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TOKYO, JAPAN

September 5, 2013

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

Attention: Mr. Perry Buckberg

Docket No. 52-021
MHI Ref: UAP-HF-13227

Subject: **MHI's Response to US-APWR DCD RAI No. 1046-7117 (SRP Section 07.08)**

Reference: 1) "Request for Additional Information No. 1046-7117, SRP Section: 07.08 - Diverse Instrumentation and Control Systems, Application Section: MUAP-07014 (Rev 5) - D3 Coping Analysis" dated July 30, 2013.

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. 1046-7117."

Enclosed are the responses to the RAI questions contained within Reference 1.

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 (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 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.

Sincerely,

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, Ltd.

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Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 1046-7117 (Proprietary version)
3. Response to Request for Additional Information No. 1046-7117 (Non-proprietary version)

CC: P. Buckberg
J. Tapia

Contact Information
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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-13227

MITSUBISHI HEAVY INDUSTRIES, LTD.
AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Executive Vice President of Mitsubishi Nuclear Energy Systems, Inc., and have been delegated the function of reviewing Mitsubishi Heavy Industries, Ltd.'s (MHI) 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 document entitled "Response to Request for Additional Information No. 1046-7117" dated September 2013, and have determined that portions of the document 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 approach for the Defense-in-Depth (D3) of the Instrumentation and Control(I&C) system and the unique post-accident response of the US-APWR crediting design features for D3, 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 supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

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 identified as

proprietary 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 D3 approach for the I&C system and the post-accident plant response of the US-APWR to the design features for D3. Providing public access to such information permits competitors to duplicate or mimic the D3 design approach without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced plant safety, and reduced operation and maintenance costs associated with the D3 approach.

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 5th day of September, 2013.

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.

Enclosure 3

UAP-HF-13227
Docket No. 52-021

Response to Request for Additional Information
No. 1046-7117

September 2013
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

09/05/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1046-7117**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**7.8.4.6**
DATE OF RAI ISSUE:	**07/30/2013**

QUESTION NO. : 07.08-35

US-APWR – D3 Coping Analysis MUAP-07014-P(R5)Phase 3 ACRS Subcommittee Review & Clarification Question

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

From the US APWR DCD, Rev. 3, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment," at hot-full power (HFP), Offsite Power case (Case C), the intermediate and large steam line breaks are terminated by OPDT (over power Delta T) and low steam line pressure reactor trips, respectively. MUAP-07014-P(R5), Section 5.1.5, "Steam System Piping Failures Inside and Outside of Containment," does not identify these trips as being available from the DAS protective system. Demonstrate that for a range of intermediate and large steam line breaks at power with offsite power meets the D3 coolablity acceptance criteria. Also, provide the D3 trip setpoints used to mitigate the transient.

ANSWER:

In the US-APWR DCD Ch.15 Section 15.1.5 Steam System Piping Failures Inside and Outside of Containment, at rated power, the increased reactivity causes an increase in core power. For small breaks, the response is similar to the steam flow increase event in that the power will not reach the reactor trip setpoint. For intermediate size breaks, the power increase results in an over power ΔT reactor trip. For large breaks, up to and including the double-ended rupture of a steam pipe, the reactor is tripped on low steam

line pressure, which also causes ESF actuation (including safety injection, main steam and feedwater isolation and emergency feedwater isolation).
#
If a digital common cause failure (CCF) is assumed in the RPS and ESFAS, the above reactor trip and ESF would not be actuated.

Based on the following signals, the Diverse Actuation System (DAS) acts to trip the reactor by opening the motor-generator set supply breakers to interrupt electrical power to the CRDM gripper coils. Turbine trip and closure of all of the main feedwater regulation valves are also actuated by the diverse reactor trip function, which is actuated by the following signals:

- High pressurizer pressure
- Low pressurizer pressure
- Low steam generator water level(*)
  (*) This signal also initiates all emergency feed water pumps.#

#
In addition, the DAS low-low pressurizer pressure initiation signal is available to automatically actuate all of the safety injection pumps. The D3 Coping Analysis Technical Report (MUAP-07014 R5) Table 4.4-1 provides DAS actuation analytical limit.

In the case of a SLB concurrent with a digital CCF occurring from hot full power conditions, pressurizer pressure does not reach the DAS low pressurizer pressure reactor trip setpoint (1840 psia). As described in the D3 Coping Analysis Technical Report (MUAP-07014 R5), the increased reactivity causes an increase in core power, and the core power comes to equilibrium at a new steady state condition. However, DNB is not expected to occur even without RTS/ESF actuation based on the following DNBR evaluation.

(1) Evaluation

Analysis Assumptions

- The initial condition is assumed to be HFP nominal conditions.
- The uniform steam release (common header break) is assumed because the total heat removal is larger than the asymmetric steam release (break between SG and main steam check valve (MSCV) which prevents the reverse flow from other SGs).
- Because no reactor trip is credited, the break area is assumed to be the largest double-ended break area.
- The initial condition uncertainties used in the DCD analysis are not applied. Nominal power, nominal RCS pressure and nominal RCS average temperature are assumed in the analysis.

Plant Transient Analysis (MARVEL-M)

- The plant transient is simulated by MARVEL-M, which calculates the plant parameters such as nuclear power, core inlet temperature and RCS pressure.

[ ]

- The Doppler power coefficient is assumed to be the lower limit described in DCD Chapter 15.
- No reactor trip signals and no ESF actuation signals are assumed in the analysis.

Power distribution evaluation (ANC)

- The three dimensional core power distributions of the first cycle core are evaluated by ANC with the state point boundary conditions calculated by MARVEL-M.

DNBR Analysis (VIPRE-01M)

- The state point plant parameters provided by MARVEL-M are used.
- The axial power distribution calculated by ANC is used.
- The radial power distribution is accounted for by $F_{\Delta H}^{N}$ value calculated by ANC.
- The W-3 correlation is used for the minimum DNBR evaluation.

(2) Results

Figures 07.08-35.1 through 07.08-35.8 show key plant parameter responses for the more severe uniform cooldown case. Figure 07.08-35.9 shows the axial power distribution and Figure 07.08-35.10 shows the axial DNBR distribution.

As shown in Figures 07.08-35.1 and 07.08-35.2, the reactivity and the reactor power increase due to the decrease of RCS temperature. The cooldown rate decreases as a result of decreasing steam pressure; however since the main feedwater control system provides feedwater to compensate for steam flow out the break, the cooldown continues until the plant reaches an equilibrium condition.

Figure 07.08-35.9 shows the axial power distribution evaluated by the ANC at the state points. The bottom-skewed power distributions are the consequence of a large temperature difference between the inlet and outlet of the core due to the large core power.

Figure 07.08-35.10 shows the axial distribution of DNBR in the hot channel. [

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Therefore, the minimum DNBR is above the safety limit even without the RTS and ESF, consistent with the conclusion in the D3 Coping Analysis Technical Report. Because no reactor trip is credited and the event continues until equilibrium conditions are reached, the cases for small and intermediate breaks are bounded by the case for the largest double-ended break area.

Figure 07.08-35.1 Core Reactivity versus Time

Figure 07.08-35.2 Reactor Power versus Time

Figure 07.08-35.3 Core Heat Flux versus Time

Figure 07.08-35.4 RCS Pressure versus Time

Figure 07.08-35.5 Core Average Temperature versus Time

Figure 07.08-35.6 Steam Generator Pressure versus Time

Figure 07.08-35.7 Steam Flow Rate versus Time

Figure 07.08-35.8 Feedwater Flow Rate versus Time

Figure 07.08-35.9 Axial Power Distribution

Figure 07.08-35.10 Axial Distribution of DNBR

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

09/05/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1046-7117**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**7.8.4.6**
DATE OF RAI ISSUE:	**07/30/2013**

QUESTION NO. : 07.08-36

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

MUAP-07014-P(R5), Section 5.6.3 (Radiological Consequences of Steam Generator Tube
Failure)

1. In Table 5.6.3-1 – the time required for Operator Actions does not add up. The total time required for manual reactor trip on DHP is listed as 15 minutes; however, it appears it should equal 16.5 minutes. If the operator action time is 16.5 minutes, how is the D3 dose bounded by the DCD dose analysis which uses a 15 minute operator action time?

2. On page 5-60 of MUAP-07014 (R5), there are statements that the time available for main steam isolation should be more than 30 minutes and that the DCD Chapter 15 assumption of the manual steam isolation with 30 minutes is conservatively applied. From DCD Subsection 15.6.3.4.3, item 1.c., page 15.6-25, "Isolating the ruptured steam generator," the main steam isolation valve is closed 1200 seconds after SGTR initiation. While 1200 seconds (20 minuutes) is within 30 minutes as stated in MUAP-07014, staff cannot locate the statements in Chapter 15 that the D3 report references. It is not clear what the SG isolation time limit is. Please clarify this inconsistency between the D3 report and DCD Subsection 15.6.3.

ANSWER:

1. Table 5.6.3-1 in the MUAP-07014-P (R5), Section 5.6.3 also shows the total time required, 15 minutes is the time from event initiation. The 15 minutes is a cumulative time and therefore includes 1.5 minutes necessary for some operator actions.

In MHI's response to US-APWR DCD RAI 830-6056 for Chapter 7 Question 07-08 Branch Technical Position-5 (UAP-HF-11412, dated November 29, 2011), MHI provided a table that showed that the total time required for manual reactor trip on DHP (15 minutes) includes: Operators move to DHP (0.5 min), Select special event EOP (0.5 min), and Operators energize DHP with Permissive Switch for DAS HSI (0.5 min). MHI previously committed to revising Table 5.6.3-1 to match the table shown in that RAI response.

2. The analysis in DCD Section 15.6.3 assumes that main steam isolation occurs at 20 minutes from event initiation. For an SGTR concurrent with a CCF in the digital I&C system under realistic conditions, the time available for main steam isolation could be more than 30 minutes. Therefore, the DCD Ch.15 assumption of the manual main steam isolation within 30 minutes (i.e., 20 minutes from event initiation) is conservatively applied to the time available in the D3 coping analysis.

In MHI's response to US-APWR DCD RAI 830-6056 for Chapter 7 Question 07-08 Branch Technical Position-5 (UAP-HF-11412, dated November 29, 2011), MHI provided a table that showed that the elapsed time to completion for manual closure of the main steam isolation valve of the ruptured SG on DHP is 20 minutes. MHI previously committed to revising Table 5.6.3-1 to match the table shown in that RAI response.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
MHI previously committed to revising MUAP-07030-P Table 5.6.3-1 in the response to RAI 830-6056 Question 07-08 Branch Technical Position-5. Those changes are also applicable to this RAI (1046-7177). There is no additional impact on the technical / topical reports from this RAI.

This completes MHI's response to the NRC's question.