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TOKYO, JAPAN

May 13, 2011

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-11136

Subject: MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0 (SPR 15.06.05)

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0". The enclosed materials provide MHI's response to Questions 15.06.05-83 to 15.06.05-86 of the NRC's "Request for Additional Information (RAI) 718-54022 Revision 0," dated March 17, 2011.

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 in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is 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 material designated as "Proprietary" in Enclosure 2 be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0 (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0 (non-proprietary)

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

Contact Information

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Telephone: (412) 373-6466

ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11136

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am General Manager, 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 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 document entitled "MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0," and have determined that the document contains 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 basis for holding the referenced information confidential is that it describes the unique design of the safety analysis, developed by MHI (the "MHI Information").

4. The MHI Information is 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. Therefore public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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 May, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11136
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 718-5402 Revision 0

May 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 718-5402 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:	**3/17/2011**

QUESTION NO.: 15.06.05-83

The response to RAI Question 15.6.5-56 provided in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1" states that "significant amount of the deborated water may flow into the core when the natural circulation is reestablished or RCPs are restarted after the potential long-term reflux condensation."

Describe the dilution scenarios involving RCP restart that have been considered in the US-APWR evaluation of core recriticality associated with the inherent boron dilution mechanism occurring during small break LOCAs. Identify and describe the conditions that are found to lead to the worst core recriticality consequences. In particular, describe the conditions in the reactor coolant system (RCS) and in the primary loops at the time of pump restart, RCP restart timing considerations, and loop transient flow characteristics following pump restart. Discuss the core recriticality consequences under the identified limiting conditions including the coupled thermal-hydraulic system response and conditions.

ANSWER:

Generic Safety Issue (GSI) 185 (Ref. 1) identified a concern of potential recriticality following a small break LOCA (SBLOCA) because of the accumulation of deborated water in the reactor coolant pump (RCP) suction piping due to the condensation of steam in the primary side of steam generator (SG) tubes during reflux condensation.

Fundamental scenarios of the boron dilution and rapid deborated water transient following a SBLOCA are described as below:

1) Blowdown: Upon initiation of the break, the reactor coolant system (RCS) primary side depressurizes until flashing of the hot coolant into steam begins. Reactor trip is initiated on the low pressurizer pressure setpoint of 1860 psia. Closure of the condenser steam dump valves isolates the SG secondary side. As a result, the SG secondary side pressure rises to the safety valve setpoint of 1296 psia, and steam is released through the safety valves. The emergency core cooling system (ECCS) actuation signal is generated at the time the

pressurizer pressure decreases to the low pressurizer pressure setpoint of 1760 psia and safety injection initiates after a time delay. Then the RCPs automatically trip after 3 seconds delay upon the reactor trip signal in the case without offsite power, [ ]. The rapid depressurization ends when the pressure falls to just above the saturation pressure of the SG secondary side, which is at the safety valve setpoint.

2) Natural circulation: When the blowdown phase ends, natural circulation is established in the RCS loops with the decay heat being removed by heat condensation and convection heat transfer to the SG secondary side. The emergency feedwater (EFW) is initiated to maintain the secondary side inventory. The natural circulation phase will continue until there is insufficient driving head on the cold leg side of the loops, due to the accumulation of steam in the loops between the top of the SG tubes and the loop seals. Then, the natural circulation is interrupted.

3) Reflux condensation: Once a quasi-steady state is established for the RCS condition, the operator action is to proceed to reactor shutdown. In this procedure, the SG secondary side is depressurized by opening the main steam depressurization valves (MSDVs) or the main steam relief valves (MSRVs), which decreases the RCS coolant pressure and temperature by removing the heat in SGs. [ ] The reflux condensation phase is characterized during this depressurization period. During natural circulation is interrupted, coolant is boiled in the core and the generated steam is condensed in the SGs to remove the decay heat. These conditions lead to boron dilution because (1) boric acid does not markedly dissolve into steam and (2) boron-free condensate can accumulate downstream of SG tubes and in the RCP suction piping. Because the boric acid is not as volatile as water at these conditions, boron is concentrated within the reactor vessel, while the deborated condensate eventually accumulates and fills the RCP suction piping and then flows to the reactor vessel at the rate of condensation along with the borated safety injection water. As long as the deborated water is returned to the reactor vessel at the rate of condensation or is lost through the break, the coolant boron concentration in the reactor vessel does not fall below the critical boron concentration needed to maintain the core subcriticality.

4) Deborated water transient: There are two potential scenarios for the rapid deborated water transient. One is initiated by a restart of RCPs, and the other is by recovery and resumption of natural circulation due to increase in ECCS flow rate as the RCS is depressurized during the reflux condensation phase. When the transient occurs, slugs of the deborated water flow into the reactor vessel, toward the reactor core. The deborated water mixes with the stagnant borated water in the vessel downcomer and in the lower plenum. If the coolant boron concentration flowing to the core is lower than the critical value, a recriticality will occur following the deborated water transient.

A narrow range of break sizes [ ], as is described in the response to RAI 15.06.05-85, is potentially susceptible to the accumulation of deborated water in the RCP suction piping. Below this range, the breaks are too small to interrupt the natural circulation before SG cool down begins. Consequently, reflux condensation does not occur following natural circulation, and deborated water does not accumulate in loops. Above this range, the breaks are large enough 1) to depressurize the RCS faster than the SG secondary side pressure is reduced by the cool down (the secondary side is the heat source to the RCS, and deborated water does not accumulate in the loops), or 2) to prevent the natural circulation resumption during the reflux condensation phase and later.

The scenario of RCP restart is difficult to occur. [ ]

[

]

Even if the RCP restart scenario is taken into account, the scenario cannot be the limiting case for the recriticality evaluation [

]. The recriticality evaluation explained in MHI's responses to RAIs 15.06.05-56 (Ref. 2) and 15.06.05-90 (Ref. 3) assumes that [

].

References:

1. USNRC, General Safety Issue 185: Control of Recriticality Following Small-Break LOCAs in PWRs, NUREG-0933.
2. Mitsubishi Heavy Industries, Ltd., MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1, UAP-HF-09384, July 2009.
3. Mitsubishi Heavy Industries, Ltd., 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (15.06.05), UAP-HF-11106, April 2011.

Impact on DCD

There is no impact on the DCD.

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.

This completes MHI's response to the NRC's QUESTION NO. 15.06.05-83.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 718-5402 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:	**3/17/2011**

QUESTION NO.: 15.06.05-84

The response to RAI Question 15.6.5-56 given in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1" states that "significant amount of the deborated water may flow into the core when the natural circulation is reestablished or RCPs are restarted after the potential long-term reflux condensation."

Describe the dilution scenarios assuming resumption of RCS natural circulation that have been considered in the US-APWR evaluation of core recriticality associated with the inherent boron dilution mechanism occurring during small break LOCAs. Identify and describe the conditions that are found to lead to the worst core recriticality consequences. In particular, discuss the RCS loop conditions preceding natural circulation resumption, the process of natural circulation resumption in individual loops, timing aspects of interruption and resumption of natural circulation, and loop flow transient characteristics during natural recirculation resumption. Provide the technical base in support of the identified and applied limiting boron dilution conditions. If test data are used as part of the technical base, demonstrate their applicability, sufficiency, and scaling with regard to the US-APWR reactor design. Discuss the results from the analysis of the core recriticality consequences under the identified limiting conditions including the coupled thermal-hydraulic system response and conditions.

ANSWER:

The fundamental evolution for the boron dilution and rapid deborated water transient is described in MHI's response to RAI 15.06.05-83. This response focuses on the scenario of a rapid deborated water transient caused by recovery and resumption of natural circulation.

Deborated water can accumulate in loops during the reflux condensation phase initiated with the steam generator (SG) cool down. As the coolant temperature in the secondary side of SGs decreases, the deborated steam carried from hot legs is cooled and condensed in the SG tubes and a part of the condensate accumulates in loops and fills the reactor coolant pump (RCP) suction piping.

During the reflux condensation, the reactor coolant system (RCS) is also depressurized due to the SG cool down. As the RCS pressure decreases, the safety injection (SI) flow rate increases and liquid level is also increased in the RCS and SG tubes, resulting in recovery and resumption of the natural circulation. Details of the plant behavior are simulated and explained in MHI's response to RAI 15.06.05-85.

Resumption of the natural circulation occurs during a period of the SG cool down and RCS depressurization phase, i.e. the reflux condensation phase. For the core recriticality evaluation, the natural circulation resumption is assumed to occur at the end of reflux condensation phase since 1) the RCS coolant temperature decreases to [ ], which requires the highest boron concentration to maintain subcriticality, and 2) the duration of reflux condensation is long enough to fill the loops with the deborated water.

An investigation given in MHI's response to RAI 15.06.05-85 indicates that [ ]

Taking the scenario of natural circulation into account, the thermal-hydraulic condition and system response assumed for the bounding analysis are conservative enough.

References:

1. Mitsubishi Heavy Industries, Ltd., MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1, UAP-HF-09384, July 2009.
2. Mitsubishi Heavy Industries, Ltd., 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (15.06.05), UAP-HF-11106, April 2011.

Impact on DCD

There is no impact on the DCD.

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.

This completes MHI's response to the NRC's QUESTION NO. 15.06.05-84.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 718-5402 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:	**3/17/2011**

QUESTION NO.: 15.06.05-85

In the response to RAI Question 15.6.5-56 in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1," it is stated that a prolonged reflux condenser phase appears probable for US-APWR cold leg breaks that range between 2 and 4 inches of equivalent break diameters. It is further claimed that breaks below 1 inch do not lead to natural circulation interruption and break sizes larger than 6 inch depressurize the primary reactor system to a degree that precludes the occurrence of significant reflux condensation.

Identify the US-APWR small break cases that have been analyzed in the US-APWR evaluation of the inherent boron dilution mechanism during small break LOCAs. Identify the US-APWR small break LOCA model and computer codes used to perform supporting analyses. Substantiate the sufficiency of the analyzed cases in terms of establishing the break size range of interest and the break cases analyzed. Indentify the examined modeling assumptions with regard to availability, quantity and distribution of ECCS injection and SG emergency feed water supply. Present assessments for the amount of steam condensate generated by reflux condensation in the steam generators (SGs). As part of this analysis, present the results for the condensate amount returned in countercurrent flow to the reactor vessel hot leg, condensate collected in SG regions including the SG exit chamber, loop crossover pipe, cold leg, and condensate transported to the vessel downcomer. Include consideration of stratified conditions involving hot boron-depleted condensate and colder borated coolant in the reactor cold leg and downcomer regions. Explain how results from performed thermal-hydraulic analyses were applied in the US-APWR boron dilution recriticality evaluation.

ANSWER:

US-APWR small break LOCA (SBLOCA) cases were analyzed using M-RELAP5 to investigate the mechanism of the potential boron dilution.

(1) Computer Code

A computer program of M-RELAP5 (Ref. 1), which is being reviewed by the NRC for its application to US-APWR small break LOCAs (SBLOCAs), was used to investigate plant behaviors expected under the boron dilution scenario.

(2) Code Applicability

M-RELAP5 applicability to US-APWR SBLOCAs has been widely assessed using test data obtained in various facilities with various break sizes (Ref. 1). In particular for a very small break size which is susceptible to induce the boron dilution as described later, the applicability was validated using test data obtained in the ROSA/LSTF facility, [ ]. M-RELAP5 assessment results using the [ ] test data are given in MHI's response to RAI 8.2.1-3 for the topical report MUAP-07013 (Ref. 2). Scalability of the ROSA test facility is examined in the scaling analysis report for the US-APWR SBLOCA (Ref. 3).

From US-APWR SBLOCA PIRT (Phenomena Identification Ranking Table), natural circulation and SG heat transfer are the important phenomena for the boron dilution scenario. M-RELAP5 ability to simulate these phenomena was assessed using SBLOCA test data obtained in the ROSA facility (Refs. 1 and 2).

M-RELAP5 ability to simulate the steam condensation in steam generators (SGs) has not been directly validated using experimental test data since the bounding analysis employs several conservative or bounding assumptions to provide a limiting evaluation for the core recriticality (Refs. 4 and 5). For reference, however, RELAP5, the base code of M-RELAP5, was well validated in its application to the reflux condensation phenomena during SBLOCAs using test data obtained in the ROSA/LSTF facility (Ref. 6). Differences between M-RELAP5 and RELAP5 are primarily due to an addition of conservative models required in 10CFR50 Appendix K, which do not affect the ability of each code to predict reflux condensation behavior.

The counter-current flow occurring in the primary side of a SG is of interest for reflux condensation behavior. The ability of M-RELAP5 was assessed using the separate-effects test (SET) data obtained in UPTF and Dukler Air-Water facilities (Ref. 1).

(3) Analysis Models and Conditions

(a) Analysis Models
M-RELAP5 was applied to investigate the US-APWR boron dilution event in the same manner as for ECCS (emergency core cooling system) performance evaluations in US-APWR DCD Chapter 15.6.5 (Ref. 7). Changes from the DCD analysis are [

] The others, plant nodding scheme except the individual representation for all loops, and applied thermal-hydraulic models and options, are identical to the models for the DCD analysis.

(b) Analysis Conditions
Analysis conditions for the US-APWR boron dilution are:

1) initial plant conditions same as for US-APWR DCD SBLOCAs
2) a single failure assumption for one ECCS train
3) one additional ECCS train out-of-service for maintenance,
4) loss of offsite power (LOOP) assumption, and

5) manual cool down of the SG secondary side.

[

]

During this period, coolant pressure and temperature are lower in the SG secondary side than in the primary side, and steam generated by the core decay heat is condensed in the SG tubes. A part of the condensate returns to the hot leg and the other falls to the RCP suction piping. Therefore, the reflux condensation occurs during this depressurization period and fills the RCP suction piping with the deborated condensate.

As for occurrence of the rapid deborated water transient, the present analysis does not directly address a scenario of the RCP restart since this scenario cannot be the limiting case for the recriticality evaluation and is bounded by the analysis in References 4 and 5 as is explained in MHI's response to RAI 15.06.05-83.

Instead, a scenario with natural circulation resumption is examined in this analysis. Of interest is 1) Is the resumption of natural circulation possible? 2) When does resumption occur? and 3) [ ]

(c) Break Case
In examining the boron dilution scenario, the US-APWR SBLOCA cases are categorized into two cases; 1) cold leg break and 2) hot leg break. For the case of cold leg break, the break is located downstream of the SG, and the relatively cold coolant discharges out. Compared with this case, the hot leg break case is different since the relatively hot coolant discharges out the break located upstream of the SG. In the latter case, the RCS depressurizes faster, resulting in less reflux condensation because the SGs behave as a heat source and avoid inducing the reflux condensation when the RCS pressure is lower than the SG secondary side pressure. Therefore, the cold leg break is more appropriate to the boron dilution scenario.

(4) Analysis Results

(a) Break Size inducing Reflux Condensation
Generic Safety Issue (GSI) 185 (Ref. 9) states that the occurrence of inherent boron dilution following SBLOCAs is susceptible in a narrow range of very small break sizes. M-RELAP5 simulations for the US-APWR SBLOCA boron dilution also indicate that the reflux condensation is possible under a range of break size [ ].
Below this range, the breaks are too small to interrupt the natural circulation before the manual cool down of the SGs. Above this range, the break flow rate is large enough to prevent natural circulation resumption during reflux condensation and later.

Table RAI-15.06.05-85.1 summarizes the event sequences for 1) time to start the SG cool down, 2) time to terminate the SG cool down, 3) time the natural circulation is interrupted, and 4) time the natural circulation is resumed.

(b) Thermal-Hydraulic System Response
The 2.0-in cold leg break case is examined in detail. Evolutions of the primary and secondary

pressure and cold inlet coolant temperature are depicted in **Figures RAI-15.06.05-85.1 and 2**. The primary pressure rapidly decreases due to break initiation. When the pressurizer pressure falls below the setpoint, the reactor trip and ECCS initiation signals are generated. Concurrently, the RCPs and feedwater pumps are tripped, and the emergency feedwater (EFW) starts filling the secondary side of SGs. Then, the primary pressure is equalized with the secondary pressure during the natural circulation phase until 2000 seconds. The primary pressure starts decreasing again around 2,000 seconds, which is caused by vapor phase discharged out the break as is shown in Reference 8. **[**

]

Figure RAI-15.06.05-85.3 shows the liquid flow rate at the top of the SG tubes for each loop. **[**

] As the RCS pressure decreases due to the SG cool down, the ECCS flow rate increases, resulting in the natural circulation resumption. During this period without the natural circulation, the boron-free steam generated in the core is condensed in the SGs tubes, and the deborated condensate accumulates in the loops **[**

]. Slugs of the deborated water rapidly enter the reactor vessel when natural circulation is resumed.

1) **[**

]

2) Condensate in SGs

The amount of condensate in the uphill-side of the SG tubes and downhill-side of the SG tubes is of interest, as shown in Figures **RAI-15.06.05-85.5 and 6**, respectively. **[**

] **Figure RAI-15.06.05-85.7** shows the liquid and vapor flow rate at the SG inlet of broken loop. **[**

]

3) Downcomer Flow

As shown in Figure RAI-15.06.05-85.4, the coolant is either stagnant or slowly flows in the cold leg, and the stratified condition is established there. **[**

]. **Figure RAI-15.06.05-85.8** shows a temporal change of the downcomer collapsed level. **[**

]

(5) Conclusions

From the analysis results, the following conclusions can be made:

1) A boron dilution event is possible under a narrow range of the break size, and the range is [ ].
2) The deborated condensate accumulates in the RCP suction piping, and the amount of the condensate during the SG cool down is large enough to fill the RCP suction piping with the deborated water.
3) The resumption of the natural circulation occurs during the SG cool down and the coolant temperature is above [ ] at that time.
4) [ ]

Therefore, the analysis conditions below are reasonably conservative for the core recriticality evaluation:

[ ]

References:

1. Mitsubishi Heavy Industries, Ltd., Small Break LOCA Methodology for US-APWR, MUAP07013 Revision2, October 2010.
2. Mitsubishi Heavy Industries, Ltd., MHI's 2nd Part Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09041, February 2009.
3. Mitsubishi Heavy Industries, Ltd., Scaling Analysis Report for US-APWR Small Break LOCAs, UAP-HF-10289, October 2010.
4. Mitsubishi Heavy Industries, Ltd., MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1, UAP-HF-09384, July 2009.
5. Mitsubishi Heavy Industries, Ltd., 1st MHI's Response to US-APWR DCD RAI No. 719-5352 Revision 0 (15.06.05), UAP-HF-11106, April 2011.
6. T. Takeda et al., Analysis of the OECD/NEA ROSA Project Experiment Simulating a PWR Small Break LOCA with High-Power Natural Circulation, Annals of Nuclear Energy, 36, pp 386-392, 2009.
7. Mitsubishi Heavy Industries, Ltd., Design Control Document for the US-APWR, Chapter 15 Transient and Accident Analyses, MUAP-DC015 Revision 3, March 2011.
8. Mitsubishi Heavy Industries, Ltd., Small Break LOCA Sensitivity Analyses for US-APWR, MUAP-07025 Revision 3, March 2011.
9. USNRC, Generic Safety Issue 185: Control of Recriticality Following Small-Break LOCAs in PWRs, NUREG-0933.

Impact on DCD

There is no impact on the DCD.

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.

This completes MHI's response to the NRC's QUESTION NO. 15.06.05-85.

Table RAI-15.06.05-85.1 Summary of M-RELAP5 Analysis Results for SBLOCA Boron Dilution

Table RAI-15.06.05-81.2 Natural Circulation Resumption Timing (2.0-in Break)

Figure RAI-15.06.05-85.1 Primary and Secondary Pressure (2-in Break)

Figure RAI-15.06.05-85.2 Core Inlet Temperature (2-in Break)

Figure RAI-15.06.05-85.3 SG Tube Top Liquid Mass Flow Rate (2-in Break)

Figure RAI-15.06.05-85.4 Cold Leg Nozzle Mass Flow Rate (2-in Break)

Figure RAI-15.06.05-85.5 Condensation Mass in Uphill-Side of All SGs (2-in Break)

Figure RAI-15.06.05-85.6 Condensation Mass in Downhill-Side of all SGs (2-in Break)

Figure RAI-15.06.05-85.7 Mass Flow Rate at Broken Loop-A SG Inlet (2-in Break)

Figure RAI-15.06.05-85.8 Downcomer Collapsed Liquid Level (2-in Break)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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US-APWR Design Certification

Mitsubishi Heavy Industries

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APPLICATION SECTION:	**15.6.5**
DATE OF RAI ISSUE:	**3/17/2011**

QUESTION NO.: 15.06.05-86

The inherent boron dilution process can take place during the event of a small break LOCA in the US-APWR. When the water level in the reactor pressure vessel drops below the hot leg inlet and steam only starts flowing to the SGs, the natural circulation in the RCS will cease and will switch to a reflux condenser cooling mode. It is the reflux condenser cooling mode during which boron-depleted condensate is generated within the primary RCS via heat extraction to the secondary side through the SG U-tubes. In this mode, a fraction of the condensate flows from the vertical SG U-tubes toward the pump seal and the remaining fraction of the condensate returns back to the upper plenum via the hot-leg. In the hot leg, the returning condensate and the steam form a counter-current flow pattern. The counter-current flow of condensate and steam in the horizontal section of the hot leg and in the connected bend and inclined piping is possible only under a certain range of flow rates, which are limited by the countercurrent flow limitation phenomenon.

The ratio of the US-APWR core thermal output to that of a current four-loop PWR is higher than the ratio of the hot leg inner diameters. Experimental evidence from PWR hot leg test facilities, including recent geometrically scaled tests at the TOPFLOW test facility, reveals that steam-liquid interaction processes in the horizontal hot leg piping as well as in the elbow and inclined section of the hot leg are described by their own distinct governing characteristics. Considering that the US-APWR hot leg was not sized up proportionately to the reactor thermal power increase when compared to current US PWRs, present the experimental data base that validates the US-APWR small break LOCA methodology for modeling reflux condenser cooling. Describe the relevant scaling methodology along with the scaling results for the US-APWR design. Include consideration of flow conditions and parameters of governing importance for reflux condenser cooling including counter-current flow limitation.

ANSWER:

The M-RELAP5 model of the counter-current flow limitation in the hot leg **[** **]** was validated for the US-APWR small break LOCAs (SBLOCAs), by test data obtained in the UPTF facility. The data is scalable to the existing operational four-loop PWRs

(Ref. 1). Although the US-APWR hot leg piping is not sized proportionally to the reactor thermal power increase compared to the existing four-loop PWRs, the applicability of the M-RELAP5 CCFL model and its validation results were addressed [ ] in MHI's response to RAIs 8.1.4-3 and 8.1.4-11 regarding the M-RELAP5 topical report (Refs. 2 and 3).

The UPTF steam-water test was conducted under 3 bar and 15 bar. [

]

References:

1. Mitsubishi Heavy Industries, Ltd., Small Break LOCA Methodology for US-APWR, MUAP07013 Revision2, October 2010.
2. Mitsubishi Heavy Industries, Ltd., MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09002, January 2009.
3. Mitsubishi Heavy Industries, Ltd., MHI's 2nd Response to the NRC's Request for Additional Information on Topical Report MUAP-07013-P (R0) "Small Break LOCA Methodology for US-APWR", UAP-HF-09512, October 2009.

Impact on DCD

There is no impact on the DCD.

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.

This completes MHI's response to the NRC's QUESTION NO. 15.06.05-86.