**REQUEST FOR ADDITIONAL INFORMATION NO. 148-1700 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)

19-271

Please address the following questions related to the alternate containment cooling discussed in Attachment 6A.14.1 of Revision 1 of the US-APWR PRA report.

(a) Alternate containment cooling is credited in the PRA when the emergency containment cooling by the CS/RHR system is lost. In Appendix 6A.14.1.1.1 it is stated: "The alternate containment cooling is performed with heat removal from train A, B of the component cooling water (CCW) system to the containment vessel recirculation units A, B, C and D…" This statement implies that cooling water is used only from CCW surge tank A (the one which is associated with pumps (trains) A and B). However, it is also stated in the same paragraph that "… the surge tanks A, B should be pressurized …," which implies that cooling water is used from both CCW surge tanks. On the other hand, the simplified system diagram (Figure 6A.14.1-1) shows a nitrogen supply system only for CCW surge tank A. Please clarify.

(b) No hardware failures are modeled. The simplified system diagram (Figure 6A.14.1-1) and the human actions listed in Section 6A.14.1.1.3 indicate the existence of several sets of valves which can fail to open or close due to common cause.

(c) The human actions listed in Section 6A.14.1.1.3 indicate that the operators have to open the recirculation unit inlet valves CH-5, CH-6, CH-7 and CH-8 for alternate containment cooling. However, the simplified system diagram (Figure 6A.14.1-1) shows valves CH-5, CH-6 and CH-7 as normally open during operation. Please clarify.

(d) The impact of the CCW system re-alignment, needed to provide alternate containment cooling, on the availability and reliability of other mitigating systems credited in the same accident sequences needs to be investigated and any dependencies be identified and modeled in the PRA. Additionally, the potential for introducing new accident sequences should be investigated. Please discuss.

(e) The probability of the operator failure to re-align the CCW system for alternate containment cooling (event NCCOO02CCW) was estimated to be 2.6E-2. This probability is based on several assumptions, such as a rule-based behavior and a moderately high stress level. However, there is no discussion about the bases of these assumptions. For example, what is the assumed time window for operators to complete the required actions? Is the assumed time window realistic? Also, there are no

requirements or guidance for the combined operating license (COL) applicant/holder to ensure the implementation of what is assumed in the design certification. For example, the assumption of rule-based behavior assumes that the COL applicant/holder will develop an emergency operating procedure (EOP) for alternate containment cooling. A COL action item should be included in the design control document (DCD) to ensure that such an EOP will be developed.

19-272

Please address the following questions related to the emergency and alternate ac gas turbine generators (GTGs) discussed in Attachment 6A.11 of Revision 1 of the US-APWR PRA report.

(a) Section 6A.11.1.4 “Test and Maintenance” does not provide any information about the assumed testing and maintenance strategy for GTGs in the PRA other than the fact that the maintenance of the class 1E GTG is performed on line. Please provide all relevant information and assumptions regarding the testing and maintenance strategy on which the assessed GTG failure probabilities are based and verify the applicability of operating reactor experience to the US-APWR design.

(b) It is stated in Chapter 7 of the PRA report that “U.S. generic data of diesel generators are conservatively applied to gas turbine generators.” However, this statement is not supported by the experience with non-safety GTGs used at some U.S. nuclear power plants and no results of studies or analyses are provided to support the assumption in the PRA that the US-APWR GTGs will be at least as reliable as diesel generators. Two sensitivity studies were performed, which are reported in Chapter 18 of the PRA report as well as in Chapter 19 of the US-APWR design control document (DCD), to investigate the sensitivity of the PRA results to the recognized uncertainty associated with GTG data. One sensitivity study was performed to investigate the impact of potentially higher failure rates than those considered in the baseline case (as the industry experience with non-safety related GTGs indicates). The other sensitivity study is based on common cause failure (CCF) parameters of “general components,” which are smaller than the CCF parameters of diesel generators used in the baseline case. However, no basis is provided to justify why the CCF parameters for the GTGs used in the US-APWR design cannot be higher than the CCF parameters for “general components.”

(c) From an examination of the reported minimum cutsets and risk importance ranking, it appears that not all combinations of two GTGs were considered in the CCF analysis. For example, explain the reason why the CCF of the sets AB and BD are not shown in the results while AC, AD, BC, and CD are shown.

(d) The CCF probability of all GTGs to fail to run at some time during the first hour (event EPSCF4DLSRDG-ALL) was estimated to be 1.6E-4. This probability is about an order of magnitude lower than what the staff estimates (1.5E-3) based on the reported GTG failure rate (2.9E-2 per demand) and the assumed CCF parameters for a diesel generator set of four (beta = 0.13, gamma = .7, delta = 0.57). Similarly, the CCF probability of both alternate ac (AAC) GTGs to fail to run at some time during the first hour (event EPSCF2DLSRDGP-ALL) was estimated to be 2.3E-4, which is about an order of magnitude smaller than what the staff estimates (2.2E-3) based on the reported

GTG failure rate (2.9E-2 per demand) and the assumed CCF parameters for a diesel generator set of two (beta = 0.077). Please justify these CCF probabilities.

19-273

The modeling of Reactor Coolant Pump (RCP) seal LOCAs in the US-APWR PRA are discussed in Appendix 6A.14.2 of Revision 1 of the PRA report. It is stated: "New type O-ring which is improved for resistance to heat and pressure will be used in the US-APWR. But the RCP seal LOCA model for the US-APWR conservatively uses a mode[l] for the old type O-ring." However, the old O-ring model was modified by assuming that the maximum leak rate of 480 gpm per RCP will occur if cooling of the seals is lost for more than one hour. Although the assumed leak rate is very conservative (even according to the old O-ring model), the assumption that the RCP seal adopted by the US-APWR can keep its integrity for at least one hour without water cooling is not consistent with either the new or the old O-ring model. According to these models, there is a small probability (about 5E-3) that a leak rate of 480 gpm will develop within 10 to 30 minutes following loss of seal cooling. Please provide a more detailed discussion and the reason for using this "modified" old O-ring model in the US-APWR PRA as well as the basis for the assumption that the adopted RCP seal can keep its integrity for at least one hour without water cooling.

19-274

Please address the following questions related to the refueling water storage pit (RWSP) discussed in Attachment 6A.14.3 of Revision 1 of the US-APWR PRA report.

(a) It is stated that no potential common cause failures were identified for fault tree RWS. However, the containment sump strainers ST01A, B, C and D as well as the two motor-operated isolation valves (002 and 003) in the refueling water recirculation line can fail due to common cause. Also, the CCF (plug) of the containment sump strainers is highly risk significant. Please explain.

(b) It is stated that the list of single failures of components associated with fault tree RWS is shown in Table 6A.14.3-4. However, only a part of such failures is listed in Table 6A.14.3-4. For example the failure of motor-operated isolation valves 8820 and 9007 are listed with other systems in different tables. Please explain.

19-275

Please address the following questions related to the heating, ventilation and air conditioning (HVAC) system discussed in Attachment 6A.14.4 of Revision 1 of the US-APWR PRA report.

(a) The following statement is made (Section 6A.14.4.1.1): "The PRA models the HVAC systems that have the potential to significantly impact the mitigation system functional reliability. Discussion is provided below for each HVAC system listed above, describing how it impacts the reliability of mitigation function and how it is treated in the model." However, the provided discussion is not detailed enough to show why most of the

mitigating systems do not need HVAC for at least 24 hours following an accident. For example, it is stated that the turbine-driven (T-D) emergency feedwater (EFW) pumps can operate under high ambient air temperature conditions and, therefore, HVAC is not modeled in the PRA. More detailed information is needed to qualify how these assumed "high ambient air temperature conditions" compare to the design limit. On the other hand, if the T-D EFW pumps can operate at high ambient air temperature conditions, they have a design feature credited in the PRA which must be listed in the appropriate Chapter 19 section (Table 19.1-115) of the US-APWR design control document (DCD). Also, no discussion is provided about the impact, and modeling in the PRA, of the loss of HVAC to the main control room (MCR).

(b) It is stated that HVAC of class 1E electric areas is not modeled in the PRA because it is running during normal operation and, therefore, its reliability during the mission time needed to mitigate an accident is expected to be high. However, this argument is not supported by the failure probabilities reported in Table 6A.14.4-4. For example, the probabilities of an essential chiller unit or an HVAC fan to fail to run for 24 hours are about 2E-3 and their associated common cause failure (CCF) probabilities, which are not reported, are most likely in the 1E-4 to 1E-5 range. These probabilities do not appear to be negligible since the availability of important safety systems, such as the class 1E electrical systems, is impacted. Please discuss.

(c) Common cause failure (CCF) of chillers (even all four) is modeled, as the reported results for large release frequency (LRF) as well as the results for internal fires and floods indicate. However, there is no discussion or reference about this modeling in Attachment 6A.14.4 where the HVAC system is discussed. Please explain.

19-276

The probability that one of the four pressurizer safety valves (PSVs) fails to reclose (stuck open) after opening for overpressure protection, thus leading to a loss-of-coolant accident (LOCA), is discussed in Appendix 6A.14.8 of Revision 1 of the US-APWR PRA. The failure rate of a stuck open PSV is taken to be 7E-5 per demand based on zero failures in a five year period from 1998 to 2003. This failure rate, which was taken from NUREG/CR-6928 "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," is significantly lower than most of the other failure rates reported in the literature and documented in Table 7.1-1 of the PRA report. For example, the failure rate recommended in the Advanced Light Water Reactor (ALWR) Utility Requirements Document for stuck open PSVs is 5E-3 per demand (which is almost two orders of magnitude higher that the failure rate used in the US-APWR PRA. NUREG/CR-6928 documents a study of the industry-average performance over a five year period including the years from 1998 to 2003. In that study, components from different systems with different operating conditions and maintenance policies were lumped together and actual failures that occurred before 1998 were not included in the database. For example, NUREG/CR-6928 reports in Table 8-1 two events of stuck open safety valves in U.S. pressurized water reactors (PWRs) of which the one that occurred at Fort Calhoun on July 3, 1992 involved a pressurizer code safety valve and resulted in high-pressure safety injection actuation, as discussed in NUREG/CR-5750. Furthermore, the frequency of a stuck open safety valve used in the US-APWR, which contributes to a small LOCA initiating event, is 3E-3/year and is taken from Table 8-1 of

NUREG/CR-6928. This frequency is consistent with a stuck open PSV failure rate of 5E-3 and not 7E-5 as assumed in the US-APWR PRA. Please discuss.

19-277

The following statement is made in Attachment 6A.14.11 of Revision 1 of the US-APWR PRA report: "The [reactor coolant] pump is designed so that the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling." This design feature of the reactor coolant pumps (RCPs) is credited in the PRA and therefore must be listed in the appropriate Chapter 19 section (Table 19.1-115) of the US-APWR design control document (DCD).