US-APWRRAIsPEm Resource

From: Ciocco, Jeff
Sent: Wednesday, January 30, 2013 6:02 AM
To: us-apwr-rai@mhi.co.jp; US-APWRRAIsPEm Resource
Cc: Reddy, Devender; Landry, Ralph; Ward, William; Hamzehee, Hossein
Subject: US-APWR Design Certification Application RAI 988-7021 (7.8)
Attachments: US-APWR DC RAI 988 SRSB 7021.pdf

MHI,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs.

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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REQUEST FOR ADDITIONAL INFORMATION 988-7021

Issue Date: 1/29/2013

Application Title: US-APWR Design Certification - Docket Number 52-021

Operating Company: Mitsubishi Heavy Industries

Docket No. 52-021

Review Section: 07.08 - Diverse Instrumentation and Control Systems
Application Section: MUAP-07014 (Rev 5) - D3 Coping Analysis

QUESTIONS

07.08-26

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

In Section 5.1.2, “Increase in FW flow as a result of FW system malfunction,” of Technical Report MUAP-07014-P (R5), the applicant refers to DCD Section 15.1.2 and states that for this event the RCS pressure limit is not challenged and the reactor power is approximately constant and DNB does not occur even if the high-high steam generator (SG) water level reactor trip is not assumed. Therefore, the core coolability is maintained and for this event concurrent with a CCF, thereby the dose associated does not exceed the guidelines limits.

Furthermore, the graph depicted in Figure 15.1.2-7, “DNBR versus Time” for increase in FW flow, shows that the DNBR curve continues downward and rises approximately at 30 seconds instead of decreasing or becoming flat. Since, the applicant referenced DCD Section 15.1.2 and used its results and categorized this event as “equivalent protection” event in the D3 coping analysis, the staff requests the applicant to provide additional information and justification for the following:

Provide justification for why this event is categorized as an “equivalent protection,” and a coping analysis is not required. What happens to the DNBR in DCD Figure 15.1.2-7 as the DNBR appears to still be decreasing. Does the curve continue to decrease without the high-high level SG trip? Demonstrate that the DNB does not occur without a reactor trip.

07.08-27

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

In Section 5.2.6, “Loss of Non-Emergency AC Power to the Station Auxiliaries,” of Technical Report MUAP-07014-P (R5), the applicant categorized this event as “equivalent protection“ and an analysis is not performed. For this event, it is assumed to result in the loss of all power to station auxiliaries and a complete loss of external (offsite) grid accompanied by a turbine-generator trip or loss of the onsite AC distribution system. The staff requests the following:

a) Revision 3 of DCD Chapter 5, Table 5.4.10.3 states that the US-APWR reactor coolant system hydrostatic test pressure is 3106 psig. Justify why 3200 psig, as identified in MUAP-07014, Revision 5, section 4.3, is an acceptable pressure acceptance criteria or provide a new pressure acceptance criteria.

b) It is not clear how reactor coolant pressure integrity is demonstrated. What is the pressurizer high-pressure setpoint to trip the reactor? Is the trip setpoint adequate to protect the 3200 psig limit or does it need actuation of the relief valves (RVs) to relieve the pressure?

c) Revision 3 of DCD Section 15.8, section 15.8.2, identifies the pressure limit as 3200 psia. MUAP-07014, Revision 5, section 4.3, says the limit is 3200 psig. The units do not match. Identify the incorrect unit and correct it.

d) DAS trip is credited for coolability for this event. What is the heat sink and what is the source of water to the heat-sink, as the AFW is not available. Because, AFW pump is motor driven and loss of AC power results in loss of power supply for the MG set. What motive force is available to drive the AFW pump?

07.08-28

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.
In Chapter 15, subsection 15.5.2, "Chemical and Volume Control System (CVCS) malfunction…" event, it states that the boration can cause an insertion of negative reactivity, which in turn can result in a power and RCS pressure decrease. There would be very little or no reactivity inserted, because the VCT and RCS are approx at the same boron concentration.

As a result, the CVS boron is assumed to be injected at the RCS boron concentration, and the event is analyzed for pressurizer over-fill only. Thus, the Chapter 15 does not indicate any analysis performed for the pressurizer relief water. Further, according to Section 5.5.2, MUAP-07014 (R5), the time available (for the operator action) is more than (at least) 60 minutes (to mitigate this event) because the pressurizer safety valve has sufficient capacity to be less than the criterion for RCPB. However, the applicant stated that the HFE analysis would confirm that sufficient margin exists between the time available and time required for local actions as discussed in Section 3.4 of the MUAP-07014-P (R5).

Therefore, the staff requests for the following additional information:

a) Justify the pressurizer safety valves are large enough to pass maximum RCS inventory increase. Describe if limiting case is a loss of letdown or additional charging pump operations.

b) How is the 60 minutes, described on pages 5-50/51, Section 5.5.2 of MUAP-07014 (R5), related to pressuizer safety valve relief capacity? Clearly state how the 60 minutes is determined.

07.08-29

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

For LB-LOCA with a concurrent CCF, according to Section 5.6.5.1 of MUAP-07014-P (R5), the pressurizer pressure decreases rapidly to reach the automated ECCS actuation setpoint, which results in the DAS SI actuation and a DHP alarm. The time available from the reactor trip actuation alarm to manual actuation of containment spray is more than 24 hours. The applicant further states that HFE analysis to confirm sufficient margin between time available and time required for local actions as discussed in Section 3.4. Further, the applicant states that PRA analysis in MUAP-07030-P (R3) provides the time (withheld as proprietary) that it takes from the initiation of a LB-LOCA to containment failure due to overpressure with failure of SI containment spray. The staff requests the applicant to provide the additional information for the following:

HFE analysis needs to confirm sufficient margin between time available and time required for local actions with respect to mitigating pressure boundary integrity. Explain the relation between containment spray actuation time of 24 hours and the PRA analysis time from the initiation of a LB-LOCA to containment failure, as it relates to pressure boundary integrity. Again, provide a brief description and justification of Section 3.4 analysis with respect to this event.

07.08-30

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

According to MUAP-07014-P (R5), section 5.1.4, the inadvertent opening of steam generator (SG) safety valves, safety relief valves, depressurization valves, or turbine bypass valves (TBVs) can cause a rapid increase in steam flow and a depressurization of the secondary system. Further, the report states that DCD Section 15.1.4 evaluates this event from hot standby conditions. Therefore, this event is not evaluated separately in the D3 coping analysis. In this regard, the staff requests for the following additional information:

a) Is there a common cause failure (CCF) which would lead to an opening of the TBVs causing a cooldown greater than that of the steam line break (SLB) accident? If so, provide the basis for the CCF failure.

b) Also, provide the number of TBVs opened, the maximum steam flowrate caused by a CCF of the turbine bypass and compare that value to the SLB D3 coping analysis steam flowrate.

c) If the CCF caused turbine bypass cooldown is greater than the SLB D3 analysis, provide plots of core power, RCS pressure, and DNBR verses time.