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May 14, 2009

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

Attention: Mr. Jeffery A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09237

Subject: MHI's Response to US-APWR DCD RAI No. 296-2254 Revision 0

References: 1) "Request for Additional Information No. 296-2254 Revision 0, SRP Section: 03.09.01 – Special Topics for Mechanical Components, Application Section: Section 3.9.1," dated 4/1/2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No. 296-2254 Revision 0."

Enclosed are the responses to all questions of the RAI (Reference 1).

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

Sincerely,

Y. Ogata

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

Enclosures:

1. Response to Request for Additional Information No. 296-2254, Revision 0

CC: J. A. Ciocco
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NRO

Contact Information

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Enclosure 1

UAP-HF-09237
Docket No. 52-021

Response to Request for Additional Information No. 296-2254, Revision 0

May, 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/14/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 296-2254 REVISION 0**
SRP SECTION:	**03.09.01 – Special Topics for Mechanical Components**
APPLICATION SECTION:	**03.09.01**
DATE OF RAI ISSUE:	**4/1/2009**

QUESTION NO.: 03.09.01-1

1. The DCD in section 3.9.1.1.1.2 provides a summary of plant heat-up limits. The DCD section states, "Plant heat-up operations are conservatively represented by uniform ramp temperature changes of 50°F per hour which bounds the heat-up rate resulting from operation of four RCPs." GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified. Standard design practices for maximum design heat-up rates have typically used a value of 100°F per hour. The maximum normal expected heat-up rate is then on the order of 50°F per hour; thus providing a margin of 2 to the maximum design limit. Provide additional information and clarify the margins between the maximum design heat-up rate and the expected maximum normal heat-up rate.

2. The DCD in section 3.9.1.1.1.2 states, "Plant heat-up and cooldown operations is assumed to each occur 120 times during the plant design life." GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified. Previously accepted practice for a standard four-loop PWR uses 5 heat-ups and cooldowns per year for a total of 300 cycles for a 60-year design life. Provide additional information and justification for assuming 120 cycles for a 60-year design.

3. The DCD in sections 3.9.1.1.1.4, 3.9.1.1.1.5, and 3.9.1.1.1.6 discuss ramp load increases and decreases between specified power levels and step load increases and decreases. The numbers of occurrences are also described in DCD table 3.9-1, "RCS Design Transients." GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified.
The following issues were identified:

In Section 3.9.1.1.1.4, the basis for selecting 600 occurrences for the ramp load increase and decrease at five percent of full power per minute is not clear. Previous accepted practice for a standard four-loop PWR uses a greater number of occurrences. Provide additional information and justification for assuming 600 cycles.

In Section 3.9.1.1.1.5, the basis for selecting 600 occurrences for the step load increase and decrease of 10 percent of full power per minute is not clear. Previous accepted practice for a standard four-loop PWR uses a greater number of occurrences, typically 50 per year. Provide additional information and justification for assuming 600 cycles.

In Section 3.9.1.1.1.6, the basis for selecting 60 occurrences for the large step load decrease with turbine bypass is not clear. Previous accepted practice for a standard four-loop PWR uses a greater number of occurrences, typically 5 per year. Provide additional information and justification for assuming 60 cycles.

4. DCD section 3.9.1.1.1.7 address steady-state fluctuations and load regulation. GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified. Previous accepted practice for a standard four-loop PWR defines the magnitude of these transients including temperature and pressure variations and duration. The description of these fluctuations is not complete in the DCD. Provide additional information and justification for these steady-state fluctuations.

5. DCD section 3.9.1.1.1.9 addresses main feedwater cycling. DCD section 3.9.1.1.2.11 addresses emergency feedwater cycling. NRC experience (NRC Bulletin 88-08 and its supplements indicates that during low feedwater flow stratification flow conditions can result in significant differences in thermal fatigue cycles that have resulted in failures of the feedwater piping pressure boundary in PWR design similar to the APWR. GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. Has this issue been addressed in the design and operation of the feedwater systems? Provide additional information on the basis for the number of cycles assumed for the main and emergency feedwater systems.

6. DCD section 3.9.1.1.1.10 addresses core lifetime extension. GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified. The use of a decreased RCS average temperature with turbine inlet valve adjustments to extend the life of the core is a new transient that has not previously been approved. Provide additional information and justification for this transient including impacts on core design and performance.

7. DCD section 3.9.1.1.1.13 primary-side leakage test addresses performance of a primary side leak test with system pressure raised to 2500 psia. The ASME boiler and pressure vessel Code no longer requires increasing pressure above the normal operating pressure to perform these leakage tests. SRP section III.1 states that the review should verify that acceptable Code limits be specified. Provide additional information and justification for raising the pressure to perform these leakage tests.

8. DCD section 3.9.1.1.1.14 secondary-side leakage test addresses performance of a secondary-side leak test with secondary system pressure raised to design pressure. GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified. Previous accepted practice for a standard four-loop PWR defines the pressure and temperatures for both the secondary-side and the primary-side to prevent damage to the steam generators and the RCS. Provide additional information and justify the pressures and temperatures for both the secondary and primary sides.

9. DCD section 3.9.1.1.2.3 addresses reactor trips from full power. GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. SRP 3.9.1 section III.1 states that any deviation from previous accepted practice be justified. Previous accepted practice for a standard four-loop PWR uses a greater number of occurrences, typically 400 occurrences. Provide additional information and justification for assuming 100 reactor trips.

10. For selected service level B conditions described in the DCD sections listed, occurrence numbers on a yearly basis are less than previous accepted practice. The reduction may be

justified based service experience however the amount of margin appears to be less than typical four-loop PWRs currently operating. GDC 14 and 15 apply as components important-to-safety are designed to postulated transients anticipated during the design life of the plant. For each service level B event listed below provide additional justification and information for the reduced number of occurrences assumed in the design basis.

DCD section 3.9.1.1.2.4 Control Rod Drop: Assumes 30 times.
DCD section 3.9.1.1.2.6 Inadvertent Safeguards Actuation: Assumes 30 times.
DCD section 3.9.1.1.2.7 Partial Loss of Reactor Coolant Flow: - Previous plant design bases have defined a partial loss of flow as the loss of a single pump at 2 times per year. Provide additional information and justification for why a partial loss of flow assumes loss of two pumps to occur once every 2 years (30 times during the plant design life).

11. For DCD section 3.9.1.1.3 Service Level C Conditions (emergency conditions) additional information is requested to confirm compliance with GDC 14 and 15 requirements for the following events:

DCD section 3.9.1.1.3.1 Small LOCA: Provide additional information on break size definitions and clarify description of the event used in the design basis.
DCD section 3.9.1.1.3.2 Small Steam Line Break: Provide additional information on break size definitions and clarify description of the event used in the design basis.
DCD section 3.9.1.1.3.4 Small Feedwater Line Break: For previous four-loop PWRs no distinction was made in previous designs between a small or large feedwater line break (Level D). The Level C small feedwater line break description states, "No main feedwater is supplied to either of SG and then all of SG water level decrease. For receipt on low SG water level signal, the reactor is tripped and emergency feedwater pumps are actuated automatically." Provide additional information and clarify as there does not appear to be a significant distinction between the Level C description and the Level D description.
DCD section 3.9.1.1.3.5 Steam Generator Tube Ruptures: Previous plant designs assumed steam generator tube ruptures to be Level D faulted events not Level C emergency events. Provide additional information and justification for why the steam generator tube rupture is included as a Level C event verses a Level D event.

12. Appendix S to 10CFR50 specifies that applicants include seismic events in the design basis. The applicant is requested to provide the basis to justify that the earthquakes dynamic events at the rated operating power conditions are not included in Table 3.9-1.

13. NRC Bulletin 79-13 addressed the fatigue loading due to thermal stratification and high cycle thermal striping during low flow emergency feedwater injection. Bulletin 88-08 and its supplements indicates that during low feedwater flow stratification flow conditions can result in significant differences in thermal fatigue cycles that have resulted in failures of the feedwater piping pressure boundary in PWR. Bulletin 88-11 requires consideration of the effects of thermal stratification on the pressurizer surge line dynamic loads. Discuss the basis for not considering the thermal stratification in Section 3.9.1.1, which is an important design transient in design of piping.

14. In response to the recent industrial experience on the vibration effects of the components and piping due to Acoustic Resonance, SRP 3.9.5 and Regulatory Guide 1.20, were revised to include the acoustic loads into consideration. MHI is requested to provide the basis for not including this acoustic cyclic loading in the design transients for EPR DCD.

ANSWER:

1. The US-APWR employs a Pressurizer Water Solid Mode of operation during RCS low pressure heatup and cooldown. The four RCPs and Pressurizer heaters provide the heat inputs for plant heat-up. The maximum heat-up rate expected to occur at RCS low

temperature is estimated at approximately 40°F/h, which is be bounded by 50°F/h. Estimated conditions are as follows:
- Four RCPs and Pressurizer heaters are operating
- Heat loss of components are assumed to be zero
- Heat loss caused by letdown is assumed to be zero

RCS heat-up rate decreases at higher RCS temperature since heat loss increases. Operating data from PWRs that employ a Water Solid Mode of operation demonstrate that actual heat-up rate is bounded by 50°F/h.

2. The US-APWR shutdown design transient frequency is two cooldowns per year, one planned and one unplanned. This is the same for heat-up. This estimate reflects the following considerations:
   - The US-APWR utilizes a 24-month fuel cycle, which means the actual planned shutdown will be 0.5 heat-up and cooldown per year
   - Based on current PWR operating data, plant heat-up and cooldown frequency rarely exceeds two per year

3. Ramp load increase and decrease. The number of ramp load increase is the sum of the number of following events which lead to hot standby conditions or low reactor power and require an increase in power to reach normal operating condition:

Transient Events	**Number**
Plant heat-up	120
Large step load decrease with turbine bypass	60
Loss of load	60
Loss of offsite power	60
Reactor Trip from full power	
- With no inadvertent cooldown	60
- With cooldown and no safety injection	30
- With cooldown and safety injection	10
Control rod drop	30
Inadvertent safeguards actuation	30
Partial loss of reactor coolant flow	30
Inadvertent RCS depressurization, umbrella case	30
Partial loss of emergency feedwater	30
Sum	550

The number of ramp load increases is 600, including margin. The number of ramp load decrease is 600, based on the number of ramp load increases. Based on data from operating PWRs, this number of occurrences is conservative.

Large step load decrease with turbine bypass and 10% step load increase and decrease of 10% of full power: MHI assumed the frequency of a large step load decrease, due to an electrical disturbance with turbine bypass as one per year. As for step load increases and decreases of 10 percent of full power, MHI assumed 10/year considering the frequency of these events experienced in operating PWR (10 times larger than large step load decrease).

Transient Events	Frequency	Number	Notes
Large step load decrease with turbine bypass	1/year	60	Assumed as same frequency of electrical disturbance
Step load increase and decrease of 10 percent of full power	10/year	600	Assumed 10 times larger than large step load decrease with turbine bypass

Based on operating data of current PWRs, these numbers are conservative.

4. Steady-state temperature and pressure variations and duration of these transients are shown below;

	Steady-State Fluctuations	Load Regulation
Temperature Variation	±3.1 F for T_{avg}	±4 F for T_{hot} ±6 F for T_{cold}
Pressure Variation	±50 psi	±50 psi
Duration	60 sec	2500 sec

Based on operating plant data that shows few steady-state fluctuations at normal operation, the transient condition noted above is expected to be conservative.

For load regulation, the transient condition is determined conservatively based on load regulation analysis.

5. Since the US-APWR Steam Generator design employs an elevated feedwater ring, thermal stratification is assumed to occur at the level of feedwater ring. So the feedwater nozzle and piping pressure boundary which is lower than feedwater ring are not assumed to experience thermal stratification.
The main feedwater cycling is assumed to occur at hot standby or no-load condition. The basis of the number of main feedwater cycling is the sum of the required cyclic main feedwater injection of following events.

Transient Events	Number of main feedwater injection
Plant heat-up and cooldown	480
Hot stand-by	1350
Hot functional test	200
Sum	2030

The number of main feedwater cycling is 2100, including margin. Based on data from operating PWRs, this number of occurrences is conservative.

The emergency feedwater cycling is assumed to occur after Reactor Trip events. The basis of the number of emergency feedwater cycling is the sum of the number of following events which requires emergency feedwater injection.

Transient Events	Number of emergency feedwater injection
Loss of load	60
Loss of offsite power	360
Reactor Trip from full power	
- With no inadvertent cooldown	60
- With cooldown and no safety injection	30
- With cooldown and safety injection	10
Control rod drop	30
Inadvertent safeguards actuation	30
Partial loss of reactor coolant flow	30
Inadvertent RCS depressurization, umbrella case	30
Sum	640

The number of emergency feedwater cycling is 700, including margin. Based on data from operating PWRs, this number of occurrences is conservative.

6. The US-APWR is not performing core lifetime extension evaluations at this time. Such evaluations will be performed in the future by the respective Licensees as part of the license renewal process in accordance with 10 CFR Part 54. This event is included among the US-APWR design transients to confirm that the stress evaluation is acceptable when core lifetime extension evaluations are conducted in the future.

   In this transient, MHI assumes 2 weeks as a maximum core lifetime extension. The required temperature decrease to achieve criticality is determined conservatively based on the analysis.

7. MHI recognizes that it is not necessary to raise RCS pressure up to the design pressure 2500 psia during a primary-side leakage test according to ASME boiler and pressure vessel Code Section XI IWB-5221. However, uses the design pressure for the primary-side leakage test transient to assure that a conservative stress evaluation is performed. This is conducted below safety valve settings. Exposure to these conditions will not damage the RCS.

8. MHI uses atmospheric pressure for the primary-side with the secondary-side pressure just below the design pressure so as to not actuate the Main Steam Safety Valves in the secondary-side leakage test transient.

   The actual RCS temperature used during the test is atmospheric temperature, which is greater than the nil ductility temperature. These conditions were selected to assure that the stress evaluations are conservative. Exposure to these conditions will not damage either the Steam Generators or the RCS.

9. US-APWR defines 3 cases of reactor trip. Each case assumes a different frequency based upon the severity of the postulated event. These cases are:

Transient Case RT From Full Power	Frequency	Number Of Occurrence	Notes
With no inadvertent cooldown	1/year	60	Assumed as the same frequency of electrical disturbances.
With cooldown and no safety injection	0.5/year	30	Assumed as the same frequency as a single failure.
With cooldown and safety injection	10/plant life	10	Assumed as the same frequency as events more severe than single failure or operating error.

Based upon PWR operating experience, these numbers of occurrences assumed by MHI is conservative.

10. Control Rod Drop, Inadvertent Safeguards Actuation and Partial Loss of Reactor Coolant Flow are assumed to occur at the same frequency as a single failure. The frequencies of these postulated events are conservative relative to ANSI/ANS 51.1-1983.

    In DCD Subsection 3.9.1.1.2.7, "Partial Loss of Reactor Coolant Flow," MHI assumes that the loss of two pumps occurs. This assumption is consistent with those postulated in the DCD Subsection 15.3.1.1 safety analysis. The partial loss of flow analysis conservatively assumes that two pumps trip at the same time. This is an extremely conservative assumption since the US-APWR is configured such that each RCP has its own source of electrical power.

11. A small break LOCA is considered to be a break of a 1-inch ID branch pipe of a reactor coolant pipe (a break with an ID smaller than 0.375-inch can be handled by the normal makeup system, which has sufficient capacity to compensate for the coolant loss with this break size, and produces no significant transient). For a break of a 1-inch ID branch pipe, the high head injection system (HHIS) is actuated to inject water at ambient temperature into the reactor coolant system (RCS).

    A small steam line break is considered to be a break equivalent to a 10-inch ID pipe of a main steam relief valve. The increase in steam generation rate caused by the postulated break removes heat from the RCS, which in turn, lowers the temperature and pressure of the RCS.

    A small feedwater line break is considered to be a break of the main feedwater cleanup line. Since main feedwater is not being supplied to any of the SGs under this scenario the water level decreases in all SGs. Upon receipt of a low SG water level signal, the reactor is tripped and the emergency feedwater pumps are actuated automatically. This transient is basically the same as a large feedwater line break (Level D) except for the break size.

    In ANSI/ANS-51.1, steam generator tube rupture is classified as a plant condition 3 event which is equivalent to severity of level B or C events in US-APWR design transient. MHI classified this event as level C, based on severity.

12. Since seismic events are evaluated in the load calculation, they are not considered in design transient. This is described in DCD Subsection 3.9.3.1.1.

13. For feedwater piping, since US-APWR employs elevated feedwater ring, thermal stratification is not assumed to occur at feedwater piping pressure boundary.

    For the pressurizer surge line, thermal stratification is assumed to occur in horizontal part of pressurizer surge line and its load is considered in the stress evaluation. This is described in DCD Subsection 3.12.5.10.

14. In the design of the US-APWR RCS components, piping and the flow conditions are similar to that of the existing and currently operating PWRs in the United States and around the world. Based on an extensive record of vibration-free operation, MHI concludes that acoustic loadings are small enough and that it is not necessary to consider this loading in the stress evaluation.

Impact on DCD

See Attachment 1 for the markup of DCD Section 3.9, Revision 2, with the following changes.

- Change the first two sentences of Subsection 3.9.1.1.3.2, to read as follows: "A small steam line break is considered as a break equivalent to a main steam relief valve pipe break. This transient is assumed to occur five times during the plant design life.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/14/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 296-2254 REVISION 0**
SRP SECTION:	**03.09.01 – Special Topics for Mechanical Components**
APPLICATION SECTION:	**03.09.01**
DATE OF RAI ISSUE:	**4/1/2009**

QUESTION NO.: 03.09.01-2

1. The DCD in section 3.9.1.2.1 provides a list of computer programs used for analysis. An additional list of computer programs is provided in DCD section 3.12.4.1.1. To meet the requirements of 10 CFR 50 Appendix B and GDC 1, a listing of the computer programs used should be provided. In addition, the review procedures of SRP 3.9.1 section III.2.B. states, "The submitted computer test problem solutions recommended in subsection II.2.C of this SRP section are reviewed and compared to the test solutions. Satisfactory agreement of computer and test solutions, usually within +/- 5% percent error band, verifies the quality and adequacy of the computer programs for the functions for which they were designed." There were no computer test problem solutions or summary tables provided in the DCD documentation. Provide additional information and details on the computer test problem methods, solution sets, and summary of the results.

2. 10 CFR Part 50, Appendix B requires provisions to assure that appropriate standards are specified and included in design documents including design methods and computer programs for the design and analysis of Seismic Category I, ASME Code Class 1, 2, 3, components and core support structures and non-Code structures. Mitsubishi is requested to confirm that the computer programs used for US-APWR design and listed in DCD Section 3.9.1.2.1 and Section 3.12.4.1.1 Computer Codes including the preprocessors and the post-processors used for the analyses, are in compliance with requirements of Appendix B to 10CFR50 and ASME NQA-1. Confirm that the documentation of these computer programs is available for staff review. The information should include the author, source code, dated version, and facility; the program users manual and theoretical description, the extent and limitation of the program application; and the benchmarking problems, the QA control and maintenance of the program.

3. The staff has requested that the applicant verify that all computer programs used for calculating stresses and cumulative usage factors for Class 1, 2, and 3 components include staff endorsed environmental effects on the fatigue curves. MHI is requested to identify the computer programs which were used to perform the fatigue analysis. Confirm that the analyses for ASME Section III Class 1 components and piping for the fatigue evaluation include environmental effects in accordance with Regulatory Guide 1.207.

4. The staff has requested that the applicant verify that all computer programs used for calculating stresses for Class 1, 2, and 3 piping include staff endorsed method when performing response spectrum analysis. MHI is requested to verify that all computer programs used for US-

APWR design of piping that use the Independent Support Motion Response Spectrum analysis method comply with the staff position for combining mode, group (absolute sum) and direction responses, as stated in NUREG-1061, Volume 4.

ANSWER:

1. & 2. MHI had verified the computer programs listed in US-APWR DCD Subsection 3.9.1.2.1 in accordance with the requirement methods of SRP II.2.C. MHI prepared a verification report of computer programs that described computer test method, assumption, analysis model, and solution set. MHI also prepared computer code documents that described author, source code, dated version, user's manual, and theoretical description. These documentations have been prepared in accordance with 10 CFR 50, Appendix B and ASME Code, NQA-1 requirements, and will be available for review during the NRC design audit.

3. The computer programs which were used to perform the stress analysis and cumulative fatigue usage factor are verified as noted in the above answers to RAI 03.09.01-02 (2-1) and (2-2). Analysis for ASME Section III, Class 1 components and piping will be performed for fatigue evaluation to include environmental effects in accordance with RG 1.207.

4. As noted in the answer to question 3.12-5 of RAI 260-2023, DCD Subsection 3.12.3.3 will be revised to incorporate NUREG-1061, Volume 4, Section 2 so that group responses are combined by absolute summation method, and inter-modal and inter-spatial responses are combined by the SRSS method. DCD Subsection 3.12.3.2.6, Seismic Anchor Motions (SAMs) will also be revised as noted in the answer to question 3.12-6 of RAI 260-2023 to address the analysis of SAM associated with the ISM method.

   Therefore, all computer programs used for US-APWR design of piping that use the independent support motion response spectrum analysis method comply with the staff position for combining mode, group (absolute sum), and direction responses; as stated in NUREG-1061, Volume 4.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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SRP SECTION:	**03.09.01 – Special Topics for Mechanical Components**
APPLICATION SECTION:	**03.09.01**
DATE OF RAI ISSUE:	**4/1/2009**

QUESTION NO.: 03.09.01-3

DCD Section 3.9.1.3, "Experimental Stress Analysis," indicates that experimental stress analysis is not used by the US-APWR to evaluate stresses for Seismic Category I components and supports. The applicant is requested to discuss the stress analysis methods used to verify the design adequacy in the design of US-APWR components consisting of piping seismic snubbers, pipe whip restraints, and the control rod drive mechanisms.

ANSWER:

Control Rod Drive Mechanism: Experimental stress analysis methods are not used for the design of the Control Rod Drive Mechanism (CRDM). The structural integrity of the CRDM pressure housing, as a RCS pressure boundary, is confirmed by stress analysis in accordance with ASME Code, Section III, Subsection NB. The results of the stress analysis have been submitted to NRC and are contained in Technical Report, MUAP-09009-P, "Summary of Stress Analysis Results for the US-APWR Control Rod Drive Mechanism."

Pipe Whip Restraints: Experimental stress analysis methods are not used for design of pipe whip restraints for the US-APWR. Analytical methods are used in the design of pipe whip restraints as discussed in Subsection 3.6.2.4.4.1 of the DCD.

Piping Seismic Snubbers: There is no plan to use experimental stress analysis methods to verify the design adequacy of snubbers used for US-APWR piping. Snubbers used as shock arrestors and seismic restraints for piping are design verified using loads from a computer dynamic piping stress analysis. The snubber manufacturer determines the conditions and the limits of use for the snubber, and employs, among others, tests as required to establish those limits. These design limits consist of the four loading conditions as established by the applicable ASME Section III, Subsection NF Code (i.e., normal, upset, emergency and faulted, maximum environmental temperature, maximum travel range, maximum allowable angularity, envelope space and any other applicable limitations). This information is listed in the Load Capacity Data (LCD) or Certified Design Report Summary (CDRS) sheet issued and Professional Engineer (PE) stamped by the manufacturer. Based on the results of the Piping Stress Analysis, the loads applied by the piping on the snubber will be categorized in the four loading conditions described above and they are compared with the corresponding allowable loads listed on the LCD or CDRS sheet to ensure that the actual loads are less than the allowables. In addition, the thermal displacements of the pipe at the snubber location, will be considered, to ensure that the maximum allowable travel range and the maximum angularity of the snubber, are not exceeded.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification
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APPLICATION SECTION:	**03.09.01**
DATE OF RAI ISSUE:	**4/1/2009**

QUESTION NO.: 03.09.01-4

1. DCD section 3.9.3.4.5, "Special Engineered Pipe Supports" addresses the application of Appendix F of ASME Code, Section III, Division 1 to ASME Code, Section III piping supports and those supports or components not built to ASME Code Section III. To meet the requirements of GDC 1, 14 and 15 when Service Level D limits are specified the methods of analysis should conform to the methods outlined in Appendix F to ASME Code Section III, Division I. The second bullet of DCD section 3.9.3.4.5 indicates that when the effects of Level D service conditions are evaluated for supports or components not built to ASME Code Section III that the allowable stress levels are based on tests or accepted industry standards "comparable" to those in Appendix F of ASME Code Section III. Provide additional information and details on the methods and allowable stress levels that will be applied for these Level D analyses to allow the staff to confirm that the methods satisfy Appendix F requirements.

2. In DCD Tier 2, Section 3.9.3, the applicant stated that all Seismic Category I equipment are evaluated for the faulted (ASME Section III Service Level D) loading conditions identified in Tables 3.9-3 and 3.9.4. The staff requested that for each of the components, supports, core support structures and RPV vessel listed in Section 3.9.3, MHI identify the computer programs that were used to evaluate the stresses for determining that the ASME Section III, Appendix F, limits were met.

3. In Section 3.9.3, the applicant indicated that calculation methods were used to evaluate RCS components and their supports for faulted loading as detailed in Table 3.9-6 and Section 3.12. The applicant is requested to identify the components evaluated in Section 3.9.3 where the inelastic Service Level D limits were met under the faulted condition loads and load combinations in Tables 3.9-3 and 3.9-4.

ANSWER:

1. MHI has determined that special engineered pipe supports will not be used in the US-APWR design and will revise DCD Subsection 3.9.3.4.5 to delete the related information. See the response to RAI 209-1803, question 18 (a)-(d).

2. Seismic category I equipment are listed in US-APWR DCD Section 3.2, Table 3.2-2. The computer program is used for elastic analysis for stress evaluation of the components, supports and core support structures, and stress limit applied from ASME Code, Section III, Appendix F requirements on Service Level D.

3. MHI will perform plastic analysis in accordance with the requirements of ASME Code, Section III, Appendix III. Load combinations in DCD Table 3.9-3 and Table 3.9-4 do not have special consideration for Service Level D limits. MHI will perform the evaluation for the set of design load combinations.

Impact on DCD

See the "Impact on DCD" for RAI 209-1803, question 3.9.3-18, change to Subsection 3.9.3.4.5.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/14/2009

US-APWR Design Certification
Mitsubishi Heavy Industries
Docket No. 52-021

RAI NO.:	**NO. 296-2254 REVISION 0**
SRP SECTION:	**03.09.01 – Special Topics for Mechanical Components**
APPLICATION SECTION:	**03.09.01**
DATE OF RAI ISSUE:	**4/1/2009**

QUESTION NO.: 03.09.01-5

(1) The staff reviewed MHI technical reports MUAP-09001-P (R0) "Summary of Design Transient" and MUAP-09002-P (R0) "Summary of Seismic and Accident Load Conditions for Primary Components and Piping," that provide a list of new computer codes (MARVEL-M, M-RELAP-5, WCOBRA/RTRAC, TWINKLE-M, VIPRE-01M and GOTHIC) that are not included in DCD section 3.9.1.2.1 OR 3.12.4.1.1. Mitsubishi is requested to provide additional information regarding how the computer codes were used, and discuss computer test problem methods, solution sets, and summary of the results in compliance with requirements of Appendix B to 10CFR50 and ASME NQA-1. Confirm that the documentation of these computer programs is available for staff review. The information should include the author, source code, dated version, and facility; the program users manual and theoretical description, the extent and limitation of the program application; and the benchmarking problems, the QA control and maintenance of the program.

(2) In Table 3.9-1 of DCD revision 1, the number of design cycles for refueling is listed at 120 cycles. Document MUAP-09001-P (R0) "Summary of Design Transient" in Table 2.2-1 lists the number of design cycles for refueling at 60 cycles. The staff request that the applicant clarify which table has the correct number of design cycles for refueling and correct the inconsistency between the two documents.

ANSWER:

(1) MHI uses several computer programs for both design transients for primary components and safety analyses. The specific computer codes mentioned in this RAI (MARVEL-M, M-RELAP-5, WCOBRA/TRAC, TWINKLE-M, VIPRE-01M) are all examples of safety analysis codes that are also used for design transients. A significant amount of information regarding these codes has already been submitted to the NRC in support of the Chapter 15 (accident analysis) review. This information is presented in Table 03.09.01-5.1 below.

The asymmetric pressurization analysis for the accident load evaluation is performed by the GOTHIC code that is also used for containment pressure and temperature analysis in Chapter 6 (Containment Functional Design).

The NRC review guidance for accident analysis codes (SRP 15.0.2) is very similar to the documentation requested in this RAI for codes used to support Chapter 3. This review guidance includes acceptance criteria for documentation and quality assurance. The MHI Quality Assurance Manual, Computer Software Control Procedure (PQF-HD-18041-024, Rev. 2), includes requirements for the development, QA control and maintenance of these computer codes.

It should be noted that the Technical Report, MUAP-09001-P (R0), "Summary of Design Transient," in Section 2.4 provides a cross reference to DCD Section 15.0.2.2 for additional details concerning all of these codes except for GOTHIC. It should also be noted, in certain cases, that MHI has provided the executable file and input data as requested by the NRC.

The input deck for the GOTHIC code, used for containment integrity analysis, was also submitted to the NRC in support of Topical Report, MUAP-07012-P (R2), "LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR," review.

Table 03.09.01-5.1 Summary of Chapter 15 Code Documentation Submittals

Accident Analysis	Computer Code	MHI Response		
		Input Manual	Executable	Input Deck
Transient and Non-LOCA Non-LOCA Topical Report (MUAP-07010, July 2007)	MARVEL-M	UAP-HF-09099, March 2009	UAP-HF-09099, March 2009	UAP-HF-09099, March 2009
	VIPRE-01M	UAP-HF-08092, May 2008	UAP-HF-08092, May 2008	UAP-HF-08092, May 2008
	TWINKLE-M	UAP-HF-08138, August 2008	UAP-HF-08138, August 2008 (same as the UAP-HF-07189, December 2007)	UAP-HF-08138, August 2008
LBLOCA LBLOCA Topical Report (MUAP-07011, July 2007)	WCOBRA/TRAC	UAP-HF-08140, August 2008	UAP-HF-07189, December 2007	-
SBLOCA SBLOCA Topical Report (MUAP-07013, July 2007)	M-RELAP5	UAP-HF-08162, August 2008	UAP-HF-09100, March 2009	UAP-HF-08081, April 2008
CV Integrity (DCD 6.2.1)	GOTHIC	-	-	UAP-HF-08048, February 2008

(2) The number of design cycles for US-APWR refueling is 60 times. MHI will change the DCD as noted below.

Impact on DCD

See Attachment 1 for the markup of DCD Section 3.9, Revision 2, with the following changes.

- Change Table 3.9-1, RCS Design Transients (Sheet 1 of 2), Refueling event number of cycles from 120 to 60.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's responses to the NRC's questions.

- Small feedwater line break
- SG tube rupture

3.9.1.1.3.1 Small LOCA

The small LOCA is considered as a break of small branch pipe of a reactor coolant pipe. This transient is assumed to occur five times during the plant design life.

3.9.1.1.3.2 Small Steam Line Break

A small steam line break is considered as a break equivalent to a main steam relief valve pipe break~~safety valve opening and remaining open~~. This transient is assumed to occur five times during the plant design life.

3.9.1.1.3.3 Complete Loss of Flow

This accident applies to a complete loss of flow from full power resulting from the simultaneous loss of power to all RCPs. The consequences are a RT on low RCP speed, and a turbine trip results from RT. This transient is assumed to occur five times during the plant design life.

3.9.1.1.3.4 Small Feedwater Line Break

This transient applies to a small break in the piping between the SG and the main feedwater isolation valve. The main feedwater flow, in the affected loop, results in the fluid spilling through the break. No main feedwater is supplied to either of SG and then all of SG water level decrease. Upon receipt of low SG water level signal, the reactor is tripped and the emergency feedwater pumps are actuated automatically. This transient is assumed to occur five times during the plant design life.

3.9.1.1.3.5 SG Tube Rupture

This transient applies to the double-ended rupture of a single SG tube resulting in decreases in pressurizer water level and reactor coolant pressure. The reactor is manually tripped. The RT initiates a turbine trip. And then, the emergency feedwater pumps are actuated automatically upon receipt of low SG water level signal. The RCS pressure decreases continuously after the trip because of continued primary to secondary leakage through the ruptured SG tube. The continued RCS leakage results in an actuation of the safety injection pump. This transient is assumed to occur five times during the plant design life.

3.9.1.1.4 Level D Service Condition (Faulted Conditions)

The RCS under faulted condition transients (PC-5 in accordance with ANS N51.1 [Reference 3.9-2]) are considered as follows:

- Reactor coolant pipe break (large LOCA)
- Large steam line break
- Large feedwater line break

ATTACHMENT 1
to RAI 296-2254

Table 3.9-1 RCS Design Transients
(Sheet 1 of 2)

Event	Cycles
Level A Service Conditions	
RCP startup	3,000
RCP shutdown	3,000
Plant heat-up	120
Plant cooldown	120
Ramp load increase between 0 and 15 percent of full power	600
Ramp load decrease between 0 and 15 percent of full power	600
Ramp load increase between 15% and 100% of full power (5% of full power per minute)	600
Ramp load increase between 50% and 100% of full power (5% of full power per minute)	19,200
Ramp load decrease between 15% and 100% of full power (5% of full power per minute)	600
Ramp load decrease between 50% and 100% of full power (5% of full power per minute)	19,200
Step load increase of 10 percent of full power	600
Step load decrease of 10 percent of full power	600
Large step load decrease with turbine bypass	60
Steady-state fluctuation and load regulation	-
Steady-state fluctuation	1×10^6
Load regulation	8×10^5
Boron concentration equalization	39,600
Main feedwater cycling	2,100
Core lifetime extension	60
Refueling	60~~120~~
Turbine roll test	10
Primary Leakage Test	120
Secondary Leakage Test	120
Level B Service Conditions	
Loss of load	60
Loss of offsite power	60
RT from full power	-
With no inadvertent cooldown	60
With cooldown and no Safety Injection	30
With cooldown and Safety Injection	10