmed in the main control room.

The EWST provides an assured suction source for all ECCS operations. Also the EWST is the only ECCS suction source, and is alarmed if the level falls below the expected low level during ECCS operation (Reference, Section 6.3.5.2.4). This alarm ensures that the operator is alerted and has time to take actions necessary to protect the ECCS pumps. The pump suction valves in the piping from the EWST are normally open, are de-energized since they are not required to be realigned for any ECCS function, and if closed would be immediately alarmed in the main control room.

The above ISS design features provide protection to eliminate the cause of low suction pressure/flow and low discharge flow, i.e.:

- o Valve alignment is alarmed
- o No valve realignments are required
- o Valves are de-energized so that more than one operator action is required to close any single valve
- o Suction source is continuously monitored and alarmed and provides time for operator action prior to loss of pump NPSH

Based on the above, it is our opinion that a safety grade, low flow alarm system is not required for ECCS.

440.253 (Module 1, Section 15.6.4) The Westinghouse LOCA evaluation model approved by the staff may not be applicable to WAPWR design with respect to plant specific configurations in node arrangement and control systems. Confirm that a new LOCA evaluation model will be prepared for the WAPWR design.

Response:

The Westinghouse 1981 Evaluation model with BASH noding was used in the W SP/90 ECCS large break analyses. Additional nodes were required in the upper plenum (to predict better transient behavior) and to account for the presence of the core reflood tanks (see W response to staff question 440.222 in RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System.") The W SP/90 large break evaluation model with BASH will be submitted to the NRC for approval concurrent with final design analyses.

440.261 (a) Generic Letter (GL) 87-12 requested information regarding lowered RCS inventory operation. Please provide a response to the generic letter with respect to the RESAR-SP/90.

(b) Please describe instrumentation provided to the operator during shutdown operations which characterize the state of the reactor coolant system (RCS). Include RCS level, RCS temperature, and residual heat removal (RHR) system performance and provide a description of the appropriateness and accuracy of each instrument with respect to its intended function. Also, include identification of audible and visual alarms used to delineate out-of-range conditions, including the values which constitute those conditions.

(c) The staff has identified that Diablo Canyon, Unit 2, was in a condition not previously analyzed by the NRC staff during the loss of RHR event of April 10, 1987 (NUREG-1269). Please describe the steps that have been taken and the future plans which will be taken to alleviate this situation for the SP/90.

(d) NUREG-1269 contains the statement "Design of the nuclear steam supply system (NSSS) did not appear to provide detail provisions for mid-loop operation." Please address this identified deficiency in PWR design with respect to the SP/90 design. Include identification of and discussion of each of the design changes in the SP/90 which represents an improvement over existing designs and establish the adequacy of the SP/90 design for lowered RCS inventory operation.

(e) NUREG-1269 identified that containment was open throughout the April 10, 1987 event, and there were no procedures to reasonably assure containment closure in the event of progression of the accident to a core damage condition. Address this situation with respect to the SP/90 design and the anticipated methods that will be used to operate the plant. Include such design considerations as the need for removal of the equipment hatch and improvements in the SP/90 design which facilitate rapid replacement of the hatch should the need arise. Similarly address other containment penetrations and potential bypass paths.

(f) The Diablo Canyon event and subsequently obtained information has shown operating procedures to be inadequate for lowered RCS inventory operation. What plans exist for recommending improved procedures and administrative controls to SP/90 owners/operators so that this situation is eliminated in the SP/90.

(g) What equipment exists in the SP/90 that can be used to assure adequate core cooling in the event of a complete loss of RHR?

(h) Evidence exists that certain Technical Specifications (TSs) may not be optimum when consideration is given to operation during non-power conditions. For example, requirements for RHR suction valve interlocks impact upon RHR reliability, RHR flow rate requirements may overly restrict flow rate range and increase the likelihood of loss of RHR due to vortexing, and TSs written on the basis of time (such as one may remove RHR from operation for an hour) perhaps are more reasonable when written on the basis of the state of the NSSS and/or of containment. Please address this topic with respect to the SP/90 design and provide recommendations for improvement, particularly with respect to the unique design aspects of the SP/90.

(i) Safety analysis reports (SARs) typically concentrate on power operation when consideration is given to many of the potential operational transients. The recent experience from the Diablo Canyon event indicated that further evaluation for plant operation at lower modes may be required. Hence, it may be prudent to address non-power operation in more depth than has been traditional. What plans exist, if any, with respect to this topic and the SP/90 program?

RESPONSE:

Introduction

This issue of lowered RCS inventory operation was raised earlier during the review of RESAR-SP/90 PDA Module 16, "Probabilistic Safety Study." The Westinghouse response contained in letter NS-NRC-87-3235 Section 3.2 addresses many of the questions raised in this enclosure and will be referred to hereafter.

Recently (12/87) the draft of Chapter 15, "Engineered Safety Systems" of the EPRI ALWR Requirements Documents was issued. Section 5.2.3.1.3 of that document provides requirements to minimize the potential for loss of RHR when the RCS level is lowered. Westinghouse has participated in the preparation of these requirements and intends to comply with them in the case of future plants such as the SP/90.

In general, Westinghouse believes that the loss of RHR function is much less likely in case of the SP/90 than for a conventional plant, however, we will certainly incorporate the results of the ongoing Westinghouse Owners Group review of this issue as applicable to the SP/90.

More detailed responses to the individual questions are provided below:

(a) Generic Letter (GL) 87-12 in general asks more detailed versions of the other parts of question 440.261. As such they will be addressed in the FDA stage.

(b) The condition of each of the four RHR pumping train is monitored by individual flow and inlet/outlet temperature instruments. Each instrument has a readout in the main control room. The RHR flow has a low flow alarm that annunciates in the MCR. The RHR inlet temperature has a high temperature alarm that also annunciates in the MCR. Note that the RHR inlet temperature provides an accurate measurement of the RCS temperature during RHR operation.

It is anticipated that the SP-90 will incorporate narrow range RCS level instruments that will be able to accurately measure the water level during mid-loop operation. These level instruments would have both readouts and low alarms located in the MCR.

(c) As stated in the introduction the issue of operation with lowered RCS inventory is much less significant for the SP/90 than it is for conventional plants. It is our position that with the addition of redundant hot leg level instrumentation and Main Control Room indication (see response to (b) above), and by proper reflection of this event in the operating procedures (see response to (f) below), this issue should be fully resolved.

(d) The Westinghouse response (Section 3.2) to the Draft BNL/NRC Report on SP/90 PDA "Probabilistic Risk Assessment" identifies the inherent improvements in the SP/90 design with regard to lowered RCS inventory operation.

(e) As identified in the Westinghouse response to the Draft BNL/NRC Report on SP/90 PDA "Probabilistic Risk Assessment," the time available before core damage would occur following loss-of-RHR is much longer for the SP/90 than for a conventional plant. The options available to the operators during this extended period of time to prevent core damage are so extensive (See response to item (g) below) that probability of core damage is essentially negligible.

During this extended period, it will also be possible to effect containment isolation, with the possib'e exception of the equipment hatch, which may require up to 8 hours to install.

(f) The lessons learned from the ongoing efforts to resolve this issue will be reflected in the SP/90 operating procedures which will be developed at the FDA stage.

(g) Following loss of all RHR pumps, the following equipment can he used to add inventory to the reactor coolant system, thereby maintaining core cooling.

- o Charging pumps (2) taking suction from the reactor makeup water and boric acid water storage tank, or from the spent fuel pit
- o High head safety injection pumps (4) taking suction from the EWST
- o Core reflood tanks (4) by opening the motor operated valve in each discharge line, which is normally closed during refueling.
- o Accumulators (4) by opening the motor operated valve in each discharge line, which is normally closed during refueling.

(h) The TS examples provided are not of particular concern in the SP/90 design e.g.

- o The RHR suction auto closure interlocks have been eliminated

- o Each RHR subsystem is intended to operate at constant flow with minimum potential for vortexing

- o Removing an RHR pump from service will still leave three RHR pumps operational

Nevertheless, in the preparation of Technical Specifications during the FDA stage, non-power conditions will be considered.

(i) As indicated before, events at non-power operation do not appear to be a matter of significant concern in case of the SP/90 design. As such, a detailed discussion of these events in the SAR does not appear to be necessary in the RESAR SP/90. Nevertheless, if as a result of operating plant problems the standard content and format of SAR's will be modified in the future, Westinghouse intends to comply in full at the FDA stage.

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 FDA 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. a,c

(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 [ ] per year. a,c

(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

- 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 ten (10) Questions/Responses were formally transmitted in Addendum 5 to RESAR-SP/90 PDA in Westinghouse Letter NS-NRC-88-3338, dated May 13, 1988. These are "Second" Round Questions in review of Amendment 1 to Module 1.

440.41 (5.4.7) Your response stated that the most recent decay heat basis used in determining the SP/90 cooldown is based on ANSI/ANS-5.1-1979, Section 3.6. Confirm that this is more conservative than the decay heat curves attached to SRP 9.2.5.

RESPONSE:

Our original response to 440.41 of Amendment 1 to Module 1, "Primary Side Safeguards System," has been revised to be consistent with ongoing ALWR requirements development.

440.42 (5.4.7) Confirm that the pressurizer PORVs and their control systems are designed to safety grade requirements.

Describe procedures for cold shutdown including time required for upper head to reach 350°F prior to RCS depressurization to prevent upper head voiding during the process. Any analysis to backup the 36 hours total time required for cold shutdown?

RESPONSE:

The pressurizer PORV's and their associated block valves are safety grade. 1 of 3 PORV's and its' associated block valves are required to provide needed RCS depressurization capability.

Reactor vessel head voiding during safety grade (natural circulation) cooldown operations is precluded by the use of the safety grade RV head vent. This head vent will be opened throughout the cooldown to provide a positive flow of cooled water through the head. This venting is performed in conjunction with HHSI pump operation in order to maintain RCS inventory.

Modifications to portions of RESAR-SP/90 PDA Module 4, "Reactor Coolant Systems," have been made to clarify the functions of safety-grade PORV's.

440.69 (5.4.7) Clarify the response to this question and describe the system function with respect to the opening of these isolation valves.

RESPONSE:

The core reflood tank and accumulator isolation valves can be opened when the RCS pressure is higher than the N_2 cover gas pressure in the tanks. This operation, as well as other valve system alignments, will be specifically addressed in the Technical Specification to be developed for each SP/90 application. The Tech. Specs. will include all valve alignments which need to be accomplished during all the plant mode changes.

440.72 (5.4.7) Mispositioning of a valve is not considered to be an active failure. How does the SP/90 factor this type of failure into its system design?

Why does Table 5.4.7-1 not include all possible single failures in the system? (Items c, d, e . . .)

RESPONSE:

The current response to 440.72 of Amendment 1 to Module 1 has been replaced with the following. The FMEA (Table 5.4.7-1) will be updated in the FDA submittal, however provided below is a description of the SP/90 single active failure criteria which was applied to the system design.

Active Failure Clarification

Valves - Active failures include the failure of a remotely operated valve to change position on demand. This includes motor-operated valves, air-operated valves and solenoid-operated valves, and excludes check valves and spring-loaded safety valves.

Other Equipment - Active failures also include the failure of a pump, fan, or diesel which is already operating, as well as failure of one of these components to start on demand. The failure of an already running pump, fan, or diesel is considered as a spurious failure. The failure of a D.C. train is not considered as a single failure.

Spurious Actuation - Another active failure is the spurious actuation of an active component; this includes the closing or opening of an MOV or the starting or stopping of a pump. As a criterion this applies only to active components in mitigating safety systems; however, as a goal it applies to all active components. An exception can be made for active components if specific design features or operating restrictions are provided that can preclude such failures (power lockout, confirmatory safety signals, position alarms, etc).

Operator Error - A single incorrect or omitted action by a human operator attempting to perform a safety related manipulation in response to an initiating event. The error is limited to the systems utilized in mitigating the initiating event and does not include thought process errors, etc., that would lead to common cause or multiple failures.

In addition, the mispositioning of a valve (manual or remote) or the disabling of a powered component (opening a circuit breaker) prior to an event is considered a single failure. An exception can be taken if specific design features are provided that can preclude such failures, such as monitor lights/alarms from limit switches, circuit continuity testing, etc.

440.74 (5.4.7) h.(2) Confirm that SP/90 has incorporated the test headers and connections for the line connecting valves 9000 and 9001.

RESPONSE:

The following paragraph has been inserted at the beginning of our original response to 440.74:

"It is noted that test connections will be added downsteam of valves 9000A (B,C,D) and 9001A (B,C,D) in the FDA submittal. These test connections will be part of the current check valve leak test system and will be utilized during each plant startup, to positively verify that valves 9000 and 9001 are both fully closed and to detect/ characterize any valve leakage. This test will enable the plant operator to detect degradation of valve isolation capability prior to high pressure power operation. This procedure should minimize the probability of intersystem LOCA by providing a periodic verification of valve integrity."

440.77 (5.4.7) What actions will be taken under the scenario described in your response to this question in responding to RHR pump runout.

RESPONSE:

Our original response to 440 77 of Amendment 1 to Module 1, "Primary Side Safeguards System," has been modified to add the following:

"This low probability, highly unlikely scenario will however be addressed in the SP/90 Emergency Operating Instructions (EOI's). If one or more RHR/CS pumps were required to be aligned; for example, for safety injection due to the failure of all four HHSI pumps: the EOI's will require that the containment spray flow path be isolated by closing the normally open spray isolation valve and that both spray isolation valves be de-energized. This will prevent a subsequent "P-signal" from opening these valves. Note that with four subsystems, one or more subsystems can be reserved for the spray function while utilizing one or more subsystems for the SI function."

440.112 (6.3) i) What criteria has been used to determine the mesh size for the water return from the containment sump to EWST to prevent debris from being carried through pumps and the reactor core?

RESPONSE:

The following has been added to our original response to 440.112 of Amendment 1 to Module 1, "Primary Side Safeguards System.":

"The mesh size for the fine screens in the EWST will be sized to prevent passage of particles greater than 3/8-inch in diameter. This is consistent with the fine screens in existing plants.

440.118 (6.3) Explain why the approach used for other W plants is adequate for W Advanced Reactor Design (SP/90).

RESPONSE:

The following has been added to our original response to 440.118 of Amendment 1 to Module 1, "Primary Side Safeguards System": "As class 1 components, these valves are analyzed to be consistent with the ASME code for faulted conditions which include the overstress conditions due to ATWS events. Note that these valves are hydrotested to ~ 3109 psig during the plant cold hydrotest.

440.121 (6.3) Does the FMEA for ECCS consider all active and passive failures and operator errors? What is the W definition of a passive failure (broken piping or a leak)?

RESPONSE:

The response to 440.121 of Amendment 1 to Module 1, "Primary Side Safeguards System," has been modified to include "passive failure criteria and criteria for operator action" used in the SP/90 system design.

440.134 (6.3) The switchover from cold leg to hot leg injection would take place one subsystem at a time. Describe the design criteria for the switchover sequence which will be backed up by ECCS analysis. (with respect to flow, timing, etc.)

RESPONSE:

The following paragraph has been added to the original response to 440.134 in Amendment 1, Module 1, "Primary Side Safeguards System," to describe the design criteria for the switchover sequence:

> "Also, since the flow delivered by one HHSI pump exceeds the core decay heat boil-off rate, the switchover of one HHSI to hot leg injection will ensure sufficient flow to effect boron dilution in the reactor vessel. At the same time the continued injection of one HHSI pump via the cold leg side of the reactor vessel will insure that the core remains covered with water."

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:

Add design criteria O.) 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 Question/Response was formally transmitted in Addendum 6 to RESAR-SP/90 PDA in Westinghouse Letter NS-NRC-88-3354, dated July 7, 1988.

210.25 The information in Table 1.8-2, Section 3.2.2 and Section 3.2.3 relative to the WAPWR alternatives to Regulatory Guide 1.26 is not currently acceptable. Specifically, the staff has not endorsed the detailed guidance in ANSI/ANS 51.1 - 1983 to determine the quality group classification of systems, components and equipment which are important to safety as defined in the Introduction to 10CFR 50, Appendix A. A discussion of the staff position on this issue is contained in question 210.35 on Module 7. Subsequent to a resolution of this issue, the information on Reg. Guide 1.26 in Table 1.8-2, Section 3.2.2 and Section 3.2.3 of Module 1 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.