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TOKYO, JAPAN

February 13, 2013

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-13029

Subject: MHI's Revised Response to US-APWR DCD RAI No. 969-6763 (SRP 15.06.05)

References: 1) "Request for Additional Information No. 969-6763, SRP Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary, Application Section: 15.6.5" dated October 15, 2012.

2) UAP-HF-13008, "MHI's Response to US-APWR DCD RAI No. 969-6763 Revision 0 (SRP 15.06.05)," dated January 18, 2013.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Revised Response to Request for Additional Information No. 969-6763".

Enclosed is the revised response to the RAI contained within Reference 1. MHI inadvertently forgot to include the Impact on DCD mark-up that shows how the technical report MUAP-13001 is added to Tier 2 Chapter 1 Table 1.6-2 and cross-referenced in Chapter 15. The enclosed response revision corrects this error, but makes no change to the technical content of the response.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version of the RAI response (Enclosure 2), a copy of the non-proprietary version of the RAI response (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosures 1 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata
2. Revised Response to Request for Additional Information No. 969-6763 (Proprietary version)
3. Revised Response to Request for Additional Information No. 969-6763 (Non-proprietary version)

CC: J. A. Ciocco
J. Tapia

<u>Contact Information</u>

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Docket No. 52-021
MHI Ref: UAP-HF-13029

Enclosure 1

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Director, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed documents entitled "Revised Response to Request for Additional Information No. 969-6763" dated February 13, 2013 and have determined that portions of the documents contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design information and analysis of Loss-Of-Coolant Accidents, developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of Research and Development and detailed design for its software and hardware extending over several years.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with development of the US-APWR Loss-Of-Coolant Accident safety analysis. Providing public access to such information permits competitors to duplicate or mimic the Loss-Of-Coolant Accident safety analysis information without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced US-APWR Loss-Of-Coolant Accident safety analysis methodology development costs.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 13th day of February, 2013.

Y. Ogata

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

Enclosure 3

UAP-HF-13029
Docket Number 52-021

Revised Response to Request for Additional Information
No. 969-6763

February 2013

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/13/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 969-6763 REVISION 0

SRP SECTION: 15.06.05 – LOSS OF COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

APPLICATION SECTION: 15.6.5

DATE OF RAI ISSUE: 10/15/2012

QUESTION NO.: 15.06.05-101

This is a follow-RAI to 6062, Questions 98.

Attachment 1 (ML12254B066) Questions,

1. What is the expected time lapse for successive loop seal clearing? Provide basis for loop seal clearing time differences.
2. [

   ]
3. Demonstrate that the test deborated water slug volume was properly scaled to the expected maximum plant deborated water slug volume.
4. It was previously stated that the SBLOCA break size of 2 inches yields the maximum deborated water slug volume. Were additional break sizes evaluated? If so, provide the data which demonstrates that the 2 inch break size yields the maximum deborated water slug volume.
5. The scaled tests did not simulate any accumulator nitrogen effect. What is the accumulator nitrogen effect on the mixing fi values?

Attachment 2 (ML12254B066) Questions,

1. In Attachment 1, it was stated that Scenario 2 is the limiting case based on the local fi factor. It is not clear to the staff that either the minimum or maximum fi factor would be the most limiting from a core reactivity standpoint (i.e., least sub-critical). Demonstrate how the limiting fi factor is determined based on core reactivity considerations.
2. Justify the assumption that a constant axial boron concentration (i.e., based on inlet boron concentration) is limiting from a core reactivity perspective.
3. What is the basis for not including the effects of the worst stuck rod when calculating core reactivity?
4. Are uncertainties applied to the physics parameters? What is the basis for those

uncertainties?

5. Provide your analysis that supports APWR's long term sub-criticality when xenon has fully decayed and the boron is fully mixed.
6. The criticality analysis assumes that the core is at hot, full power, equilibrium xenon conditions. What would be the impact on the core reactivity analysis if the transient occurred at part power conditions with the corresponding lower xenon worth?

ANSWER:

MHI plans to submit a new technical report for GSI-185 in January 2013. This response identifies several technical details and clarifications that are addressed in technical report MUAP-13001, "US-APWR Criticality Evaluation Following Small Break LOCAs," which is provided as an enclosure to this RAI response.

Attachment 1 (ML12254B066)

Question 1:

[

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In addition, the integral effects tests simulating reflux condensation following a small-break LOCA in the PKL facility also empirically demonstrate that **[**

][1)] as shown in **Figure RAI-15.06.05-101-1**. The test facility is a mock-up of a 4-loop PWR of the 1,300 MW class, which replicates the entire primary system and most of the secondary system with a full-height volume and power scaling of 1:145.

The primary dimensions between the US-APWR and PKL test facility are compared in **Table RAI-15.06.05-101-1**. Natural circulation behavior is affected by the elevations of primary components such as the core, SG, hot and cold legs, which are comparable and mostly equivalent between the US-APWR and the test facility. Therefore, the experimental test data obtained in the PKL facility are applicable not only to the 1,300 MW class 4-loop PWR, but also to the US-APWR.

From the above investigations based on both numerical simulation and existing experimental test data, **[**

]

In the M-RELAP5 scenario analysis to be described in Section 2.2 of the technical report, natural circulation resumes 500 seconds earlier in one loop. And the natural circulation resumes in the other 3 loops within a short period. As a result of this timing, a criticality calculation assuming simultaneous resumption of natural circulation in all four loops will be provided in Section 5 of the technical report.

Question 2:

[

]

[

]

Question 3:

The volume of the deborated slug is assumed as the total volume [ ]. This is the expected maximum plant

deborated water slug volume. Since the test section is scaled by 1/7 in length, the test deborated water slug volume was scaled as $1/7^3$ of [ ].

The above-mentioned explanation will be described in Section 3.2.6.2 of the technical report.

Question 4:

The total volume of the [ ]. M-RELAP5 scenario simulations for the other break sizes are described in MHI's response to RAI 15.06.05-85[4)], which indicates that the volume of deborated condensate for all of the scenarios is more than [ ]. Therefore, the 2-in break case is selected as a reference case which can maximize the deborated condensate volume accumulated in loops.

Question 5:

In the M-RELAP5 scenario simulation, the RCS pressure does not decrease to the pressure that causes the accumulator to empty. Therefore, no non-condensable gas flows into the RCS.

Attachment 2 (ML12254B066)

Question 1:

The demonstration for determining the limiting fi factor will be provided in Section 4.3 of the technical report.

Question 2:

The justification of using constant axial boron distribution will be provided in Section 4.2.3 of the technical report.

Question 3:

In the criticality evaluation, the most reactive rod is assumed to be stuck out of the core as will be described in Section 4.2.1 of the technical report.

Question 4:

The uncertainty for the total control rod worth will be described in Section 4.2.1 of the technical report.

Question 5:

The analysis for long term sub-criticality has been performed and will be summarized in Section 5.2 of the technical report.

Question 6:

The criticality analysis assuming the equilibrium xenon worth at partial power conditions will be described in Section 4.2 of the technical report.

References:

1) K. Umminger et al., 'Experiments on Boron Dilution in the Integral Test Facility PKL,' The 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Soel, Korea, October 5-9, 2003.
2) [ ]
3) [ ]
4) Mitsubishi Heavy Industries, Ltd., MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0 (15.06.05), UAP-HF-11136, May 13, 2011.

Impact on DCD

DCD Table 1.6-2 "Material Referenced as Technical Reports", Table 15.0-1 "Summary of Event Classification, Initial Conditions and Computer Codes" Section 15.6.7, new DCD Sections 15.6.5.2.4, 15.6.5.3.1.4, 15.6.5.3.2.4, 15.6.5.3.3.4, and modifications to DCD Sections 15.6.5.6 and 15.6.7 are added as indicated in the attached DCD mark-up to cross-reference the new technical report, MUAP-13001 "US-APWR Criticality Evaluation Following Small Break LOCAs".

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Topical/Technical Report

A new technical report, MUAP-13001, "US-APWR Criticality Evaluation Following Small Break LOCAs," was provided as an enclosure to the original RAI response, MHI Letter NO. UAP-HF-13008

Table RAI-15.06.05-101.1 Scaling Comparison between US-APWR and PKL

Figure RAI-15.06.05-101.1 PKL III Test E2.2 Measurement for Loop Behavior (Ref. 1)

Table 1.6-2 Material Referenced as Technical Reports (Sheet 5 of 5)

MIC-03-01-00002
MIC-03-01-00002 S1

Report Number[1]	Title	DCD Section Number[2]
MUAP-10023-P MUAP-10023-NP	Initial Type Test Results of Class 1 E Gas Turbine Generator System, Revision 3, September 2011	1.5.2
MUAP-10024	Structural Design Criteria for US-APWR Access Building, Revision 1, November 2011.	3.7.2
~~MUAP-11001~~	~~Auxiliary Building Model Properties, SSI Analyses, and Structural Integrity Evaluation for the US-APWR Standard Plant, Revision 1, June 2011.~~	~~3.7.2~~
MUAP-11002	Turbine Building Model Properties, SSI Analyses, and Structural Integrity Evaluation, Revision 0, January 2011.	3.7.2
MUAP-11003-P MUAP-11003-NP	Summary of Stress Analysis Results for Pressurizer Surge Line, Revision 1, March 2011.	3.6.3
MUAP-11005	Research Achievement of SC Structure and Strength Evaluation of US-APWR SC Structure based on 1/10th Scale Test Results, Revision ~~0~~1, ~~February 2011~~December 2012.	3.8.3
~~MUAP-11006~~	~~FE Model Development and Verification, Revision 0, June 2011.~~	~~3.7.2, 3H.1, 3H.2.1, 3H.2.3, 3H.2.4, 3H4.1, 3H.4.2, 3H.4.3~~
MUAP-11007	~~Results of Evaluation Using LMSM for R/B Complex, Revision 0, June 2011.~~Ground Water Effects on SSI, Revision 2, November 2012.	3.7.2, 3.8.5
~~MUAP-11011~~	~~Effects of Structure-Soil-Structure Interaction (SSSI) on Standard Seismic Design of US-APWR Plant, Revision 0, June 2011.~~	~~3.7.2~~
MUAP-11012-P MUAP-11012-NP	US-APWR RCCA Insertion Limit Load Test Report, Revision 0, March 2011.	3.9.5
MUAP-11013	Design Criteria for SC Modules, Revision 1, August 2011.	3.8.3
MUAP-11014-P MUAP-11014-NP	Over Temperature ΔT and Over Power ΔT Trip Function and Setpoint Determination Process, Revision 0, June 2011.	7.2.1
MUAP-11017-P MUAP-11017-NP	Hydraulic Test of the Full Scale US-APWR Fuel Assembly, Revision 0, May 2011	4.2
MUAP-12002-P MUAP-12002-NP	Sliding Evaluation and Results, Revision 0, May 2012	3.7.2
MUAP-13001-P MUAP-13001-NP	US-APWR Criticality Evaluation Following Small Break LOCAs	15.6.5

MIC-03-01-00022 (MUAP-10024)
MIC-03-01-00022 (MUAP-11005)
MIC-03-01-00022 (MUAP-11006)
MIC-03-01-00022 (MUAP-11007)
MIC-03-01-00022 (MUAP-11011)
MIC-03-01-00022 (MUAP-12002)
DCD_15.6.5-101 (MUAP-13001)

NOTE(1): -P(proprietary), -NP(non-proprietary)
(2): If actual section number is indicated as x.y.z.a.b, a x.y.z level is used for the DCD Section Number. (ex. When actual section number is 6.3.2.1.2, only 6.3.2 is used in Table.

MIC-03-01-00022

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 4 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler*7	
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve	AOO	MARVEL-M	min	–	max feedback Figure 15.0-2	4466
15.6.2	Radiological consequences of the failure of small lines carrying primary coolant outside containment	PA	RADTRAD	–	–	–	4540*3
15.6.3	Radiological consequences of SGTR	PA	MARVEL-M	min	–	max feedback Figure 15.0-2	4555*2
15.6.5	Loss-of-Coolant Accidents	PA	WCOBRA/TRAC, HOTSPOT	*4	–	*4	4466
			M-RELAP5	*5	–	*6	4555*2
			ANC	–	–	–	–

DCD_15.0.0-30

DCD_15.06.05-101

Notes:
*1Steady state analysis
*2102% of 4466MW_t (NSSS thermal power)
*3102% of 4451MW_t (core thermal power)
*4Applicability confirmed (Ref.15.0-18).
*5 Conservative Moderator Density Coefficient changes with moderator density assumed (Ref.15.0-20).
*6Conservative Doppler Temperature Coefficient changes with moderator density assumed (Ref.15.0-20).
*7Doppler feedback may be modeled by a Doppler power coefficient as well as a Doppler fuel temperature coefficient. Unless otherwise noted, the two coefficients are assumed to be consistent (i.e. both minimum or both maximum). Values for the Doppler fuel temperature coefficient are provided in Subsection 15.0.0.2.4.
*8For the over cooling events from hot zero power (Subsections 15.1.4 and 15.1.5), the maximum negative value of the Doppler fuel temperature coefficient is used to conservatively add more reactivity to the core due to the coolant temperature decrease. The minimum Doppler power coefficient is chosen since power is increasing during the return to criticality.

DCD_15.0.0-30

Attachment

In the small break LOCA, the RCS pressure may not fall below the pressure that allows water injection from the accumulators. In this case, the HHIS alone provides the core cooling function. Continued operation of the SI pumps supplies borated-water during long term cooling. Core temperatures are reduced to long term, steady state levels associated with the dissipation of residual heat generation.

15.6.5.2.3 Description of Post-LOCA Long Term Cooling

There are two considerations in the post-LOCA long term cooling that must be addressed: maintaining long term decay heat removal and the potential for boric acid (H_3BO_3) precipitation. After the quenching of the core at the end of reflood phase, continued operation of the ECCS supplies borated water from the RWSP to remove decay heat and to keep the core subcritical. Borated water from the RWSP is initially injected through DVI lines (RV injection mode). If left uncontrolled, boric acid (H_3BO_3) concentration in the core may increase due to boiling and reach the precipitation concentration. Boric acid precipitation in the core could affect the core cooling. To prevent the boric acid precipitation, the operator switches over the operating DVI lines to the hot leg injection line (simultaneous RV and hot leg injection mode).

In the case of a hot leg break, almost all ECCS water injected through DVI lines passes through the core and exits from the break point. As a result, the boric acid concentration in the core does not increase. Even after the switchover, sufficient ECCS water passing through the core for decay heat removal is assured, and that simultaneously prevents any increase in boric concentration in the core.

In the case of a cold leg break, the ECCS water through DVI lines is not effective in flushing the core. As the result, boric acid concentration in the core may increase. After the switchover, almost all ECCS water injected into the hot leg passes the core. Therefore, the boric acid concentration in the core decreases.

The main objective of the post LOCA long term cooling evaluation is to determine the switchover time from RV injection mode to the simultaneous RV and hot leg injection mode to prevent the boric acid precipitation, hence the long-term cooling is assured.

15.6.5.2.4 Description of Small Break LOCA Boron Dilution

DCD_15.06.05-101

Generic Safety Issue No. 185 (GSI-185) "Control of Recriticality Following Small-Break LOCAs in PWRs," identified a concern of potential recriticality following a small break LOCA. Under the condition with a small break, the RCS depressurizes and then core decay heat is removed by natural circulation. In the scenario of GSI-185, an operator action to depressurize the secondary side of the SGs is anticipated after termination of natural circulation such that the RCS is further depressurized towards the reactor shutdown. Therefore, the RCS transitions to the reflux condensation period, in which boron-free steam generated by core decay heat is cooled and condensed in the SGs and the deborated condensate can accumulate in the loop seals.

As the RCS depressurizes during the reflux condensation period, ECCS flow rate increases and the RCS inventory starts recovering once ECCS flow rate is larger than the break flow rate. When natural circulation resumes, the accumulated condensate gets transported to the reactor vessel as a deborated slug, which has the potential to cause a

core recriticality and fuel damage. DCD_15.06.05-101

The purposes of the small break boron dilution evaluation are to identify the small break LOCA scenario resulting in the rapid insertion of the deborated water to the reactor vessel and core, and to assess the safety margin to the core recriticality under the identified GSI-185 scenario.

15.6.5.3 Core and System Performance

15.6.5.3.1 Evaluation Model

The reactor is designed to withstand thermal effects caused by a LOCA event including the double-ended severance of the largest RCS pipe. The reactor core and internals together with the ECCS are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core is preserved following the accident. The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the requirements of 10 CFR 50.46. The requirements are:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

b. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

d. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In this best-estimate large break LOCA analysis, the analysis method and inputs are identified and assessed to estimate the uncertainty of the calculated results. This uncertainty is accounted for, in order to obtain a high probability that the criteria (a) through (c) above are not exceeded.

15.6.5.3.1.1 Large Break LOCA Evaluation Model

Large Break LOCA Calculation Methodology

The 10 CFR 50.46 permits the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. In particular, best estimate thermal-hydraulic models may be used to predict the peak cladding temperature (PCT), local maximum cladding oxidation (LMO), and maximum core wide cladding oxidation

The RWSP, accumulator, and RCS are considered as the sources of borated water. The initial boric acid concentration is assumed to be the maximum allowed for operating conditions. The water mass in the RWSP and accumulators is assumed to be the maximum allowed for operating conditions, because large quantity of borated water in the sources yield higher concentration of boric acid in the mixing volume. A minimum amount of RCS water mass is assumed because boric acid concentration in the RCS is lower than that in the RWSP and accumulators. Boric acid concentration is also considered in calculating liquid mass density.

Effects of System Pressure

The effects of system pressure are as follows.

- Higher system pressure gives a lower void fraction in the core and consequently, more water mass in the mixing volume.
- Higher system pressure increases the boiling rate because of the decrease in the latent heat.
- SI system injection flow rate decreases with an increase in system pressure.

The first item implies that a higher system pressure reduces boric acid concentration in the mixing volume, while the second one yields a reverse effect. In the evaluation, the atmospheric pressure is assumed for the large break LOCA and a higher pressure for the small break.

Criterion of Boric Acid Precipitation

From Reference 15.6-28, the boric acid precipitation criterion is conservatively assumed to be 29.27 wt.%, which is the precipitation concentration in the atmospheric pressure. Core pressure is higher than the atmospheric pressure, due to the downcomer head and the flow-resistances around the loop. Therefore, the core boiling temperature and the boric acid solubility will be higher than the assumed values. Furthermore, no credit is taken for the RWSP pH additive that increases the boric acid solubility. Hence, this criterion is conservative.

15.6.5.3.1.4 Small Break LOCA Boron Dilution Evaluation Model DCD_15.06.05-101

M-RELAP5, which is used in the evaluation of US-APWR small break LOCAs for 10 CFR 50.46, is used to identify the break sizes that result in deborated water accumulation in the loop seal and potential rapid insertion to the reactor vessel as a slug. The ANC code is used to evaluate the core recriticality following the deborated water insertion to the reactor vessel, in conjunction with the vessel mixing model determined from the experimental data obtained from MHI's scaled vessel test facility.

In evaluating the acceptability of the US-APWR response to a GSI-185 scenario, *Criterion 27 "Combined reactivity control systems capability,"* from Appendix A to 10 CFR 50 "General Design Criteria for Nuclear Power Plants" (GDC 27) is applied.

MUAP-13001, "US-APWR Criticality Evaluation Following Small Break LOCAs" (Reference 15.6-29) provides the analysis models used to evaluate the GSI-185 scenario and the expected core recriticality margin following the small break LOCA. DCD_15.06.05-101

15.6.5.3.2 Input Parameters and Initial Conditions

15.6.5.3.2.1 Large Break LOCA

Table 15.6.5-1 lists the major plant parameter inputs identified for use in the large break LOCA analysis. An initial transient run was made with mostly nominal values, or in some cases, a conservative one. Confirmatory WCOBRA/TRAC runs were performed by varying these limiting parameters over their normal operational ranges to determine the limiting value. The limiting values were used for the reference transient. The other parameters, which are not limiting parameters, are treated as randomly sampled over their operating range in the ASTRUM calculations. Table 15.6.5-1 also lists the major uncertainty parameters and ranges to perform the ASTRUM runs for large break LOCA of the US-APWR based on the operating ranges and other aspects.

- The limiting single failure in the large break LOCA analysis is assumed, which is the loss of one train of ECCS and a second train out of service for maintenance; In this case, only two SI pumps are available.
- Minimum ECCS safeguards are assumed, which results in the minimum delivered ECCS flow available to the RCS.
- Minimum containment pressure is applied for conservatism as described in Section 6.2.1.5.

15.6.5.3.2.2 Small Break LOCA

Spectrum analysis is performed to determine a limiting break size within the small break LOCA category. In addition, sensitivity analyses are reported in Reference 15.6-16, which covers the entire spectrum of break size, break orientation and break location, also noding, time-step size and input sensitivity studies. The sensitivity analyses are performed by complying with the requirements set forth in 10 CFR Appendix K to Part 50 on ECCS Evaluation Models. The objective is performed to determine the effects of various modeling assumption on the calculated PCT, LMO and CWO. Three small break LOCA cases are reported in this section. They are as follows:

- 7.5-inch downside break, which is the limiting break for PCT during the loop-seal clearance phase.
- 1-ft^2 downside break, which is the limiting break for PCT during the boil-off phase.
- 3.4-inch break, which is a DVI line break, with only 1 train of SI system is assumed to operate.

The major plant parameters inputs used in the Appendix-K based small break LOCA analysis are listed in Table 15.6.5.2. The top-skew axial power shape is chosen because it provides the distribution of power versus core height that maximizes the PCT.

15.6.5.3.2.4 Small Break LOCA Boron Dilution

DCD_15.06.05-101

Input parameters and analysis conditions to evaluate the GSI-185 scenario and core recriticality following the small break LOCA are described in Reference 15.6-29.

15.6.5.3.3 Results

15.6.5.3.3.1 Large Break LOCA Analysis Results

The Result of Reference Transient Calculation

The reference transient calculation is performed based on the confirmatory calculation results in order to obtain the conservative estimation. Figures 15.6.5-1 through 15.6.5-7 present the results of the reference case for the best estimate large break LOCA analysis. The transient is initiated from the end of a steady-state run. The sequence of events for the reference case large break LOCA is listed in Table 15.6.5-6, which shows the plant actions (e.g. trips, etc) and those phenomena observed in the calculation (e.g., end of blowdown, etc).

(1) Blowdown phase

During the first few seconds of the transient, the core water inventory decreases rapidly. During the blowdown phase, the initial stored energy is the main contributor to the temperature rise and boiling. The decay heat is a secondary contributor. The RCPs are presumed to trip concurrent with the break in the LOOP scenario. Consequently, DNB occurs and the cladding temperature rises quickly even though the core power decreases. The hot rod cladding temperature at the limiting elevation for large break LOCA is shown in Figure 15.6.5-1. At six seconds into the transient, an ECCS actuation signal is generated due to the low pressurizer pressure. In the early blowdown phase, an upward flow takes place in the core removing the core decay heat by way of two-phase heat transfer. About 13 seconds into the transient, the accumulator begins to inject water at a high rate into the cold leg regions.

Figure 15.6.5-2 shows the hot assembly exit vapor, entrainment, and liquid flowrates transients. This figure displays the flow rates for the vapor, entrained liquid and continuous liquid at the top of the hot assembly.

The core pressure transient is illustrated in Figure 15.6.5-3. Following the break, the vessel rapidly depressurizes during the subcooled break flow. The pressure reduction rate then decreases as boiling begins in the vessel and the break flow becomes two-phase. As the RCS pressure falls and approaches the containment atmosphere pressure, the break and core flows reduce accordingly. The blowdown phase ends at 33 seconds.

(2) Refill phase

Figure 15.6.5-4 presents the transient of liquid level in the lower plenum. During the refill phase, core heat up occurs because the primary heat transfer mechanism is convection to steam. The lower plenum is filled with borated water supplied by the accumulators. At approximately 37 seconds, the lower plenum fills to the bottom of the core, which ends the refill period and begins the reflood period.

hours. Figure 15.6.5-43 also shows the dilution effect of the hot leg injected flow after the switchover.

Core Cooling after Switchover to Hot Leg

Evaluation is also performed to clarify the effect of early switchover from RV injection mode to the simultaneous RV and hot leg injection mode. If switchover is performed too early, then the injected water to the hot legs is circulated around the RCS loops by entrainment and there may not be sufficient water for core cooling and boron dilution in the core. Entrainment threshold calculations similar to those reported in Reference 15.6-10 demonstrates that significant hot leg entrainment will not occur after 100 minutes. Therefore, the evaluation demonstrates that both hot leg injection and DVI are sufficient to provide core-cooling flow at four hours after the LOCA.

15.6.5.3.3.4 Small Break LOCA Boron Dilution Evaluation Results

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The US-APWR is only susceptible to a potential recriticality scenario under a narrow range of small break sizes (approximately 1.5-in diameter and 2.5-in diameter). Below this range, the breaks are too small to interrupt the natural circulation and deborated water does not accumulate in the loops. Above this range, the breaks are large enough to depressurize the RCS faster than the SG secondary side pressure and deborated water does not accumulate in the loops since the secondary side behaves as a heat source. In addition, the breaks above this range prevent the restart of natural circulation, which also eliminates the potential for a rapid insertion of deborated water.

The mixing of the deborated slug with the borated water in the vessel downcomer and lower plenum was investigated using experimental data obtained in MHI's scaled test facility. The boron concentration distribution at the core inlet was also quantified from the experimental test data.

The core recriticality is evaluated based on the boundary conditions described above with conservative assumptions for the core boron concentration prior to the deborated slug insertion, the core boron concentration distribution following the slug insertion, and the xenon worth. The evaluated results show that the US-APWR core remains subcritical after the deborated slug insertion. In conclusion, the US-APWR conforms to the related safety criterion GDC 27 under the postulated GSI-185 scenario.

Reference 15.6-29 provides the detailed evaluation results.

15.6.5.4 Barrier Performance

The Barrier Performance is discussed in detail in the Chapter 6, Section 6.2 on the Containment System. In general, it discusses the evaluation of the containment vessel pressure and temperature transients that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public during and after a LOCA.

input parameters for the MCR and TSC consequence analysis during the LOCA are summarized in Table 15.6.5-5, and Tables 15A-18 through 15A-24.

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix A of RG 1.183 (Ref. 15.6-4).

15.6.5.5.3 Results

The doses calculated for the EAB and the LPZ boundary are listed in Table 15.6.5-16. The TEDE doses for the limiting 2 hours are calculated to be 13 rem at the EAB and 13 rem at the LPZ outer boundary. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The doses calculated for the MCR personnel due to airborne activity entering the MCR are listed in Table 15.6.5-16. Also listed on Table 15.6.5-16 are the doses due to direct shine from the activity in the containment, from the radioactive plume and from the MCR emergency filtration unit. The total of the four dose pathways is within the dose criteria of 5 rem TEDE as defined in GDC 19. The dose for TSC is bounded by the MCR doses.

15.6.5.6 Conclusions

The US-APWR satisfied all criteria for the postulated LOCA transient:

- The best-estimate analysis of the large break LOCA demonstrates that the acceptance criteria of 10 CFR 50.46 are satisfied.
- The conservative analysis of the small break LOCA, which is based on the Appendix K, demonstrates that the acceptance criteria of 10 CFR 50.46 are satisfied.
- The switchover to the simultaneous RV and hot leg injection mode at four hours after a LOCA prevents boric acid precipitation in the core, and the post-LOCA long term cooling is assured.
- The core criticality analysis demonstrates that the core remains subcritical following the small break LOCA, which satisfies the criterion given in GDC 27 for GSI-185. DCD_15.06.05-101
- The EAB and LPZ doses are shown to meet the 10 CFR 50.34 dose guidelines.
- The dose for the MCR personnel is shown to meet the dose criteria given in GDC 19.
- The requirements of the TMI Action Plan items are met.

15.6-27 H. C. Yeh, Modification of Void Fraction Correlation, Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, April 5-9, 1994.

15.6-28 Cohen, P., Water Coolant Technology of Power Reactors, Chapter 6, Chemical Shim Control and pH Effect, ANS-USAEC, 1969.

15.6-29 US-APWR Criticality Evaluation Following Small Break LOCAs, MUAP-13001-P Rev.0 (Proprietary) and MUAP-13001-NP (Non-Proprietary), January 2013. DCD_15.06.05-101