REQUEST FOR ADDITIONAL INFORMATION NO. 151-1824 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)

19-285

Please address the following questions regarding the key sources of uncertainty and key assumptions listed in Table 19-1 of Section 19 “Uncertainty Analysis” of the US-APWR PRA report:

(a) The following statement is made: “The failure modes of the advanced accumulators are assumed similar to existing accumulators in the current PWR plants. Advanced accumulators are not significant contributors to CDF.” However, at existing PWR plants, accumulators are needed only to mitigate large LOCAs. In the US-APWR design, check valves 8956A, B, C and D are kept closed for a long time under large differential pressures and are credited to mitigate accidents for which they may be required to open under relatively small differential pressures. In addition, advanced accumulators may not be significant contributors to CDF but they are risk significant equipment with a risk achievement worth (RAW) value of 430. Please discuss.

(b) The following statement is made regarding the “summary results of qualitative assessments” about digital I&C: “Applied requirement or reliability for digital I&C.” Please explain.

(c) The following statement is made regarding the “summary results of qualitative assessments” about success criteria analysis: “Appropriate simplifying evaluations for the US-APWR have been performed.” Please explain.

(d) The following statement is made regarding the “summary results of qualitative assessments” about data analysis: “Potentially valuable generic data sources were collected…..” Please clarify and explain.

(e) The following statement is made regarding the “summary results of qualitative assessments” about CCF of inter-systems not being included in the PRA: “The environment, operation or service conditions, design and maintenance are different between systems.” Please list such differences, for each of the mentioned attributes, regarding the accumulator check valves 8956A, B, C, and D and the high head injection check valves ACC01A, B, C, and D.

19-286

Section C.I.19.2 of RG 1.206 lists several uses of PRA to support design certification. Items A (ii), E and F, discuss the use of PRA to eliminate or reduce known significant risk contributors of existing operating plants and demonstrate that the plant design represents a reduction in risk compared to existing operating plants. Section C.I.19.6 of RG 1.206 (second paragraph) discusses an acceptable approach to the staff for demonstrating that a plant referencing the US-APWR design will represent a reduction in risk compared to existing operating plants (e.g., a qualitative comparison by initiating event category can be performed using the results reported in NUREG-1560 "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant performance.") Please provide a list of the major features that contribute to the reduced core damage frequency (CDF) of the US-APWR design, as compared to operating pressurized water reactor (PWR) designs, for each of the initiating event categories contributing the most to this reduction.

19-287

It is stated In Chapter 25 of the US-APWR PRA: "ANSI/ANS-58.21-2007 "external-events PRA methodology" notes that the fundamental screening-out criteria of other external events are (a) if it meets the criteria in the NRC's 1975 Standard Review Plan (NUREG-75/087)or later version; or (b) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than 1.0E-05 per year and that the conditional core damage probability is less than 1.0E-01, given the occurrence of the design basis event; or (c) if it can be shown using a demonstrably conservative analysis that the CDF is less than 1.0E-06 per year." However, the ANSI/ANS-58.21-2007 is applicable to operating reactors. Applicants for new reactor design certifications, per 10 CFR Part 52, have to demonstrate how the risk associated with the design compares against the Commission's goals of less than 1E-4/year for core damage frequency and less than 1E-6/year for large release frequency (see RG 1.206 C.I.19.2 (C)). Using the ANSI/ANS-58.21-2007 criteria one cannot compare to the Commission's goals and conclude that the total large release frequency (LRF) is less than 1E-6/year. In addition, the staff believes that only those external events that do not contribute significantly to the total CDF and LRF of the plant can be screened out from further evaluation. Therefore, the criteria for screening out external events from the quantitative evaluation should be adjusted so that (a) it can be possible to demonstrate how the Commission's goals are met, and (b) it can be possible to identify significant external events contributors to the total plant risk. Please revise accordingly or discuss, as necessary.

19-288

In Section 19.1.7.6 of the US-APWR design control document (DCD) it is stated: "At the design stage, PRA results have been used as input in the development of the technical specifications (Chapter 16). PRA insights are utilized to develop risk-managed technical specifications (RMTS) and surveillance frequency control program (SFCP)." Please discuss in more detail (e.g., by providing a few examples) how the PRA results were used at the design stage in the development of TS. In addition, for demonstration purposes, please include at least one example of using the US-APWR PRA to apply the RMTS guidance (NEI 06-09). The terms LERF (large early release frequency) and ILERP (incremental large early release probability) in the NRC- approved NEI guidance can be substituted with the terms LRF (large release frequency) and ILRP (incremental large release probability) used in the US-APWR PRA. The selected example(s) should be realistic but "challenge" the process with respect to PRA key assumptions, the use of insights from external events, and the presence of uncertainties. A good example for implementing NEI 06-09 could be the following: While the plant operates at power with one emergency ac power gas turbine generator (GTG), one alternate ac (AAC) GTG, and one turbine-driven (T-D) emergency feedwater (EFW) pump out for preventive maintenance, one of the two motor-driven (M-D) EFW pumps is found to be inoperable. Suppose that neither one of the M-D or the T-D pumps can be returned to service within the required completion time (CT) and NEI 06-09 guidance is used to extend the CT. While the plant is at power in the above described configuration within the extended CT, HVAC is lost to the room where the remaining M-D EFW pump is located. Please provide results showing the ICDP and ILRP values versus time. In your discussion, include (1) specific compensatory risk management actions that may be credited in the calculations, (2) key modeling assumptions that are important to ensure that the RMTS decision-making process is robust, and (3) any important assumptions made in the external events calculations and how it is determined that the PRA models for internal fires and flooding ensure reliable or bounding results consistent with NEI guidance and, thus, suitable for use in the RMTS decision-making process.

19-289

The staff has approved guidance for implementing risk-managed technical specifications (RMTS) and surveillance frequency control program (SFCP) which is applicable to operating reactors. This guidance for operating reactors is documented in NEI 06-09 for RMTS (Initiative 4b) and NEI 04-10 for SFCP (Initiative 5b). In its application for certification of the US-APWR design, Mitsubishi Heavy Industries (MHI) has indicated that the PRA submitted in support of the design certification application satisfies the requirements specified in the NEI 06-09 and NEI 04-10 that are associated with PRA technical adequacy, such as scope of PRA, level of detail to provide plant configuration specific impacts and operating modes, with the exception of site-specific information that will be provided by the COL applicant/holder. Please perform a self-assessment of the US-APWR design certification PRA quality and indicate, in the response to this request for additional information (RAI), how it satisfies the requirements specified in the NEI 06-09 and NEI 04-10 that are associated with PRA technical adequacy. Your response should address the following statement made in NEI 06-09 (Section 4): "The PRA model attributes and technical adequacy requirements for RMTS applications must be compatible with established ASME standards requirements, as modified by NRC Regulatory Guide 1.200 Rev 0…… It is expected that, in general, the PRA which supports RMTS shall meet Capability Category 2 requirements and any exceptions to meeting those requirements shall be justified." For the PRA Level 1 and 2 portions addressing internal events (including internal flooding) at power operation (Modes 1 and 2), please indicate whether and how each ASME "high level" and "supporting" requirement is met with respect to Capability Category II. For those areas that ASME requirements are not fully met, identify what is needed to be done and by whom (e.g., MHI or the COL applicant/holder) so they can meet Capability Category II requirements or justify why such a capability is not necessary for implementing RMTS and SFCP. Also, please discuss assumptions and attributes of the US-APWR internal fires, seismic analysis and other external events, and shutdown PRA models which ensure reliable or bounding results and contribute to the robustness of the RMTS and SFCP decision-making processes.

19-290

Please address the following questions regarding the seismic margins analysis (SMA) discussed in Section 19.1.5.1 “Seismic Risk Evaluation” of the US-APWR design control document (DCD) and in Chapter 24 of Revision 1 of the US-APWR PRA report:

(a) A major assumption of the SMA model is that no credit is taken for nonsafety-related systems (assumption b, listed on page 19.1-64 of the DCD). However, the seismic failure of nonsafety-related systems can have adverse interaction with safety-related systems which otherwise survive the earthquake. In another major assumption (assumption i on page 19.1-65 of the DCD) it is stated: “Seismic spatial interactions between SSCs design[ed] to be seismic Category I and any other buildings will be avoided by proper equipment layout and design.” This statement is an assumption about a design feature that will be demonstrated in the future. Please discuss, or provide reference if it is discussed elsewhere in the DCD, how this assumption will be verified (e.g. through an ITAAC).

(b) A major assumption of the SMA model (assumption j on page 19.1-65 of the DCD) states that “Relay chatter does not occur or does not affect safety functions during and after seismic event.” Please provide the basis of this statement. In Section 24.3.2.1 of Revision 1 of the US-APWR PRA report it is stated: “Electrical equipment ….. could fail due to relay chatter which may trip the circuits or lead to inadvertent change of state. However, solid-state relays that are not prone to chatter are used in the design of the US-APWR. Even if there is a need to use electro-mechanical relays, they are qualified to the seismic response from the SSE with sufficient margin. Therefore, relay chatter is not considered a credible failure mode of electrical equipment in this evaluation.” Please clarify whether electro-mechanical relays are used in the US-APWR design and explain what is meant by “sufficient margin” of electro-mechanical relays which are qualified to the seismic response from the SSE. Also, the above quoted statement makes an “assumption” about a feature of the US-APWR design (i.e., use of solid-state relays that are not prone to chatter) that must be documented in Section 19.1.7 of the DCD (e.g., Table 19.1-115) with proper disposition (e.g., provide cross-reference to other DCD sections or identify specific design certification requirements to ensure that these assumptions will remain valid for the as-to-be-built, as-to-be-operated plant).

(c) A major assumption of the SMA model (assumption f, listed on page 19.1-64 of the DCD) is that “piping will fail prior to failure of associated pressure boundary valves.” Please provide the basis for this assumption and discuss how it will be verified that this assumption will remain valid for the as-to-be-built, as-to-be-operated plant.

(d) A major assumption of the SMA model (assumption g, listed on page 19.1-64 of the DCD) is that “Failure of the RHRS isolation valves is not included in the analysis, because the pipe sections are assumed to fail before the valves fail and these valves are normally closed.” However, the staff notes that the motor-operated containment spray/residual heat removal (CS/RHR) suction isolation valves 9007A, B, C, and D are normally open. Please clarify.

(e) The following statement is made in Section 19.1.5.1 of the DCD: "SSCs of seismic Category I are designed for SSE of 0.3g PGA with such conservatisms that they have high seismic capacity. Therefore, HCLPF of 0.5g PGA would be reasonable achievable for seismic Category I SSCs. This value is assigned for those SSCs at design certification phase. The fragilities of those SSCs will be confirmed that the HCLPFs of the SSCs are greater than 0.5g PGA at the detailed seismic design phase." This statement is an assumption about a design feature that will be demonstrated in the future. Please discuss, or provide reference if it is discussed elsewhere in the DCD, how this assumption will be verified (e.g. through an ITAAC).

(f) Table 24.4-9 of Revision 1 of the US-APWR PRA report provides the dominant mixed cut sets containing random failure probability higher than 1E-3. Please clarify whether random common cause failures (CCFs) were included in the models and provide the basis for the assumed cutoff of 1E-3. The staff notes that the basic event probabilities, reported in the PRA results from internal events at power operation, include the CCF probability of gas turbine generators (GTGs) to run for more than one hour which is 1.1 E-3. However, no mixed cut set including this random failure is reported in the seismic risk analysis. The staff believes that mixed cut sets comprised from the seismic failure of the switchyard ceramic insulators (with HCLPF 0.08g PGA), which leads to loss of offsite power, and random common cause failures of the emergency GTGs to start and run are important mixed cut sets which should be reported and discussed. Please explain.

(g) Dominant mixed cut sets labeled "Combination 2" and "Combination 4" in Section 19.1.5.1 of the DCD, do not appear to be realistic because if there is seismic failure of the turbine-driven EFW pumps (which have a HCLPF value of 0.75g) there should be also seismic failure of the motor-driven EFW pumps (which have a lower HCLPF value, i.e., 0.62g). Please explain.