REQUEST FOR ADDITIONAL INFORMATION NO. 104-1622 REVISION 1

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19.1

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 1 (AP1000/EPR Projects) (SPLA)

19-201

The following statement is made in Section 6A.9.1.3 of Revision 1 of the PRA report: "These valves [Reactor Coolant Pump Thermal Barrier Cooling Water Outlet Valves] automatically close on a high flow rate signal at the outlet of the component cooling line in the event of in-leakage from the thermal barrier, and prevents the in-leakage from the entire CCWS." However, these valves are not shown in the simplified system diagrams included in the PRA and their failure is not listed in the fault tree basic events (Table 6A.9-7). Please explain whether and how the failure of these valves to close automatically was modeled in the PRA. Is the capability of automatic closure of these valves an important design feature that prevents or mitigates certain accidents? If this is true, please describe such accidents and discuss any assumptions that were made in evaluating their associated risks or in screening out such accidents from a detailed evaluation in the PRA.

19-202

The success criteria of the component cooling water (CCW) system are discussed in Section 6A.9.2.2 and Table 6A.9-2 of Revision 1 of the US-APWR PRA report. However, these success criteria are not clearly defined and appear to be inconsistent with the definition of the corresponding fault tree top events as well as with event tree top events. For example, fault tree CWS-00A (Table 6A.9-3) is described as the failure to supply adequate CCW through supply header line "A" during an accident that is not initiated by a very small LOCA (VSLOCA) or a loss of offsite power (LOOP). However, the VSLOCA and LOOP event trees indicate that CCW is needed to cool several credited mitigating equipment, such as the containment spray/residual heat removal (CS/RHR) pumps. Therefore, the fault trees CWS-00A, CWS-00B, CWS-00C and CWS-00D (appropriately modified to address event-specific conditions, if any) are also applicable to VSLOCA and LOOP accidents. Although it is not explicitly stated, the fault trees that are reported as specific to VSLOCA and LOOP accidents in Section 6A.9.4.1 (e.g., CWS-VS-00A1 and CWS-R2) are needed to model the loss of CCW to the non-safety related charging pumps and the reactor coolant pump (RCP) seals, respectively. In addition, the definition of the fault tree top events CWS-R2 and CWS-R4 for LOOP sequences, and the associated success criteria, are not clear. For example, the fault tree top event "CWS-R4" is defined as "CCWS fails to restart given B.O sequence" and indicates that the re-start of one of four CCW pumps following a LOOP event is needed

for success without explaining how this fault tree is used and why CCW flow to any one of the four supply headers is a success. Please clarify and revise as necessary.

19-203

The component cooling water (CCW) supply and return header isolation valves are tested every 24 months at refueling. Please verify the applicability of operating reactor demand failure rates to equipment used in the US-APWR which has much longer testing intervals (e.g., 24 months). In addition, please address the following: (1) The presence of the same event descriptions for different events in Revision 1 of the US-APWR PRA report (e.g., CWSPCBDCWPD and CWSPCBDCWPD-R in Table 6A.9-7); (2) The reason for not considering common cause failure (CCF) of the CCW supply and return header isolation valves to control; and (3) The probability of 1.2E-8 used for the CCF of all four CCW pumps (basic event CWSCF4PCYR-FF) and all four emergency service water system (ESWS) pumps (basic event SWSCF4PMYR-FF) to continue running for their entire mission time.

19-204

The CS/RHR heat exchanger cooling water outlet motor-operated valves (labeled 114A, B, C and D in the PRA) are discussed in Attachment 6A.9 (Component Cooling Water/ Essential Service Water System) of Revision 1 of the US-APWR PRA report but they are modeled in Attachment 6A.3 (Containment Spray/Residual Heat Removal System). A clear definition of a system's boundary in the fault tree analysis is necessary to avoid missing important failures in the PRA model. Please include a statement in Attachment 6A.9 referring to the modeling of these valves in Attachment 6A.3. The CS/RHR heat exchanger cooling water outlet motor-operated valves are tested every 24 months at refueling. Please verify the applicability of operating reactor demand failure rates to equipment used in the US-APWR which has much longer testing intervals (e.g., 24 months).

19-205

Please address the following questions related to "alternate component cooling" (i.e., cooling of the charging pumps when component cooling water is lost by using the fire protection water supply system or the non-essential chilled water system) discussed in Attachment 6A.10 of Revision 1 of the PRA report:

(a) Motor-operated valves ACWCH6A and B are shown as normally open valves which must close for "alternate component cooling." However, Tables 6A.10-4 (Basic Events) and 6A.10-5 (Common Cause Failure Events) do not include a failure to close event but a failure to open event. This is also the case in the associated fault tree ACW (gates ACW-CH06A and B) where the failure of these valves to open, instead of the failure to close, is shown.

(b) The probability of operator failure (human error) to establish "alternate component cooling" by using the fire protection water supply system (ACWOO02FS) or the non-essential chilled water system (ACWOO02CT) is assumed to be 2E-2. This probability is the "lower bound" of the human error probability (HEP) estimated in Chapter 9 of the US-APWR PRA report for events ACWOO02FS and

ACWOO02CT. It is argued that the lower bound of the HEP is a good estimate because "the frequent training has made operators very familiar with the accident sequence" and this reduces the "extremely high" stress level used in the calculation of the HEP. However, there is no COL action requirement proposed to ensure that such "frequent training" will take place.

(c) Explain how conditional HEPs were estimated. For example, the failure probability to establish "alternate component cooling" by using the non-essential chilled water system (event ACWOO02CT) given failure to establish "alternate component cooling" by using the fire protection water supply system (event ACWOO02FS probability set to 1) is assumed to be 5.1E-1 without any discussion or explanation.

(d) There is no reference to test and maintenance requirements for the motor-driven or diesel-driven pumps of the fire protection water supply system and the non-essential chilled water system or for the motor-operated isolation valves that are required to change status to establish "alternate component cooling." Please verify the applicability of operating reactor demand failure rates to equipment used in the US-APWR which has much longer testing intervals (e.g., 24 months) and different testing and maintenance requirements.

(e) The failure to establish "alternate component cooling" by using the fire protection water supply system, due to common cause failure (CCF) of motor-operated valves (MOVs) in both the supply and return lines, was modeled. However, it appears that not all CCF combinations of MOVs were considered, such as the CCF of MOVs ACWCH1A and ACWCH2B. Please discuss.