ArevaEPRDCDocsPEm Resource

From: WILLIFORD Dennis (AREVA) [Dennis.Williford@areva.com]
Sent: Wednesday, June 26, 2013 5:28 PM
To: Snyder, Amy
Cc: Ford, Tanya; ANDERSON Katherine (EXTERNAL AREVA); DELANO Karen (AREVA); HONMA George (EXTERNAL AREVA); LEIGHLITER John (AREVA); LEWIS Ray (EXTERNAL AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); SHEPHERD Tracey (AREVA); VANCE Brian (AREVA); NOXON David (AREVA); LENTZ Tony (EXTERNAL AREVA); RITCHEY Calvin (AREVA)
Subject: Advanced Response to U.S. EPR Design Certification Application Final RAI No. 579, FSAR Ch.19, Question 03.08.05-31
Attachments: Advanced Response RAI 579 Q 19-366 and 19-367 - US EPR DC.pdf

Amy,

Attached is an Advanced Response to RAI 579, Questions 19-366 and 19-367 prior to the final response date of July 31, 2013 .

To keep our commitment to send a final response to these questions by the commitment date, we need to receive all NRC staff feedback and comments no later than **July 23, 2013**.

Please let me know if NRC staff has any questions or if the response to these questions can be sent as final.

Sincerely,

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
7207 IBM Drive, Mail Code CLT 2B
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Phone: 704-805-2223
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From: RYAN Tom (RS/NB)
Sent: Friday, May 10, 2013 3:11 PM
To: Snyder, Amy
Cc: DELANO Karen (RS/NB); LEIGHLITER John (RS/NB); ROMINE Judy (RS/NB); RYAN Tom (RS/NB); WILLS Tiffany (CORP/QP); ANDERSON Katherine (External AREVA NP INC.); WILLIFORD Dennis (RS/NB); Miernicki, Michael; NOXON David (RS/NB)
Subject: Response to U.S. EPR Design Certification Application Final RAI No. 579, FSAR Ch.19

Amy,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 579 Response US EPR DC.pdf," provides a schedule since a technically correct and complete response to the two questions cannot be provided at this time.

The following table indicates the respective pages in the response document, "RAI 579 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 579 — 19-366	2	3
RAI 579 — 19-367	4	4

The schedule for technically correct and complete responses to these questions is provided below.

Question #	Advanced Response Date	NRC Comment Request Date	Response Date
RAI 579 — 19-366	June 26, 2013	July 23, 2013	July 31, 2013
RAI 579 — 19-367	June 26, 2013	July 23, 2013	July 31, 2013

Sincerely,

Tom Ryan for
Dennis Williford, P.E.
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From: Snyder, Amy [mailto:Amy.Snyder@nrc.gov]
Sent: Friday, April 12, 2013 9:42 AM
To: ZZ-DL-A-USEPR-DL
Cc: Mrowca, Lynn; Pohida, Marie; Ford, Tanya; Segala, John; RYAN Tom (RS/NB)
Subject: U.S. EPR Design Certification Application Final RAI No. 579, FSAR Ch.19

Attached please find the subject request for additional information (RAI). A draft RAI was provided to you on March 19, 2013. On March 26, 2013, you informed us that the draft RAI does not contain proprietary information and that the draft RAI is clear and no further clarification is needed; However, you noted one editorial error. As result, the RAI was change to correct the editorial error.

The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs,. For any RAIs that cannot be answered **within 30 days or May 13, 2013**, it is expected that a date for receipt of this information will be provided to the staff within the 30-day period so that the staff can assess how this information will impact the published schedule.

Thank You,

Amy

Amy Snyder, U.S. EPR Design Certification Lead Project Manager
Licensing Branch 1 (LB1)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission

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Sent Date: 6/26/2013 5:27:51 PM
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From: WILLIFORD Dennis (AREVA)

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Advanced Response to

Request for Additional Information No.579, Questions19-366 and 19-367

4/12/2013
U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
SRSB Branch

Question 19-366:

In FSAR Section 5.4.7.2.1, Design Features Addressing Shutdown and Mid-Loop Operation, the FSAR states, " Safety injection via MHSI with reduced discharge head during low loop level ensures availability of the LHSI pumps for RHR function."

On page Table 3.3.1-1 (page 12 of 14), DCS Sensors, Function Processors, Manual Actuation Switches, and Trip Actuation Devices, the FSAR states that, " SIS Actuation on Low Hot Leg Loop in Modes 5 and 6 with a setpoint of 18.9 inches in the hotleg" with note (o) "With P15 permissive validated" and note (u) "With Manual SIS - Loop Level Bypass inhibited."

The staff then reviewed insight 20 on Table 19.1-108 - US EPR PRA Based Insights. Insight 20 reads, "In shutdown operation, automatic MHSI actuation on a low RCS (hot leg) loop level....". The staff found no discussion in Chapter 19 in the FSAR on the use of the Loop Level Bypass switch that is shown on FSAR Figure 7.3-2 - SIS actuation. The staff also found no discussion in Section 5.4.7.2.1 on the use of the bypass switch. Based on discussions with AREVA at the US EPR audit in February of 2013, use of this bypass switch that would prevent automated safety injection was not modeled in the PRA. Based on conversations with AREVA at the US EPR audit, this bypass switch may be used to prevent automatic actuation of Safety Injection when personnel are performing hot leg and cold leg work (such as installing nozzle dams), the highest risk period during an outage. It was also unclear if this bypass switch would be used during reactor vessel head removal operations. Based on discussions with AREVA staff at the audit, it seems that no criteria was developed documenting when the loop level bypass would be used.

(1) The staff is requesting that FSAR Chapters 19.1.6 and Chapter 5.4.7.2.1 be revised to account for use of this RCS loop level bypass switch that would prevent automated safety injection during reduced inventory operation when used.

(2) The staff is requesting AREVA to re-quantify the EPR LPSD PRA, the sensitivity studies, the importance analysis, and dominant cutsets with the SIAS removed since there is no documented criterion as to when this bypass would be used for POSs CB and D.

(3) The staff is requesting AREVA to update the FSAR Section 5.4.7.2.1 and Chapter 19 to include information specifying the location of the upper and lower instrumentation taps for the hot leg level indication

(4) The EPR PRA did not specifically model nozzle dam installation since it was considered an infrequent evolution. The EPR PRA assumes that the core will offloaded every fuel cycle. The FSAR should either include the full core offload as a COL holder item or assess the risk from nozzle dam installation. If nozzle dams are used, to ensure that the risk of a sudden loss of RCS inventory during nozzle dam installation/removal and cold leg work does not invalidate the results of the EPR PRA, the staff is requesting EPR to update the US EPR risk insight 82 and Section 5.4.7 of the FSAR to state that the following recommendations will be implemented consistent with GL 88-17 and IN 88-36. These important operator actions needs to be documented as risk significant operator actions in Chapter 19 of the FSAR:

(a) A pressurizer manway (if analysis shows this to be a sufficient vent path) or otherwise create a suitable opening to limit the pressurization which could follow an extended loss of DHR while the nozzle dams and the reactor vessel head are in place.

(b) A hotleg manway will be the first manway to be opened, and a hot leg nozzle dam will be the last dam to be installed.

(c) A hot leg manway and its associated hot leg pipe will be kept open to provide an adequate vent path whenever any cold leg openings are made.

(d) The expeditious actions in GL - 88-17 will be implemented.

(5) The EPR CDF from overdraining the RCS to reach midloop conditions (UCLD) is reported as approximately 8E-9 per calendar year. This reduction, which is orders of magnitude lower than current PWR operating data, is based on (1) failure of automatic isolation of CVS on low hot leg level and (2) failure of the operator to terminate the drain down if the automatic isolation fails. This operator failure was assumed to be an exceptionally small likelihood of 5E-5. In order for the staff to evaluate the risk of overdraining, AREVA is requested to document in Chapter 19 and Section 5.4.7.2.1 of the FSAR (a) the hotleg level at which SG nozzle dams are to be installed (b the hot leg level/set point at which this automatic isolation is supposed to occur, (c) the highest hot leg level at which vortexing is expected to initiate in the hotleg given a RHR flow rate of 2250 gpm required by TS surveillance testing, (d) the highest anticipated RCS drain rate during reduced inventory operations, and (e) a discussion of how the midloop tests referenced in FSAR Section 14.2.12.2.5 Mid-Loop Operations Verification (Test #017) will confirm item (d) above.

Response to Question 19-366:

Item (1):

U.S. EPR FSAR, Tier 2, Sections 5.4.7.2.1 and 19.1.6.1.7 will be revised to describe the function of the safety injection system (SIS) Loop Level Bypass Switch. Additional clarifying statements about the use of nozzle dams and the conditions under which the nozzle dams will be used will also be described in U.S EPR FSAR Tier 2, Section 5.4.7.2.1. U.S. EPR FSAR, Tier 2, Chapter 16, Technical Specifications, Section 3.3 will be revised in Revision 5 to address the use of the SIS Loop Level Bypass Switchand specify how long the function can be bypassed.

Item (2):

U.S. EPR certified design lower power shutdown probabilistic risk assessment (LPSD PRA), sensitivity studies, the importance analysis, and dominant cutsets with the safety injection actuation system (SIAS) removed, were not modeled due to the assumed infrequent use of the SIS bypass switch and the allowable time that the SIS bypass switch will be used. It is anticipated that nozzle dams will be used approximately once every twenty years to perform a mid-cycle steam generator tube leak repair. This is based on the performance history of similar steam generators for the N4 plants in France, which now have approximately twenty years of experience without having had any mid-cycle forced outages.

U.S. EPR FSAR, Tier 2, Sections 5.4.7.2.1 and 19.1.6.1.7 will be revised, as indicated in the attached markup, to document the criteria describing when the bypass switch will be used and the allowable duration of use.

Item (3):

The upper and lower instrument taps for the hot leg indication are installed on the top and bottom of the horizontal part of the hot leg. The level sensor taps are located on each hot leg approximately ten feet from the steam generator centerline and approximately six feet closer to the steam generator than the low head safety injection residual heat removal (LHSI RHR) discharge nozzle. U.S .EPR FSAR, Tier 2, Sections 5.4.7.2.1 and 19.1.6.1.7 will be revised to describe the location of the instrument taps.

Item (4):

The U.S. EPR certified design is based on a full fuel offload for refueling outages. U.S. EPR FSAR Tier 2, Section 5.4.7.2.1 and Table 1.8-2 will be revised to add a COL item to assess the risk (impact on the PRA and risk significant human actions) associated with the use of nozzle dams if they are to be used for normal refueling outages.

During nozzle dam installation and removal there is a risk of a sudden loss of reactor coolant system (RCS) inventory. This risk is reduced by the mid-loop level control and protection features described in U.S. EPR FSAR Tier 2, Section 5.4.7.2.1 but not totally eliminated.

PRA sensitivity analysis performed for this RAI response indicates that a mid-cycle outage once every twenty years that involves installation of nozzle dams and bypass of the SIAS for a duration of one hour or less per steam generator, will result in an incremental increase in core damage frequency (CDF) of approximately 6.0E-9 per reactor year, which is a very small change.

If the COL applicant determines that nozzle dams are to be used on a more frequent basis, then this will be evaluated via the applicant's design change control procedure, and the impact on the U.S. EPR LPSD PRA will be evaluated based on the COL applicant's PRA maintenance and update program.

U.S. EPR FSAR, Tier 2, Section 5.4.7.2.1 will be modified to:

- Require removal of the pressurizer manway if nozzle dams are used with the reactor head installed.
- Require that a hot leg manway will be the first manway to be opened, and a hot leg nozzle dam will be the last dam to be installed.
- Require a hot leg manway and its associated hot leg pipe to be kept open to provide an adequate vent path whenever any cold leg openings are made.
- Require the expeditious actions in GL 88-17 to be implemented.

Item (5):

U.S. EPR FSAR Tier 2, Section 5.4.7.2.1 will be modified to include a discussion of the following information.

a) The hot leg level at which steam generator (SG) nozzle dams are to be installed:

The SG nozzle dams will be installed with the RCS loop level control at the automatic control set point.

b) The hot leg level/set point at which automatic isolation is supposed to occur:

The automatic control system has limitation functions that will isolate or open the CVCS letdown if the level gets outside the range of the normal control set point. In addition, safety functions of the protection system will actuate if the control and limitation functions described above fail to prevent drain down of the RCS.

c) The highest hot leg level at which vortexing is expected to initiate in the hot leg given an RHR flow rate of 2250 gpm required by TS surveillance testing:

The highest hot leg level at which RHR vortexing will occur will be determined by testing at OL3 and/or Tiashan. The final Min1p setpoint will be based on the tested value, instrument uncertainty, and operating margin. The other setpoints will be selected to provide as much margin as practical between normal operation and other setpoints.

It is not reasonable to anticipate that the plant would perform in-service testing “IST” during mid-loop operations or any evolution that would cause high RHR flow. AREVA NP recommends that the plant establish a maximum allowable RHR flow during mid-loop that provides adequate core cooling but reduces the probability of suction pipe vortexing. This is consistent with NRC Generic Letter guidance and industry experience developed in the late eighties.

d) The highest anticipated RCS drain rate during reduced inventory operations:

The RCS drain rate is ultimately limited by the CVCS low pressure letdown system. The maximum capacity of the letdown system is more than sufficient to reduce level. Operation procedures and senior reactor operator (SRO) directions will limit the flow to approximately 48 gpm (10 percent of maximum) as the steam generators (SG)s begin to burp and further reduce letdown flow as the SGs finish the burping process and level reduces to the control setpoint.

e) A discussion of how the mid-loop tests referenced in U.S. EPR FSAR Tier 2, Section 14.2.12.2.5 Mid-Loop Operations Verification (Test #017) will confirm item (d) above:

The mid-loop instrument and controls will be verified by using one loop to manually raise and lower the water level in the loop that is being tested and verify the response of the loop being tested. When testing is completed in one loop, another pair of loops will be selected as the test loop, as well as, the loop that initiates raising and lowering of loop levels manually. The intent is to test all loops while providing as many operators as possible real time training on mid-loop operations. This information can also be used to baseline the training simulator
.

AREVA NP provides the following explanation of the human error probability used in the PRA for termination of the drain down associated with an uncontrolled level drop (ULD) during mid-loop operation. As discussed above, the U.S. EPR design includes an automatic isolation of the letdown on low RCS loop level, which is performed by process automation system (PAS). If the automatic isolation by PAS fails, then the U.S. EPR design also includes an automatic actuation of MHSI on low-low RCS loop level, which is performed by the protection system. The setpoints of both of these signals will be high enough to protect the RHR pumps from vortexing. In the PRA, an operator action is modeled to terminate the ULD prior to the MHSI being challenged. This operator action has a failure probability of 1E-2. There are also ULD scenarios where

automatic isolation fails and MHSI is actuated. For these scenarios, the PRA requires an operator action to isolate the letdown prior to excessive in-containment refueling water storage tank (IRWST) depletion. This operator action has a failure probability of 2E-5. The top ranking cut sets involving the ULD initiating event result from failure of automatic CVCS isolation and failure of the long-term operator action to terminate the IRWST inventory depletion after successful MHSI actuation (the LPSD PRA assumes approximately eight hours to deplete half of the IRWST volume at a CVCS flow rate of 480 gpm).

FSAR Impact:

U.S. EPR FSAR Tier 2 Sections 5.4.7.2.1 and 19.1.6.1.7, and Table 1.8-2 will be revised as described in the response and indicated on the enclosed markup.

Question 19-367:

As a result of the PRA Audit on February 4-7, 2013, the staff finds that several staff's findings from its U.S. EPR DC review have not been incorporated into the PRA Revision 1, i.e., asymmetric configuration modeling, the use of EPRI TR-102266 in estimating flooding frequency which may not include non-piping components, the use of superseded RES/OERAB/S02-01, "Fire Events – Update of U.S. Operating Experience 1986-1999," to estimate U.S. EPR fire ignition frequencies, etc. These findings are considered to be "key sources of uncertainty" that may have significant impact on the use of PRA for regulatory decision making.

The risk associated with these findings were assessed by performing sensitivity studies. However, to ensure that the uncertainties are addressed in future PRA applications, please perform a systematic search to identify all key sources of uncertainty from all PRA areas and list them in the U.S. EPR FSAR as a separate table as part of the risk insights.

Response to Question 19-367:

See the response to RAI 289 Question 19-329; U.S. EPR FSAR Tier 2 Table 19.1-131 was previously added to identify key sources of uncertainty associated with the PRA.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question

U.S. EPR Final Safety Analysis Report Markups

Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 19 of 41

Item No.	Description	Section
5.4-1	A COL applicant that references the U.S. EPR design certification will identify the edition and addenda of ASME Section XI applicable to the site specific Steam Generator inspection program.	5.4.2.5.2.2
5.4-2	A COL applicant that references the U.S. EPR design certification will assess the risk (impact on the PRA and risk significant human actions) associated with the use of nozzle dams if they are to be used for normal refueling outages.	5.4.7.2.1
6.1-1	A COL applicant that references the U.S. EPR design certification will review the fabrication and welding procedures and other QA methods of ESF component vendors to verify conformance with RGs 1.44 and 1.31.	6.1.1.1
6.1-2	A COL applicant that references the U.S. EPR design certification will define a coating application and maintenance program for components that cannot be procured with DBA qualified coatings in accordance with 10 CFR 50 Appendix B, Criterion IX.	6.1.2.3.2
6.1-3	A COL applicant that references the U.S. EPR design certification will define the coatings program and its implementation, including maintenance and repair of coatings.	6.1.2.2.2
6.2-1	A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the CLRT program described under 10 CFR 50, Appendix J.	6.2.6
6.3-1	A COL applicant that references the U.S. EPR design certification will describe the containment cleanliness program which limits debris within containment.	6.3.2.2.2
6.4-1	Deleted.	Deleted
6.4-2	A COL applicant that references the U.S. EPR design certification will provide written emergency planning and procedures in the event of a radiological or a hazardous chemical release within or near the plant, and will provide training of control room personnel.	6.4.3
6.4-3	A COL applicant that references the U.S. EPR design certification will evaluate the results of the toxic chemical accidents from Section 2.2.3, address their impact on control room habitability in accordance with RG 1.78, and if necessary, identify the types of sensors and automatic control functions required for control room operator protection.	6.4.1

- Inherent redundancy in the design of the four divisions~~trains~~ safety-related U.S. EPR SIS/RHRS, with each train having separate RCS connections.

- Automatic SIS Actuation (Protection System) and automatic stop of the LHSI pumps in RHR mode (PAS) in the event of a low loop level or low delta-P_{sat} (difference between the RCS hot leg temperature and the RCS hot leg saturation temperature). See Figure 7.3-2 - SIS Actuation.

- Manual opening and closure of the RHR suction isolation valves (in addition to interlocks) prevent unwanted RHR connection or isolation on irregular RCS pressure. See Figure 7.6-11 - RHR Isolation Valves Interlock.

- Safety injection via MHSI with reduced discharge head during low loop level ensures availability of the LHSI pumps for RHR function. This feature can be overridden briefly by use of an SIS loop level bypass switch while personnel are physically in danger. The use of the SIS loop level bypass switch is controlled by a note contained in Technical Specification 3.3.

- The RHR connection will be automatically isolated in the event of a break outside of the containment, based on the safeguard building sump level and pressure sensors. This non-safety function is performed by PAS.

- Spring-loaded safety relief valve, located at the RHR hot leg suction line, protects the SIS/RHRS against over-pressurization when in RHR mode.

- During mid-loop operations, a maximum RHR flow rate will be established which minimizes the probability of suction pipe vortexing while providing adequate decay heat removal.

- Redundant hot leg level sensors that initiate RCS make-up (safety and non-safety related) when the RCS hot leg has reached low level.

- When nozzle dams are installed the following recommendations will be implemented:

  - Removal of the pressurizer manway while the nozzle dams are installed and the reactor vessel head is in place. This action limits the pressurization of the RCS and inboard side of the nozzle dams which could follow an extended loss of decay heat removal.

  - A hot leg manway will be the first manway to be opened.

  - A hot leg nozzle dam will be the last dam to be installed.

  - A hot leg manway and its associated hot leg pipe will be kept open to provide an adequate vent path whenever any cold leg openings are made.

  - The expeditious actions in GL 88 17 to be implemented any time that nozzle dams are installed.

- During mid-loop operation, the RCS loop level is normally controlled by the CVCS low pressure reducing valve to ensure there is sufficient RCS water inventory for operation of the LHSI pumps in RHR mode. The level control, limitation, and protection features are described below:

Loop Level Control Function	The RCS loop level control during mid-loop operation is regulated by the CVCS high pressure charging pumps and CVCS low pressure reducing station. After the loop level control mode has been manually validated, certain automatic protection functions are actuated.
Max1 RCS Loop Level Limitation Function	This setpoint initiates an open command for the CVCS low pressure letdown control valve in order to prevent inadvertent filling of the steam generator bowls (without nozzle dams).
Min1 RCS Loop Level Limitation Function	This setpoint initiates full closure of the CVCS low pressure letdown control valve and the RHR and CVCS isolation valves in order to protect the LHSI pumps that are operating in RHR mode. This function covers the entire temperature range of the RHR system operation.
Min1p RCS Loop Level Safety Function	This setpoint initiates the SIS in case of low RCS level in the primary loops in the event of a sudden drop in RCS level during mid-loop operation in order to protect the RHR pumps and maintain adequate core cooling.

- The reactor pressure vessel (RPV) water level is continually monitored during outage with a level sensor. The level sensor taps are located on the top and bottom of each hot leg approximately ten feet from the steam generator center line and approximately six feet closer to the steam generator than the LHSI RHR discharge nozzle.

- Temperature sensors, located at the RCS hot legs, allow temperature measurement of each hot leg when in a reduced inventory condition.

Nozzle dams are not anticipated to be used during normal refueling outages to extend the steam generator maintenance window. The U.S. EPR certified design is based on a full fuel offload, and assumes use of nozzle dams only in case of a rare mid-cycle outage for a steam generator tube repair. A COL applicant that references the U.S. EPR design certification will assess the risk (impact on the PRA and risk significant human actions) associated with the use of nozzle dams if they are to be used for normal refueling outages.

5.4.7.2.2 Design Features Addressing Intersystem LOCA

The design features of the SIS/RHRS that address the intersystem LOCA section of Reference 17 and SECY 90-016 (Reference 18) are as follows:

- Codes and Standards / Seismic Protection – The portions of the SIS/RHRS interfacing with the RCS and located outside the containment building (in the

- RCS/RHR temperature and pressure.
- RPV level.
- IRWST temperature.
- Containment pressure and temperature.

Operator Actions in Response to Loss of Inventory

The following key operator actions are identified in the accident sequence analysis.

- Establish reactor coolant makeup.
- Start the standby RHR train.
- Establish primary bleed (given a loss of secondary cooling).
- Isolate flow diversion or letdown.
- Establish IRWST cooling.

Alarms and indications available for diagnosis for operator actions, may differ from action to action, but generally include the following:

- RHR failure cues (e.g., system trouble, no flow).
- RCS/RHR temperature and pressure.
- VCT level and coolant storage level.
- IRWST level and temperature.
- Containment pressure and temperature.

19.1.6.1.7 System Analysis

The following summarizes differences in system models versus at-power modeling:

- RHR – The system is modeled as normally operating with suction from hot legs rather than in standby with suction from IRWST. SIAS actuation is removed from the model, since this is disabled by the P14 permissive during shutdown. Therefore, a start of RHR standby pump requires operator action.
- SIAS – The safety injection signal is changed in the MHSI model to low delta P_{sat} in POS CA and to low loop level in POS CB, POS D, and POS E. .The RCS low level actuation of MHSI is described on Figure 7.3-2. This safety function initiates the SIS in the event of a sudden drop in RCS level during mid-loop operation in order to protect the RHR pumps and maintain adequate core cooling.

- The SIAS feature can be overridden briefly by use of a SIS loop level bypass switch while personnel installing nozzle dams are physically in danger. The use of the SIS loop level bypass switch is controlled by a note contained in Technical Specification 3.3 and is used only in case of a rare mid-cycle outage for a steam generator tube repair. Since the SIAS override feature is used infrequently and for a brief period it is not modeled in the PRA.

- RCS - Use of nozzle dams is not modeled in the LPSD PRA because they are not anticipated to be used in normal refueling outages. The U.S. EPR design is based on a full fuel offload, and use of nozzle dams is assumed only in case of a rare mid-cycle outage for steam generator tube repair. If nozzle dams are installed, precautions that will be taken to reduce the risk are described in Section 5.4.7.2.1.

- CVCS – Charging pumps are~~system is~~ not credited in shutdown.

- EFW – Auto reset of the P13 permissive is required for automatic EFW operation during POS C. Also, only the normal pressure control mode of MSRTs is required (MSSVs are not credited). A PCD function is disabled by P14 permissive, the MSRT pressure is set to 145 psia and is not automatically reset.

- RCP – Only two pumps are running during POS CAD and would be required to trip upon loss of motor cooling. Seal cooling is not required during shutdown.

The following summarizes LPSD systems with auto actuation signals modeled:

- RHR protective trip – Low loop level will trip the operating RHR pumps to protect the pumps and allow them to be restarted post trip either in RHR or the LHSI mode of operation. The level sensor taps are located on the top and bottom of each hot leg approximately ten feet from the steam generator center line and approximately six feet closer to the steam generator than the LHSI RHR discharge nozzle. This non-safety function is performed by PAS. Failure of this trip function is included as a failure mode of the RHR pumps. Success allows the pump to be manually recovered later.

- RHR isolation – High sump level in the SB automatically isolates the respective RHR train and trips the pump. This non-safety function is performed by PAS. and ~~This~~ is modeled in the RHR ISLOCA initiating event fault tree.

- Low pressure reducing station isolation – During an uncontrolled drain down event (ULD), low loop level automatic isolation of the low pressure reducing station is modeled. This non-safety function is performed by PAS; it initiates closure of the CVCS letdown in order to protect the RHR pumps and prevent challenge of the SIAS safety function. Failure is also assumed to result in diversion of IRWST water outside containment requiring operator response.

The probability of plugging the IRWST suction strainers is modeled the same as at-power operation (i.e., CCF). Maintenance work during shutdown could result in a higher probability of plugging. However, the IRWST design is somewhat unique in comparison to the PWR plants operating in the USA. The structure is very large with