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TOKYO, JAPAN

July 18, 2012

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-12199

Subject: MHI's Response to the Questions at the US-APWR ACRS Subcommittee Meeting on May 27, 2011 Regarding DCD Chapter 5 (REACTOR COOLANT AND CONNECTING SYSTEMS)

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the responses to the questions that were discussed during the US-APWR ACRS Subcommittee meeting on May 27, 2011 regarding DCD Chapter 5 (REACTOR COOLANT AND CONNECTING SYSTEMS).

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,
Director- APWR Promoting Department
Mitsubishi Heavy Industries, Ltd.

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Enclosure:

1. Affidavit of Yoshiki Ogata

2. MHI's Response to the Questions at the US-APWR ACRS Subcommittee Meeting on May 27, 2011 Regarding DCD Chapter 5 (REACTOR COOLANT AND CONNECTING SYSTEMS) (proprietary version)

3. MHI's Response to the Questions at the US-APWR ACRS Subcommittee Meeting on May 27, 2011 Regarding DCD Chapter 5 (REACTOR COOLANT AND CONNECTING SYSTEMS) (non-proprietary version)

CC: J. A. Ciocco
J. Tapia

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-12199

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 document entitled "MHI's Response to the Questions at the US-APWR ACRS Subcommittee Meeting on May 27, 2011 Regarding DCD Chapter 5 (REACTOR COOLANT AND CONNECTING SYSTEMS)" dated July, 2012 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 information 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 Fluid System Engineering. Providing public access to such information permits competitors to duplicate or mimic the Fluid System Engineering information without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced US-APWR Fluid System Engineering development costs associated with the pH Control System.

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 18th day of July, 2012.

Y. Ogata

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

Enclosure 3

UAP-HF-12199
Docket No. 52-021

MHI's Response to the Questions at the US-APWR ACRS Subcommittee Meeting on May 27, 2011 Regarding DCD Chapter 5 (REACTOR COOLANT AND CONNECTING SYSTEMS)

July 2012
(Non Proprietary)

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **05**

CHAPTER TITLE: **REACTOR COOLANT AND CONNECTING SYSTEMS**

DATE OF MEETING: **05/27/11**

QUESTION: ACRS 05-1 (Section 5.2.2)

Pressurizer relief valve capacities are sized for design basis events, but ATWS events are beyond design basis events which can result in severe pressure transients. Has MHI performed any best-estimate ATWS analyses and, if so, how do the pressure transients compare to the pressurizer safety valve sizing? Are the pressurizer safety valve capacities sufficient to handle potential ATWS events?

ANSWER:

Since the US-APWR has the diverse actuation system (DAS) including a diverse scram system (DSS) function, the frequency of ATWS occurrences is extremely low. Therefore, the pressurizer safety valve sizing is not designed in consideration of a long-term ATWS event that would result in the dry out of the steam generators. Therefore, MHI has not performed any best estimate ATWS analyses for the US-APWR Chapter 15. However, the pressurizer safety valve sizing is designed in consideration of conditions where reactor trip does not occur immediately on demand, or is delayed for a short period, but before the SG dries out.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **05**

CHAPTER TITLE: **REACTOR COOLANT AND CONNECTING SYSTEMS**

DATE OF MEETING: **05/27/11**

QUESTION: ACRS 05-2 (Section 5.2.3)

MHI is requested to justify not specifying <0.03% carbon for all stainless steels. Even though the APWR dissolved oxygen is low, ACRS believes to be better to specify the low carbon requirement for all 316 SS. The argument that will be applied in stagnant regions is inconsistent with specifying low carbon for the pressure vessel cladding. MHI needs to respond that the reason for not requiring <0.03% carbon everywhere.

ANSWER:

In MHI PWRs, stainless steels with equal to or less than 0.05% carbon content have been the standard practice for the past 40 years, and PWRs including MHI plants have excellent performance for the stress corrosion cracking, so we have decided to use equal to or less than 0.05% carbon stainless steels except when there are local areas where flow stagnation may be present resulting in dissolved oxygen content greater than 0.10 ppm.

Mechanical properties of low carbon grade stainless steels are lower than that of normal grade stainless steel. If we use low carbon grade (L-grade) stainless steel, wall thickness of each part of the components will be much thicker than the current design. For cladding materials, severe mechanical properties are not specified because the cladding is not a pressure boundary.

MHI recognizes that there are low-carbon and high-nitrogen (LN-grade) materials, with mechanical properties equal to normal grade stainless steels. However in MHI PWR plants, these materials are not used because of the resultant increase in wall thicknesses.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **05**

CHAPTER TITLE: **REACTOR COOLANT AND CONNECTING SYSTEMS**

DATE OF MEETING: **05/27/11**

QUESTION: ACRS 05-3 (Section 5.2.5)

Adding T/Cs downstream of valves to detect leakage is good; but in most cases there are two valves in series – it would be good to instrument the pipe between the two valves so that the operator knows if the 1st valve is leaking. The RV flange O-rings have leakage monitoring between the two O-rings as well as downstream of them. Does MHI has any thought for the detection of leakage through the first valve to inform operators of the first leakage as reactor vessel head seal and O rings give early warning of leakage through the first seal?

ANSWER:

The purpose of leak detection system is to detect leakage from the boundary between Class 1 equipment and other equipment. The two valves are both Class 1 valves, so the T/Cs are installed downstream of the outer valve. This approach is also used on the R/V flange O-ring leak detection system, for which the first (inside) O-ring is the Class 1 boundary. The leak detection system is installed on the outer side of the Class 1 boundary, so this design is consistent with other leak detection systems.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: 05

CHAPTER TITLE: REACTOR COOLANT AND CONNECTING SYSTEMS

DATE OF MEETING: 05/27/11

QUESTION: ACRS 05-4 (Section 5.4.1.1)

Did the fracture mechanics analysis for the RCP flywheel account for different orientation associated with the flywheel loading? And the "probabilistic fracture mechanics used to establish inspection intervals"? What probability is applied to the various overspeed conditions?

ANSWER:

Flywheel loading is based on centrifugal force generating the circumferential tensile stress during RCP flywheel operation. Therefore, fracture mechanics analysis was carried out for the prospective crack plane emanating from the flywheel keyway. MHI provided the evaluation results in Technical Report MUAP-07035 Rev.0 and RAI response of RAI 738-5663, UAP-HF-11151, dated on May 26, 2011.

Probabilistic fracture mechanical analysis was utilized to establish the RCP flywheel inspection interval. The detailed calculational results for the RCP flywheel are shown in Technical Report MUAP-09017 Rev.0, dated July 2009. These results indicate the computed failure probabilities were very small (on the order of [ ] cumulative in 60 years). Therefore, the risk due to failure of the RCP flywheel within the 20 year inspection interval is negligibly small.

The predicted loss of coolant accident (LOCA) is applied for RCP overspeed condition in accordance with Regulatory Guide (RG) 1.14 recommendations. The US-APWR uses leak-before-break (LBB) methodology. Therefore, failure of RCS branch lines are the limiting break sizes in the US-APWR design. Two postulated LOCAs were evaluated for the reactor coolant system (RCS). These are a Hot Leg Branch Line break at the 10 inch Schedule 160 Residual Heat Removal (RHR) / Safety Injection (SI) line nozzle and Cold Leg Branch line break at the 8 inch Schedule 160 RHR return line nozzle.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **05**

CHAPTER TITLE: **REACTOR COOLANT AND CONNECTING SYSTEMS**

DATE OF MEETING: **05/27/11**

QUESTION: ACRS 05-5 (Section 5.4.1.2)

Are there any automatic trips of the reactor coolant pumps? If there is no automatic trip, what is the pump response to loss of all components cooling water if it remains running? If there is an automatic trip, provide set points and the specific signals which are adequate margin to protect from bearing damage or seal damage. What are the effects from loss of all components cooling water to the pump for both cases?

ANSWER:

There is no automatic trip of the reactor coolant pump (RCP) except from the ESF actuation signal (S signal). If the cooling flow to the thermal barrier decreases, the operator can be notified by the low flow rate alarm in the thermal barrier cooling line. In such a situation, the RCP can continuously operate because cooling is supplied from the seal water that is injected to the RCP. On the other hand, if seal water injection is lost, the RCP also can operate when the thermal barrier cooling is kept.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **05**

CHAPTER TITLE: **REACTOR COOLANT AND CONNECTING SYSTEMS**

DATE OF MEETING: **05/27/11**

QUESTION: ACRS 05-6 (Section 5.4.7)

Figure 6.3-4, the safety injection pumps have a rated flow of 1540 gpm at a nominal discharge pressure of 710 psig (1640 feet developed head) and 1400gpm at 900 psig. Both of those flow rates are higher than the capacity of the CS/RHR pump suction relief valves. How do the CS/RHR pump suction relief valves provide overpressure protection if the safety injection pumps are started spuriously when the pressurizer is water solid?

ANSWER:

The design point for the safety injection pump (SIP) is based on the Large Break LOCA at atmospheric pressure of the RCS. If the RCS pressure is high, the flow rate of SIP is decreased based on the generic characteristic of centrifugal pump. When an overpressure event occurs at low temperature (the pressurizer is water solid), the RCS pressure is increased by spuriously actuated SIP to the set point of CS/RHR pump suction relief valve. Therefore, SIP flow rate is the corresponding pump flow rate at the relief valve set point pressure based on the pump curve. CS/RHR pump suction relief valve capacity is designed based on this flow rate, and Technical Report MUAP-09016 shows the validity of the overpressure protection system.

Technical Report MUAP-09016 shows that the CS/RHR pump suction relief valve is sized to relieve the maximum volumetric SIP flow rate that the system can deliver at the relief valve setpoint.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **05**

CHAPTER TITLE: **REACTOR COOLANT AND CONNECTING SYSTEMS**

DATE OF MEETING: **05/27/11**

QUESTION: ACRS 05-7 (Section 5.4.7)

If the RHR pumps are air bound, how does US-APWR maintain for long term cooling?

ANSWER:

If the loss of RHR function would occur, the following countermeasures are provided:

- Heat removal via SGs using motor-driven EFW pumps and MSDVs (SG reflux cooling)
- Water injection by the charging pump
- Water injection by the safety injection pump
- Water injection by gravity from the Spent Fuel Pit

The effective countermeasure depends on the plant conditions such as the RCS vent path and RCS pressure. The scenarios for which these countermeasures are effective are described in DCD Chapter 19, Subsection 19.1.6.1.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: 05

CHAPTER TITLE: REACTOR COOLANT AND CONNECTING SYSTEMS

DATE OF MEETING: 05/27/11

QUESTION: ACRS 05-8 (Section 5.4.12)

In DCD, motor operated DVs are powered by Class 1E power with alternate power. Please confirm the specific power supplies for those valves. If power supplies are AC, why aren't they DC in order to be operable in a station blackout situation?

ANSWER:

The motor operated DVs are driven by Class 1E ac power and Class 1E ac power is backed-up by non-Class 1E alternate ac power; although no battery back-up is provided.

Use of DC driven valves is a potential option to improve safety during SBO, however this design option was not adopted for the following reasons.

The US-APWR has two alternate ac (AAC) gas turbine generators (GTGs), which is the design feature to cope with a station blackout (SBO) situation. If any of the four Class 1E GTGs is not available, operators manually connect one of the two AAC GTGs to the Class 1E ac buses that supply power to the depressurization valves (DVs). This design satisfies the NRC requirement 10 CFR 50.63, loss of all alternating current power.

In addition, operation of the DVs under the SBO situation has limited benefit for improving plant safety for the following reasons:

For prevention of core damage

The valves are used to depressurize the reactor coolant system (RCS) and then safety injection (SI) pumps are also used to inject coolant into the RCS (feed and bleed operation). In the SBO situation, no SI pumps powered by Class 1E ac are available and there is no effective countermeasure to inject coolant for RCS make-up, regardless of operability of the DVs.

For maintaining containment integrity

If the DVs can be used in the SBO situation, then the probability of temperature induced steam generator tube rupture (TI-SGTR) in conjunction with the SBO is reduced. Therefore a sensitivity analysis using a zero TI-SGTR probability SBO has been

performed. The resulting LRF and CCFP using the zero TI-SGTR probability are summarized in Table 1 below.

There is a small decrease in LRF and CCFP using a zero TI-SGTR probability during an SBO with depressurization.

Note that, the US-APWR PRA results in US-APWR DCD Rev.3 will be updated using the revised TI-SGTR probability for SBO described in US-APWR RAI#872-6144 19-561. The revised TI-SGTR probability is about three times larger than the probability used in DCD Rev.3. Therefore, the result of the sensitivity analysis is compared with the result of US-APWR RAI#872-6144 19-561, because it maximizes effect of DV operation during a SGTR.

Table 1 Comparison between RAI#872-6144 19-561 and Sensitivity Analysis Results

	CDF DCD Rev.3	LRF and CCFP US-APWR RAI#872-6144 19-561		LRF and CCFP Sensitivity Analysis ACRS Q&A Ch05		Delta in LRF and CCFP
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.16E-07	0.113	1.02E-07	0.099	-12.25%
Fire	8.60E-07	1.96E-07	0.227	1.82E-07	0.211	-7.05%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	-0.04%
Total	2.78E-06	4.68E-07	0.168	4.40E-07	0.158	-6.01%