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March 13, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09086

Subject: MHI's Responses to US-APWR DCD RAI No.151-1824

Reference: 1) "Request for Additional Information No. 151-1824 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: Chapter 19.1," dated January 12, 2009
2) Letter MHI Ref: UAP-HF-09045 from Y. Ogata (MHI) to U.S. NRC, "MHI's Responses to US-APWR DCD RAI No.151-1824," dated February 6, 2009

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No. 151-1824 Revision 1".

Enclosed are the second responses to the RAIs contained within Reference 1. In the initial responses submitted with Reference 2, MHI committed to submit responses to19-285, 19-286 , 19-290 within 60 days after RAI issue date.

Of these RAIs, questions, the following 2 RAIs will not be answered within this package.

19-288
19-289

MHI will need additional analyses and surveys for the responses to these RAIs. The responses to these RAIs will be submitted by 12th April.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:

1. Responses to Request for Additional Information No.151-1824 Revision 1.

D081
NRO

CC: J. A. Ciocco
C. K. Paulson

Contact Information

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Docket No. 52-021
MHI Ref: UAP-HF-09086

Enclosure 1

UAP-HF-09086
Docket Number 52-021

Responses to Request for Additional Information No. 151-1824 Revision 1

March 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.151-1824 REVISION 1

SRP SECTION: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

APPLICATION SECTION: 19.1

DATE OF RAI ISSUE: 1/12/2009

QUESTION NO. : 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.

ANSWER:

(a) Advanced Accumulators

The component types of the check valve used for the advanced accumulators are same with conventional accumulators. Therefore, the reliability of the check valves at the large flow rate mode is same as conventional plants.

The reliability of accumulator injection has large impact on core damage scenarios of large and medium break LOCA events. Following these initiating events, check valves at the accumulator discharge line will open and will be kept open until the accumulator's small flow rate injection function for core re-flood completes. Hence the failures of check valves potentially occur only at the initial stage of LOCA events, where the advanced accumulator operates same as conventional type accumulators. The reliability of check valves is expected to be same as those of conventional accumulators.

The dual flow rate function of the advanced accumulator is provided by its inner structure, which has no active structures. The accumulator tank in kept under atmospheric temperature and boron concentration is maintained below the level precipitation occur. The inner structure of the advanced accumulator is not kept under severe environment, and therefore, failure during standby is unlikely to occur.

In order to assess the uncertainty of the reliability of advanced accumulators, sensitivity study has been performed. In the sensitivity study, failure probabilities of advanced accumulator check valves as well as its common cause failure probabilities were set ten times higher than the generic data. The resulting internal event core damage frequency (CDF) at power was approximately 3% higher than the base case CDF.

(b) Digital I&C

The base case assumes that application software and support software common cause failure to be 1.0E-05/demand and 1.0E-07/demand, respectively. Since these probabilities have high uncertainties, sensitivity analyses (Case 3-4) concerning software common cause software have been performed in the Chapter 18 subsection 18.3.3 of the PRA report (MUAP-07030(R1)). If the probability of software common cause failure that results in failure of all safety related signals modeled in the PRA is assumed as higher than the probability of the application software common cause failure of the base case, the CDF results in approximately 1.5 times higher than the base case.

(c) Success Criteria Analysis

The success criteria which are described in Chapter 5 of the PRA report (MUAP-07030(R1) are determined from the design, engineering judgment and thermal/hydraulic analysis results in a manner that allows a margin to account for the uncertainties in models used for the thermal/hydraulic analyses and grouping of initiating events.

(d) Data Analysis

The procedural steps of data analysis are described in the Chapter 7 subsection 7.1.1 of the PRA report (MUAP-07030(R1).

For each component type and failure mode, the failure rates are selected from the available generic data sources. The following steps are performed to develop the appropriate data set for the US-APWR

- Collect generic failure data sources
- A list of component types, failure modes, failure rates and error factors is compiled for each data source.
- Compare the failure data among the data sources.
- Identify the component types for US-APWR plant
- The most applicable failure modes and failure rates are selected for the US-APWR PRA.

The component boundaries are defined by generic data sources and the boundaries of the basic events are set to be consistent with the component boundaries.

(e) Common cause failure analysis

The treatment of common cause failures are described in the Chapter 6 subsection 6.1.2.3 of the PRA report (MUAP-07030(R1)) and in the responses to the RAI 19-123.

The major differences of the accumulator (ACC) check valves (VLV103A, B, C, and D) and the high head injection (HHI) check valves (VLV013A, B, C, and D) that impact the possibility of common cause failure are the following:

- Design
  Design features and sizes are different.
    Accumulator Check Valves: Accumulator check valves set on the piping of 14 inches diameter.
    HHI Check Valves: HHI check valves set on the piping of four inches diameter.

- Environment
  Environment practices during normal plant operation are different.
    Accumulator Check Valves
      Accumulator side: 700 psig, 300F (Accumulator design Pressure & Temperature)
      RCS cold leg side: 2296 psig, 553 F (Cold leg Pressure & Temperature)
    HHI Check Valves
      Safety injection pump side: Atmospheric pressure and temperature
      Reactor Vessel side: 2296 psig, 553 F (Cold leg Pressure & Temperature)

Taking into consideration of the differences listed above, it is judged that common cause failures between the check valves in the accumulators and the high head injection are unlikely to occur.

Impact on DCD

No impact on DCD.

Impact on COLA

No impact on COLA.

Impact on PRA

Incorporate these responses into the next revision of the PRA report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.151-1824 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/12/2009**

QUESTION NO. : 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.

ANSWER:

Major features that contribute to the reduced CDF of the US-APWR design are listed below for each initiating event categories.

(1) Loss of coolant accident (LOCA) events

- Advanced accumulators provide integrated function of low head injection system in the event of LOCA. This feature reduces the risk of failing to inject coolant during the initial stage of LOCA events
- The in-containment refueling water storage pit (RWSP) eliminates the need for recirculation switching. This feature eliminates the risk of failing to switchover to recirculation mode.

- Safety injection system consists of four independent trains. This feature enhances redundancy of the safety injection and improves the reliability core injection function.

- Direct vessel injection is adopted for safety injection from safety injection pumps. This feature reduces the risk of degrading safety injection system during large LOCA events.

- Alternate containment cooling utilizing the containment fan cooler unit can be established when containment spray system has failed. Alternate containment cooling provides diversity in containment cooling function and improves reliability of containment cooling.

- Alternate core injection utilizing the containment spray system / residual heat removal system can be established when safety injection has failed. Alternate core cooling provides diversity in core injection function and improves reliability of core injection during medium and small LOCA events.

(2) Steam generator tube rupture

- Feed water to SG with high water level is automatically isolated by the feed water isolation valve. This function prevents fill up of the faulted SG.

- Alternate containment cooling utilizing the containment fan cooler unit can be established when containment spray function has failed. The alternate containment cooling provides diversity in containment cooling function and improves reliability of containment cooling required after postulated bleed and feed operation.

(3) Inter-systems LOCA

- Piping design pressure for the residual heat removal system is upgraded and a return path to the RWSP is adapted in case of inter-system LOCA. This feature reduces the risk from inter-systems LOCA.

(4) Steam line breaks and Feed water line break

- Emergency feed water system consists of four independent trains. This feature enhances redundancy of the feed water and improves reliability of feed water function after the event of secondary side piping break.

- Alternate containment cooling utilizing the containment fan cooler unit can be established when containment spray function has failed. The alternate containment cooling provides diversity in containment cooling function and improves reliability of containment cooling required after postulated bleed and feed operation.

(5) General transients and Loss of feed water

- Emergency feed water system consists of four independent trains. This feature enhances redundancy of the feed water and improves the reliability of SG cooling.

- Alternate containment cooling utilizing the containment fan cooler unit can be established when containment spray function has failed. The alternate containment cooling provides diversity in containment cooling and hence improves reliability of containment cooling required after postulated bleed and feed operation.

(6) Loss of component cooling water

- Component cooling water (CCW) system is separated into two subsystems which has no interconnections. This design feature reduces dependencies between each train and reduces the risk of loss of CCW.

- Non-essential chilled water system and the fire suppression system can provide Alternative component cooling water to the charging pump cooling line. This feature enables the charging pump to provide reactor coolant pump (RCP) seal injection during the event of loss of CCW and prevents occurrence of RCP seal LOCA.

(7) Loss of offsite power, Loss of ac bus and Loss of dc bus

- Emergency power system consists of four independent trains with Class 1E gas turbine generators on each train. This feature enhances redundancy of the electrical system and hence improves reliability of emergency power during loss of offsite power.

- Two alternate ac (AAC) power sources diverse from Class 1E gas turbine generators can provide ac power to the Class 1E bus in the event of station blackout. This feature enhances diversity in power sources available during offsite power and reduces the risk from SBO events.

- Two turbine driven emergency feed water pumps provide feed water to the SG in the event of loss of ac power. This feature enhances the diversity of feed water function from ac power and reduces risk from loss of offsite power events.

(8) Anticipated transient without scram (ATWS)

- Diverse actuation system is installed as a counter-measure against common cause failures in software of safety I&C. This feature improves diversity in reactor protection function and reduces the risk of ATWS.

<u>Impact on DCD</u>

No impact on DCD.

<u>Impact on COLA</u>

No impact on COLA.

<u>Impact on PRA</u>

No impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.151-1824 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/12/2009**

QUESTION NO. : 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 non-safety-related systems (assumption b, listed on page 19.1-64 of the DCD). However, the seismic failure of non safety-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 1.1E-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.

ANSWER:

(a) In DCD section 3.2, it is stated that non safety-related SSC are designed or located to avoid adverse interaction with safety-related SSC.

(b) As noted in the page 19.1-65 of the DCD Revision 1, the assumption "j. Relay chatters does not occur or does not affect safety functions during and after seismic event " is one of the key assumption. This assumption will be added in the DCD section 19.1.7.1, Table 19.1-115. The descriptions in the PRA report are the general requirements for the relays to prevent the functional failures due to seismic. The specific fragilities of the relays will be confirmed from the fragility analysis using the plant specific design features through the update of PRA by the fuel loading.

(c) This assumption is about the structural failure of valves. Also this assumption is based on the following consideration.

Valves itself is normally built as very robust steel structure and very high acceleration is needed to cause structural failures of valves generally. If the piping systems are overtaken by such high acceleration, piping support structures or connection parts between valve and piping are thought to be damaged prior to the failure of valve body. This will be confirmed by the fragility analysis using the plant specific design features through the update of PRA by the fuel loading.

(d) The motor-operated CS/RHR suction isolation valves 9007A, B, C, and D are normally open and they are not required to change position. Therefore, their seismic induced failures are not included in the SMA model on the basis of assumption f. on page 19.1-64 of the DCD. Therefore, valves that are not required to change positions are not included in the analysis.

(e) This assumption will be confirmed by the fragility analysis using the plant specific design features through the update of PRA by the fuel loading. This will be added in the DCD section 19.1.7.1, Table 19.1-115 from the responses to the RAI 19-207.

(f) The random failure CCFs are included in the SMA model. The CCFs of GTGs failure are provided in Chapter 24 of Revision 1 of the US-APWR PRA report (Attachment 24C.4, minimal cutset number 527, basic event ID: EPSCF4DLLRDG-ALL, which is 9.9E-4).

Cutoff value of 1E-3 is assumed from the features of HCLPF. HCLPF is defined as the High Confidence (95%) of Low Probability of Failure (5%) capacity and it has the potential failure probability of about 5E-2. Cutoff value of 1E-3 is one order of magnitude lower than 5E-2.

Although the seismic failure of the switchyard ceramic insulators (HCLPF is 0.08g PGA) and random failure CCFs of GTGs to start and run have been involved in the model, the probabilities of the cutsets are lower than the cut off value of 1E-3. This is described implicitly in the third identified risk insight on page 19.1-71 of the DCD revision 1.

(g) The dominant mixed cut sets in the section 19.1.5.1 of the DCD describe the combination of the seismic failure of SSC and the random failure of SSC only. The combination of the seismic induced failures of SSCs such as the motor-driven EFW pumps and the turbine-driven pumps are also considered in the SMA process using "min-max" method.

Impact on DCD

The assumptions on relay chatters will be involved in the Table 19.1-115 of DCD next revision as described in the responses to the RAI 19-207.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.