Jeff Ciocco

From: Jeff Ciocco
Sent: Tuesday, July 29, 2008 10:33 AM
To: us-apwr-rai@mhi.co.jp
Cc: Theresa Clark; Lynn Mrowca; Jin Chung; Ruth Reyes; Harrison Botwin; Larry Burkhart
Subject: US-APWR Design Certification Application RAI No.39-548
Attachments: US-APWR DC RAI 39 SPLA 548 _2_.pdf

MHI,

Attached please find the subject request for additional information (RAI). This RAI was sent to you in draft form. The schedule we are establishing for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. However, questions 19-44, 19-55, 19-64, 19-65, 19-67, 19-68, 19-69, 19-71, 19-76, 19-77, 19-78 will come in 60 days. While for RAI 19-45, MHI will determine when they can respond and this answer will be provided within 30 days. Please submit your RAI response to the NRC Document Control Desk.

Thanks,

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REQUEST FOR ADDITIONAL INFORMATION NO. 39-548 REVISION 0

7/29/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation

Application Section: 19

SPLA Branch

QUESTIONS

19-44

Gravity injection to the reactor coolant system (RCS) from the spent fuel pool (SFP) during shutdown is a mitigation strategy that is not typically seen for reactors in the United States. Overdraining the SFP could result in damage to the stored fuel. Remove discussion of this mitigation strategy and all credit in the shutdown probabilistic risk assessment (PRA), or provide the following information for the staff's evaluation:

a. Detailed elevation drawings of the RCS and SFP, with the elevation of the high point vent (e.g., pressurizer manway) and expected equilibrium level clearly indicated
b. Design features and associated inspections, test, analyses, and acceptance criteria (ITAAC) to ensure that the SFP cannot be drained to a level that would endanger the spent fuel
c. Analysis results showing the gravity injection flow rate required to prevent boiling in the RCS
d. Graphs of the driving head and gravity injection flow rate expected at various SFP levels
e. Analysis of the consequences of overdraining the SFP (e.g., zirconium fire)
f. Detailed procedural guidance for the evolution, including precautions and limitations provided to the operators
g. Discussion of the controls to ensure that gravity injection does not occur inadvertently

19-45

(Follow-up to Question 19-6) Additional information is needed on reflux cooling during shutdown in the US-APWR. Specifically:

a. Provide a description and results of design-specific analyses demonstrating the effectiveness of reflux cooling in the US-APWR at the RCS levels assumed in the plant operating states (POS) that credit the steam generators for heat removal. Include the calculated pressures and temperatures. NUREG-1410, cited in response to Question 19-6, showed different responses at different RCS levels.
b. Discuss the impact of the time delay, temperature and pressure increase, and

any associated mode changes on subsequent plant response.
c. Provide the input parameters (including temperature, pressure, and decay heat load) used to calculate operator action timing after losses of residual heat removal (RHR) in all POS.
d. Provide the assumed contents of the steam generator tubes during all phases of plant shutdown. Will nitrogen be injected in the steam generator tubes to speed draining? If so, how does the nitrogen content affect the steam condensing surface for reflux cooling and any subsequent repressurization?

19-46

Clarify the success criteria for RHR in all POS (both in the initiating event assessment and as a mitigating system). Provide a description and results of the calculations performed to justify these success criteria. If the number of trains required is different from the numbers used to support system analyses and/or development of technical specifications (TS), state why.

19-47

(Follow-up to Question 19-20) Provide additional information on how the sensitivity study included in the response to Question 19-20 was performed. Does the stated CDF include contributions from all POS? Were operator actions related to systems without TS requirements assumed to fail? Were operator actions depending on sensors and indication without TS requirements assumed to fail? Was automatic isolation of the low-pressure letdown line on low level assumed to fail? Why does Table 19-20-1 indicate that the charging pumps, the refueling water storage pit (RWSP), and the refueling water storage auxiliary tank (RWSAT) are available? There are no TS requiring these components to be available during MODES 5 and 6. If changes to the sensitivity case are made, provide updated results for all POS after considering any PRA changes resulting from other question responses.

19-48

TS 3.9.5 and 3.9.6 for RHR during MODE 6 add a new note to the language of the standard technical specifications (STS) in NUREG-1431. Note 2 in TS 3.9.5 states that "[o]ne RHR pump operation is permitted, provided that decay heat is sufficiently small." Note 3 in TS 3.9.6 states that "[o]ne or two RHR loops operation is permitted, provided that decay heat is sufficiently small." The term "sufficiently small" is not defined in either TS or their bases. Define "sufficiently small." How are the operators expected to determine that this condition is met? Revise the TS to provide the operators with a clear understanding of the requirements for RHR during refueling.

19-49

(Follow-up to Question 19-7) Discuss how the initiating events, mitigating systems, and operator actions that are not modeled in POS 8-1, but are modeled in other POS, have been considered in the development of PRA-based insights and input to other programs such as the reliability assurance program (RAP), TS, and human factors engineering (HFE).

19-50

(Follow-up to Question 19-10) Provide a more detailed response to Question 19-10. The statement "[t]he effects of the flood barriers are considered only separation barriers between the east side and the west side of the reactor building" is unclear. Page 19.1-87 of the Design Control Document (DCD) states that the east and west side of the reactor building are physically separated by barriers such as watertight doors and that propagation between the two sides is not considered. Are these barriers maintained during shutdown? If so, how are they controlled? Clarify what other flood barriers credited in the at-power analysis could be removed during shutdown and how these barriers will be controlled. If barriers are not controlled during shutdown, revise the shutdown flooding PRA to reflect the propagation paths.

19-51

(Follow-up to Question 19-18) The response to Question 19-18 states that safety-related trains are separated by fire-resistant walls or doors. Are these barriers maintained during shutdown? If so, how are they controlled? Clarify what other fire barriers credited in the at-power analysis could be removed during shutdown and how these barriers will be controlled. If barriers are not controlled during shutdown, revise the shutdown fire PRA to reflect the propagation paths.

19-52

Revise DCD Sections 19.1.6.3.2, internal fire at low power and shutdown (LPSD), and 19.1.6.3.3, internal flood at LPSD, to include the information discussed on page 19.0-15 of Standard Review Plan (SRP) Section 19.0. For each POS, provide:

- Mean core damage frequency (CDF) and large release frequency (LRF)
- Significant core damage and large release sequences
- Significant initiating events
- Significant functions; structures, systems, and components (SSC), and operator actions
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

19-53

(Follow-up to Question 19-11) The response to Question 19-11 states that the failure rate of the suction strainers is unchanged from the at-power model. State the additional assumptions (e.g., related to containment cleanliness and foreign materials exclusion programs) that support this key assumption. Revise the table of risk insights in Chapter 19 of the DCD to include the assumptions and their dispositions.

19-54

Page 19.1-29 of the DCD indicates that the human error probabilities (HEP) for type B human failure events (i.e., over-drain and human-induced loss-of-coolant accidents (LOCA) during shutdown) are taken directly from NUREG/CR-1278 and that performance shaping factors (PSF) are not considered. NUREG-1842, the comparison of human reliability analysis (HRA) methods to good practices, states that involvement of HRA specialists is a prerequisite to a valid THERP analysis, and that treatment of everything as "nominal" without justification may indicate inadequate HRA and human factors considerations in making the THERP judgments. Provide additional information on how the THERP analysis of the type B human failures was performed. Discuss the involvement of HRA practitioners and human factors specialists in the development of the US-APWR PRA.

19-55

(Follow-up to Question 19-14) The response to Question 19-14 states that if HEPs are set to high values, the conditional core damage probability (CCDP) will increase and will depend on the error assumed. The staff needs more information to understand the importance of human reliability assumptions to the shutdown CDF. Provide the results of a sensitivity study with all HEPs related to both initiating events and mitigating systems set to a high value, such as 0.5 or the 95th percentile value.

19-56

(Follow-up to Question 19-9) Provide additional information on flow diversions during shutdown. Specifically:

a. How does the frequency of a flow diversion from the RHR system to the RWSP via motor-operated valves (MOV) 9815A/B/C/D account for the four possible valves that could cause it?
b. How is the contribution of both spurious operation and inadvertent opening considered in the evaluation of this flow diversion?
c. Justify the exclusion of all other failures (e.g., spurious operation or inadvertent opening of particular valves) that could result in a loss of RCS inventory inside or outside containment. For each, state how long the operator would have to respond to the flow diversion in each POS, including the flooded-cavity POS 5 and 7.
d. Discuss how inadvertent opening of these valves has been considered in the design of the control room and in administrative controls.

19-57

(Follow-up to Question 19-5) Page 19.1-98 of the DCD states that LOCAs caused by pipe rupture are unlikely to occur during shutdown. Provide further justification for the exclusion of LOCAs both inside and outside containment from the shutdown PRA. The response to Question 19-5 appears to indicate that pipe breaks are assumed to result in a loss of RHR and steam generator function, but pipe breaks also could cause a loss of inventory requiring mitigation.

19-58

(Follow-up to Question 19-26) Revise the DCD to include a combined license (COL) information item or similar commitment that ensures the COL applicant will develop shutdown response guidelines that satisfy NUMARC 91-06, as stated in response to Question 19-26.

19-59

(Follow-up to Question 19-27) Footnote 8 in RG 1.206, Section C.I.19, Appendix A, states that: "'PRA-based insights' are those insights identified during the DC [design certification] process that ensure that assumptions made in the PRA will remain valid in the as-to-be-built, as-to-be-operated plant and include assumptions regarding SSC and operator performance and reliability, ITAAC, interface requirements, plant features, design and operational programs, and others. The usage of the phrase is intended to be consistent with its use in Table 19.59-29 of the AP600 design control document [DCD]." In the AP600 DCD, each insight receives a disposition such as a reference to another portion of the DCD, an ITAAC, or a COL information item. Question 19-27 requested that such a disposition be added to Table 19.1-113 for each shutdown entry, as well as the inclusion of additional features that reduce shutdown risk. Amend DCD Table 19.1-113 to add the requested dispositions for all entries, and include the assumptions and insights provided in response to Question 19-27.

19-60

Provide the assumed water volume in the RWSP during shutdown and the assumed water volume that is transferred to the refueling cavity. Would safety injection (SI) draw from the RWSP following a LOCA in POS 5 or 7 when the cavity is flooded? If so, clarify, with supporting drawings as needed, whether the suction point from the RWSP remains covered with water in POS 5 and 7.

19-61

Page 5.4-42 of the DCD states that "[a]t this water level [0.33 feet above mid-loop], the air/water interface is at close proximity to the RHR suction nozzles located on the hot legs, and thus, reduces the possibility of air entrainment into the RHR pump suction." This statement appears to contradict itself. Clarify how an air/water interface close to the RHR suction nozzles reduces the possibility of air entrainment, or revise the DCD to correct the statement. In addition, discuss any design improvements made to the RCS and RHR system to reduce shutdown risk, such as self-venting suction lines, suction nozzle modifications, or vertically offset hot and cold legs.

19-62

Justify the exclusion of low-temperature overpressure (LTOP) events from the shutdown PRA. Discuss the mass or energy input that would cause the RHR suction relief valves to open when the RCS is water-solid during shutdown. Provide the likelihood that the valves will stick open, and discuss how the shutdown PRA handles this scenario.

19-63

Page 19.1-97 of the DCD states that "[r]eactivity insertion event will progress phenomena very slowly by boron dilution and long grace periods so that this event has enough time to recovery." Provide further justification for the exclusion of boron dilution events from the shutdown PRA, including quantitative discussion of the "long grace periods" and associated automatic and manual mitigating actions.

19-64

(Follow-up to Question 19-3) The response to Question 19-3 states that the RCS is opened by opening of the steam generator manhole lids, and that other openings such as the pressurizer manway or pressurizer safety valve (PSV) vent are opened at approximately the same time or later. Clarify whether an RCS vent is open during draining to mid-loop to prevent drawing a vacuum in the RCS.

19-65

Tables 19.1-76 and 19.1-77 of the DCD indicate that the RCS is closed and the steam generators are isolated in POS 4-3 and 8-1. Clarify the vent status of the RCS in these POS. The list of expeditious actions in Generic Letter (GL) 88-17 includes a direction to "[i]mplement procedures and administrative controls that reasonably assure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the [reactor vessel]." Discuss how this condition is met during shutdown in the US-APWR.

19-66

Confirm that the reactor vessel bottom head has no penetrations that could lead to inadvertent draining of the RCS during shutdown. Discuss this design improvement in the context of shutdown risk and add it to the list of risk insights in Table 19.1-113 of the DCD.

19-67

TS 3.9.5 and 3.9.6, related to RHR during MODE 6, require containment closure within four hours whenever no RHR loops are available. The bases for this TS state that "[t]he Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time." Provide descriptions and results of time-to-boil calculations from the shutdown PRA that support this statement.

19-68

(Follow-up to Question 19-21) The response to Question 19-21 states that availability of offsite power is assumed the same as at power and that a sensitivity analysis increasing the LOOP frequency 3 times resulted in a CDF increase of 40 percent. Generic data in NUREG/CR-6890 indicates a shutdown LOOP frequency of 0.196 per reactor shutdown year (/rsy), nearly five times higher than the value assumed in the shutdown PRA. Revise the shutdown PRA to use a shutdown-specific LOOP frequency; alternatively, provide a list of assumptions and associated requirements and controls that justify the use of an at-power LOOP frequency during shutdown. Clarify why Table 19.1-80 of the DCD indicates that the offsite power transformers are in standby status during shutdown.

19-69

(Follow-up to Question 19-1) As stated in the response to Question 19-1, page 19.1-103 of the DCD indicates that the "allowable" LOOP recovery time is one hour. Provide justification for this assumption. Do any LOOP-initiated loss-of-RHR scenarios result in boiling in the RCS in less than an hour? If so, describe the scenario and provide a description and results of the time-to-boil calculation. Describe procedures and training related to closure of the equipment hatch and other containment penetrations without offsite power. State how long containment closure is expected to take both with and without offsite power.

19-70

Discuss whether any gravity-driven sources of borated water (other than the SFP discussed above) are available for injection following a loss of inventory during shutdown. At operating plants, the ability to inject from the refueling water storage tank (RWST) is an important mitigation strategy during shutdown, but the RWSP in the US-APWR is below the RCS elevation. Discuss how this design feature, which enhances safety by eliminating the need for recirculation switchover following a LOCA, affects shutdown risk.

19-71

(Follow-up to Question 19-25) The response to Question 19-25 appears to assess the impact of Type A and B outages only on LOCA-initiated accident sequences. Amend the response to include all initiating events modeled in the shutdown PRA. If the impact is significant, the baseline PRA results should be revised to reflect realistic plant outages rather than treating the exclusion of certain outage types with a sensitivity study.

19-72

(Follow-up to Question 19-8) The response to Question 19-8 discusses the impact of modeling the charging and SI systems differently from TS requirements. Will the next update of the US-APWR PRA modify the success criteria for these systems so they match outages required by TS for LTOP? Although the impact is not large, the discrepancy is not merely a modeling assumption to be justified with a sensitivity study, but rather a condition not allowed by TS during shutdown.

19-73

What indication of temperature, pressure, and level is available to the operators during shutdown? For each, state the type of sensor, its location in the RCS, any associated alarms and trips, and the controls that ensure the indication is available during shutdown. Discuss whether these sensors are susceptible to errors identified at current plants (e.g., errors in differential pressure caused by RCS inventory swept into the pressurizer, failures of Tygon tubing, and inaccurate hot leg temperature measurement after a loss of flow).

19-74

Define "mid-loop" for the US-APWR. To what elevation will the RCS be drained to allow steam generator maintenance and nozzle dam installation? Provide the location of the RHR hot leg suction nozzle.

19-75

What is the design pressure of the nozzle dams to be used during shutdown of the US-APWR? Discuss the analysis performed to calculate this design pressure. Compare the design pressure of the nozzle dams to the expected pressure following a loss of RHR in all POS.

19-76

So that the staff can understand the US-APWR shutdown strategy, describe the expected shutdown sequence of events from entry into MODE 5 until the reactor cavity is flooded for refueling and during startup from the time when reactor cavity draining begins until entry into MODE 4. Describe the approach taken (e.g., tasks performed, systems and equipment used) for each step, including but not limited to:

a. Depressurization before draining the RCS

b. Reduction of RCS level to mid-loop

c. Draining the steam generator tubes

d. Level control during mid-loop

e. Draining the refueling cavity after refueling

f. Vacuum fill of the RCS

19-77

The shutdown PRA appears to credit injection from SI and charging, but the staff could not find discussion of a primary bleed path as in the at-power model. Revise the DCD (and PRA, if necessary) to include a discussion of feed and bleed during shutdown, including how equipment and operator failures of the primary bleed path are modeled in the PRA, calculations supporting operator action timing, and the success criteria for both injection and bleed capacity in all POS.

19-78

(Follow-up to Question 19-13) Table 19.1-79 in the DCD indicates that the RCS leakage test occurs between POS 9 (cold shutdown) and POS 11 (cold and hot shutdown). However, current plants' RCS leakage tests are generally performed at operating pressure and temperature, which would appear to place the plant in MODE 3. Provide further clarification to the staff on this state, specifically:

a. Describe the general procedure for the test.
b. Provide the temperature, pressure, and TS MODE achieved during the RCS leakage test state (POS 10).
c. Confirm at what point in the outage the test is performed.