MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 25, 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-13313

Subject: Transmittal of the Responses to October 1, 2013 ACRS Comments on GSI-191/LTCC

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") an official document entitled 'Responses to October 1, 2013 ACRS Comments on GSI-191/LTCC'.

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 response (Enclosure 2), a copy of the non-proprietary version of the response (Enclosure 3), and the Affidavit of Tatsuya Hashimoto (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 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,

T. Hashimoto for

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

Enclosures:

1. Affidavit of Tatsuya Hashimoto
2. Responses to October 1, 2013 ACRS Comments on GSI-191/LTCC (proprietary)
3. Responses to October 1, 2013 ACRS Comments on GSI-191/LTCC (non-proprietary)

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NRO

CC: J. A. Buckberg
J. Tapia

Contact Information

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-13313

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Tatsuya Hashimoto, being duly sworn according to law, depose and state as follows:

1. I am Manager, US-APWR Project of Global Nuclear Project Department, of Mitsubishi Heavy Industries, LTD. (MHI), 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 "Responses to October 1, 2013 ACRS Comments on GSI-191/LTCC", dated December 2013, 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 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 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 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 GSI-191-related design. Providing public access to such information permits competitors to duplicate or mimic the long-term core cooling design information without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced US-APWR design 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 25th day of December, 2013.

T. Hashimoto

Tatsuya Hashimoto,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, LTD.

Enclosure 3

UAP-HF-13313
Docket No. 52-021

Responses to October 1, 2013 ACRS Comments on GSI-191/LTCC

December 2013

(Non-proprietary)

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 1

Check the design specification for the refueling cavity drain line and provide an explanation for why the line is not expected to be plugged up / clogged by debris.

ANSWER:

MUAP-08001 "US-APWR Sump Strainer Performance" Section 3.7.1 identifies two possible choke points in the containment spray/blowdown return pathway for the US-APWR – the refueling cavity drain lines and the RWSP overflow lines. In both cases, the conclusion is that the lines will not be clogged by debris. The rationale for this conclusion with respect to the refueling cavity drain line is further explained below.

1. Debris Source for the Refueling Cavity Drain Line
   For the US-APWR, the majority of debris will be generated in the SG compartments. There is no direct flow path between the SG compartments and the opening of the refueling cavity drain lines (two 8" lines), which are located at the bottom of the refueling cavity. Therefore, latent and miscellaneous debris are the primary type of debris that may be expected to reach the refueling cavity. However, grating with an opening size of 1" by 4" is arranged above the 8" diameter refueling cavity drain lines as shown in Figure 3-9 of MUAP-08001. This grating ensures that only "fine" and "small" latent fiber debris and particulate latent debris, which are smaller than 8" diameter, could reach the refueling cavity drain lines.

2. Refueling Cavity Drain Line Configuration
   The refueling cavity drain lines have three upper sections one of which is located at the fuel container upending area on EL. 29'-10" and the other two are located at the upper reactor internals storage area on EL 19'-14". The line from the fuel container upending area and one of the lines from the upper reactor internals storage area merge to one line. The other line from the upper reactor internals storage area does not merge with the others. In other words, both drain lines are independently arranged with the lower ends draining to the Header Compartment area. (See Figure 1-1) The total length of the drain lines are approximately 60 ft and 42 ft, respectively, and both lines are arranged with approximately 1/100 slope.

3. Flow Rate Estimation of the Refueling Cavity Drain Line
   To compare the flow velocity of the relevant line during the recirculation phase with the settling velocity and incipient tumbling velocity of fiber debris, the minimum flow velocity was calculated with the following assumptions.

- Two trains of the containment spray pumps are operated.
- Only containment spray water is considered for the reactor cavity water source.
- Although the flow regime is actually separated flow in the drain line, single phase flow is assumed for conservatism.

The estimated minimum flow velocity is 0.85 ft/sec. With reference to NEI 04-07, the maximum fiber settling velocity for NUKON™ is 0.41 ft/sec for 6 in size, and its incipient tumbling velocity is 0.16 ft/sec maximum for a 1/4" by 1/4" clump. The 0.85 ft/sec estimated minimum flow velocity of the drain line is much higher than that of the NUKON™ fiber settling velocity and incipient tumbling velocity. Therefore, MHI concludes that there is a very small chance for the drain line to be clogged by debris in the line.

4. Conclusion

The refueling cavity drain lines are not expected to be clogged by debris for the following reasons.

- As discussed in the item 1 above, the debris size which could reach those drain lines is expected to be smaller than the corresponding drain line inner diameter.
- As discussed in the item 2 above, the drain lines are relatively short and are sloped.
- As discussed in the item 3 above, the estimated minimum flow velocity of the drain line is much higher than the NUKON™ fiber settling velocity and incipient tumbling velocity.

Additionally, the hold-up volume in the refueling cavity has been conservatively calculated assuming a case where one line out of two does not function properly.

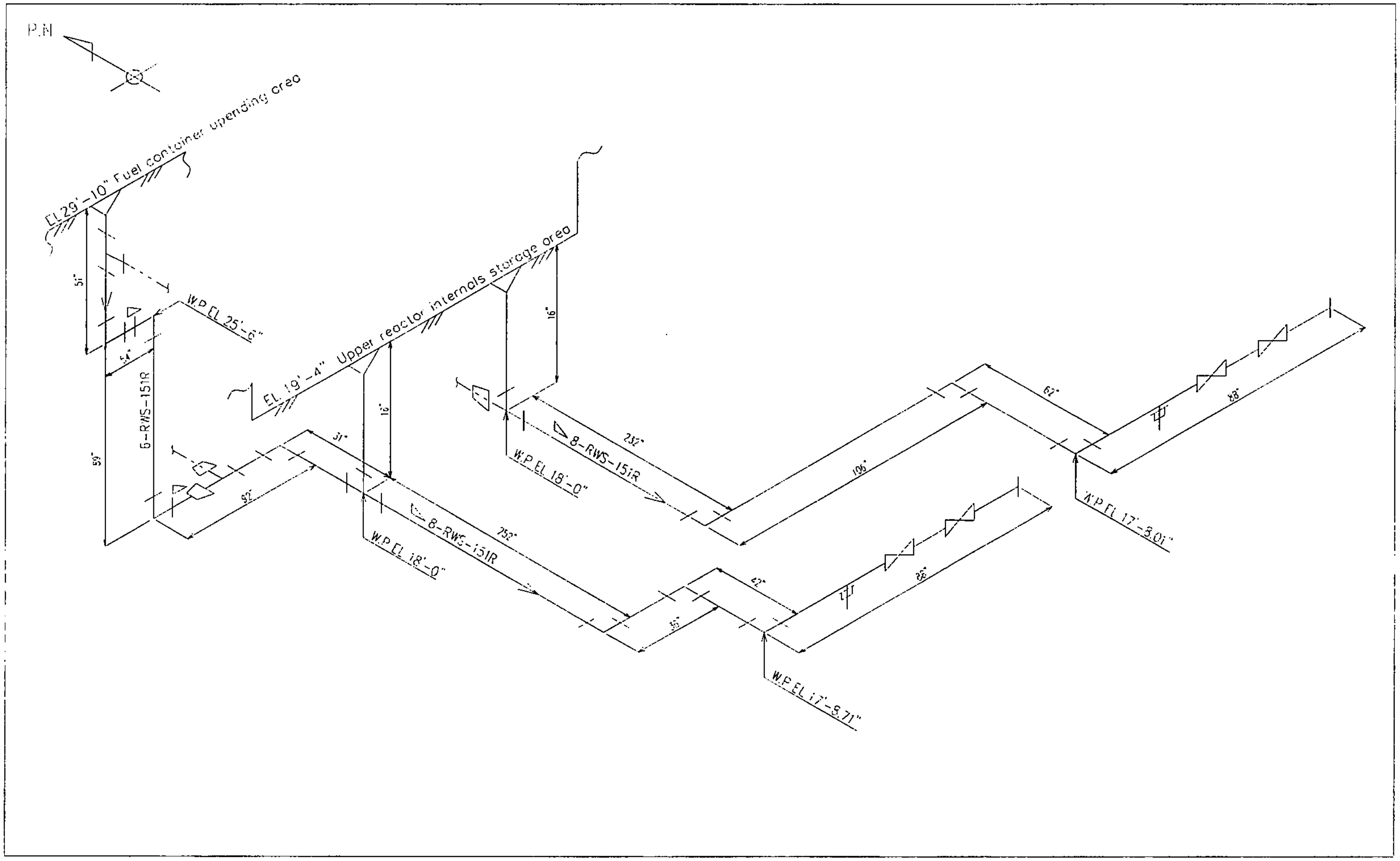

Figure 1-1 Isometric Drawing of the Refueling Cavity Drain Line

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: 6

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 2

Is there any height / depth requirement (minimum or maximum) for the reactor cavity in terms of what is used in the PRA?

ANSWER:

The design requirement for the reactor cavity depth is described in DCD Section 19.2.3.3.3 discussion about the core debris coolability. It is stated as "Reactor cavity depth is also designed to provide a sufficient degree of debris break-up due to interaction of molten core and coolant water for better coolability. The depth is equal to or greater than 20 ft from the bottom of the reactor vessel."

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 3

Perform a literature search of existing strainer testing to determine if anyone has performed tests at higher temperatures (200°F-250°F). Determine whether existing data support / validate that the tests performed by MHI provide an upper bound on the realistic plant temperature conditions.

ANSWER:

JNES (Japan Nuclear Energy Safety Organization) and CREIPE (Central Research Institute of Electric Power Industry) in Japan executed many strainer head loss tests for BWR plants and PWR plants between 2005 and 2010. The final report, 10 GENNETSUHOU-0006, was published in Japanese in October 2010. However, MHI confirmed that all of the tests were executed at 140°F (60°C).

MHI believes that the ACRS member's questioning was related to ensuring that the test temperature was representative of the actual plant conditions. The MHI test conditions were performed at a temperature that is lower than the expected recirculation fluid temperature. However, MHI considers this to be conservative because there is no chemical debris precipitation at temperatures above 150°F. MHI selected a test temperature representative of worst case debris conditions, which would include chemical debris.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: 6

CHAPTER TITLE: ENGINEERED SAFETY FEATURES

DATE OF MEETING: 10/01/2013

QUESTION: Item 4

What is the physical basis for the autoclave test results for aluminum concentration being considered acceptable / explain why MHI's result is ok when it does not follow the expected trend. [ ]

ANSWER:

( )

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 5

What correlation is used in WCOBRA/TRAC for the dispersed flow?

ANSWER:

The dispersed droplet flow regime occurs when the continuous liquid field becomes completely entrained. Interfacial heat transfer in then due to droplet only.

The interfacial heat transfer coefficient to superheated vapor is given by[ ]

[ ]

where the mass transfer number B from Yuen and Chen is

$$B = \frac{H_v - H_f}{H_{fg}}$$

For the interfacial heat transfer coefficient to superheated liquid, a constant value is assumed:

$$h_{i,ve} = 27.8 \frac{BTU}{ft^2 s ^\circ F}$$

The interfacial heat transfer coefficient to subcooled liquid droplets is calculated using the equation by Andersen (1973):

[ ]

For subcooled vapor, a large interfacial heat transfer coefficient is assumed:

$$h_{i,SCV} = 2780 \frac{BTU}{ft^2 s ^\circ F}$$

k_v: thermal conductivity of vapor field
k_f: thermal conductivity of saturated liquid
D_d: droplet diameter
Re_d: droplet Reynolds number
Pr_v: vapor Prandtl number
H_v: enthalpy of vapor
H_f: enthalpy of saturated liquid
H_{fg}: enthalpy of vaporization
R_d: droplet radius

Reference

Andersen, J. G. M., "REMI/HEAT COOL, a Model for Evaluation of Core Heat Up and Emergency Core Spray Cooling System Performance for Light Water Cooled Nuclear Power Reactors" Report 296, RISO National Lab Denmark, 1973

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: 6

CHAPTER TITLE: ENGINEERED SAFETY FEATURES

DATE OF MEETING: 10/01/2013

QUESTION: Item 6

Annotate the table on Slide 33 to include the flow velocity, DP criteria and k-values to allow confirmation of the calculation process used to come up with the DP acceptance criterion for each CIB case.

ANSWER:

A detailed description of the calculation process to obtain the K-value criteria, which directly indicate the geometric effect of debris accumulation, was previously provided as part of MHI's response to an ACRS question about the CIB test acceptance criteria in MHI Letter UAP-HF-13016 dated January 25, 2013 (ML13028A406).

Table 6-1 summarizes the core inlet flow rates and corresponding DP-values calculated in the WCOBRA/TRAC (WC/T) confirmatory analysis, which were used to estimate the K-value criteria. Conservativeness of the current DP criteria for the core inlet blockage tests is confirmed from the comparison of K-values between "Test Criterion" and "WC/T Calc. Result".

Table 6-1 Core Inlet Flow Rates

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 7

What is the source of apparent discoloration on the fuel rods shown in the CIB report figures? Is it due to the drain down?

ANSWER:

The figures included in the technical report are taken after drain down. A lab technician visually confirmed that in all test cases there were no chemical debris accumulation on the surface of the fuel assembly, except at the bottom nozzle and grids.

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 8

How is the Al concentration measured (methodology) in the chemical effects test?

ANSWER:

After the dissolution treatment, both the dissolved and undissolved aluminum were simultaneously measured using ICP-AES (Inductively Coupled Plasma – Atomic Emission Spectroscopy).

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

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CHAPTER TITLE:	**ENGINEERED SAFETY FEATURES**
DATE OF MEETING:	**10/01/2013**

QUESTION: Item 9

The ACRS noted their traditional longstanding disagreement with the staff over the use of containment accident pressure. The ACRS SC has stated that they do not wish to accept the use of CAP credit for new designs, even though NRC regulations allow its use. The staff's SE and ACRS presentation stated approval of MHI's methodology. Through various discussions, the ACRS SC members requested justification for MHI's statement "The design change required to strictly meet the wording stated in RG 1.82 for high temperature condition would be impractical or otherwise negatively affect overall system health." Specifically, how could installing different pumps, different heat exchangers or increasing water levels available negatively affect overall system health?

ANSWER:

MHI understands the ACRS position that with an existing plant there are clear arguments for why certain design changes to eliminate containment accident pressure credit are not practical. MHI also understands the ACRS' expectation that with a new plant the vendor should be able to change the design to eliminate CAP credit. However, at some point in the design certification process it actually become impractical to implement design changes that cascade into small updates to a rather large portion of the otherwise complete design. This economic impact is especially difficult to justify when the current design meets the current applicable regulatory requirements.

MHI's intent when stating that safety implications would "negatively affect overall system health" was an acknowledgement that at this stage of the licensing process for the US-APWR significant design changes, such as installing different pumps or increasing water levels, could have huge impacts on the overall plant design. Changing the plant to eliminate the use of CAP credit at this stage of the design is not considered rational in terms of the economic and structural impacts.

The following information is provided in order to aid the ACRS understanding of how MHI arrived at their current design.

<u>Heat Exchanger Capacity</u>:
Heat exchanger capacity would have to be increased significantly to maintain sump water temperature below approximately 212 °F in the short-term. During an accident, CCW (Component Cooling Water) temperature will be increased, so the differential temperature between the RWSP water and the CCW is not so large. In the case that more heat needs to be removed with a small differential temperature, the UA value of the heat exchanger would be huge.

The layout and the building would have to be significantly re-designed to accommodate significantly larger heat exchangers than the one's utilized in the current US-APWR design. This large increase in heat exchanger size and supporting structure re-design would be impractical considering that the current systems have sufficient capacity to meet the safety requirements.

Pump Elevation:
The static head could be increased by lowering the pump installation elevation (approx. 45 ft. lower than the current elevation). The resulting increase in overall building height would negatively impact the seismic design and was thus considered impractical as well.

Water Volume:
The RWSP water volume would need to be increased significantly to provide additional cooling water inventory. The differential temperature between the initial and the peak RWSP fluid temperature is approximately 136°F (120°F vs. 256°F). The differential temperature between the initial RWSP temperature and 212 °F is 92°F. Therefore, approximately 1.5 times as much volume of RWSP water is needed to maintain the RWSP water temperature below 212°F. Changing the RWSP from 84,750 ft^3 to 127,125 ft^3 would significantly impact the overall layout and structural design in the containment. Additionally, the increase in height would negatively affect the overall building height and the seismic design.

Pump NPSH required:
A different style of pump may be designed such that it meets the NPSH available flow, pressure, and reliability requirements. However, use of a different pump may entail a redesign and reanalysis of the ECCS system, due to the magnitude of the additional NPSH available. Such a redesign may induce additional risk and is thus not desirable as this point in the overall licensing process.

Currently, there is no regulation against the use of containment accident pressure. As stated above, it is MHI's position that it is not reasonable or practical to apply these potential design changes to eliminate CAP credit at this time. Furthermore, the current design utilization of an in-containment emergency cooling water source was chosen to maximize overall system health, which would be negatively impacted by the proposed design changes described above.

In addition, the proposed methodology at high temperatures conservatively bounds the actual containment pressure for all accident scenarios. Therefore, MHI considers the use of the sump vapor pressure for containment pressure in the NPSH evaluation to provide adequate and reasonable assurance that the ECCS system pumps will be able to fulfill their functional requirements with respect to available NPSH.

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 10

In DCD Chapter 6, when you calculate the minimum containment pressure, you maximize passive heat removal. The ACRS needs to be sure that MHI identified all of the possible modes of passive heat removal before accepting MHI's claim that they have determined the lowest possible drop in containment pressure. As a result, the ACRS requested that MHI provide additional detail, beyond what is in the DCD, to allow the ACRS to confirmation that the minimum pressure calculation in Ch 6 adequately addressed all possible heat removal paths (heat sink, fluid flows, operator actions such as the restart of the containment fan coolers, etc.)

ANSWER:

The possible passive and active heat sinks in the primary containment system of the US-APWR are shown below.

Passive heat sinks

- Containment structures, such as containment shell, inner concrete, supporting, piping, and electrical instruments and other various components. (Refer to Subsection 6.2.1.5.7 and Table 6.2.1-30 of the DCD, Table 1 in this response)
- Spillage of subcooled water from the primary system. (Refer to Subsection 6.2.1.5.6 of DCD)
- RWSP water with lowest temperature and maximized surface cooling. (Refer to Subsection 6.2.1.5 of DCD)
- Containment vapor (Refer to Subsection 6.2.1.5.3 of DCD)
- Heat transfer with outside atmosphere via containment shell or cylinder. (Refer to Subsection 6.2.1.5.3 of the DCD)

Active heat sinks

- Containment spray (safety) (Refer to Table 6.2.1-29 of the DCD).
- Containment fan cooler system (non-safety)
- CRDM cooling system (non-safety)
- Reactor cavity cooling system (non-safety)
- Containment exhaust fan (non-safety)
- Nominal purging through the HVAC system up to the containment isolation valve closure. (Refer to Subsection 6.2.1.5.9 of the DCD)

As described in DCD Section 6.2.1.5, all the passive heat sinks and the containment spray are considered in the minimum containment pressure analysis. Meanwhile, the active heat sinks other than the containment spray are not considered because the HVAC systems in the containment (Containment Fan Cooler System, CRDM Cooling System, Reactor Cavity Cooling System, and Containment Exhaust Fan) are automatically terminated by the ECCS actuation signal. Therefore, the effects of circulating air using these fans does not need to be considered. In addition, restart of these HVAC systems requires operator action to reset ECCS actuation signal. Since such action is not indicated in the emergency response guidelines, there is very low possibility of inadvertent restart of these HVAC fans by operators and hence it is excluded from the initial condition of the design accident analysis.

Moreover, the containment spray temperature is assumed to be 32°F, which is the freezing temperature. As for steam condensation, the combined Tagami-Uchida correlation multiplied by 4 is conservatively assumed in the analysis. Therefore, MHI's minimum containment pressure analysis is judged to have sufficient overall conservativeness to bound other unknown heat sinks.

Due to the above reasons, MHI believes the assumptions of the heat sinks considered in the minimum containment pressure analysis are appropriate and sufficiently conservative.

Table 10-1 Passive Heat Sinks for Minimum Containment Pressure (1/2)
(Same as Table 6.2.1-30 of the DCD)

Passive Heat Sinks	Heat Transfer Area (ft^2)	Material	Thickness (in)
(1) Containment Dome	36,710	Carbon Steel Concrete	0.257 44.1
(2) Containment Cylinder	73,170	Carbon Steel Concrete	0.400 53.9
(3) Thick Concrete - Internal Separation Walls, Connection Paths, C/V Reactor Coolant Drain Pump Room, Header Compartment, SG Compartments	40,944	Concrete	31.7
(4) Thin Concrete - Internal Separation Walls, Header Compartment, Letdown Hx Room, Regenerative Hx Room	19,430	Concrete	7.54
(5) Lined Concrete (Stainless Steel) - Web Plate, Refueling Cavity Walls, RWSP Inner Walls	27,342	Stainless Steel Carbon Steel Concrete	0.118 0.472 45.6
(6) Lined Concrete (Stainless Steel) - Web Plate, Refueling Cavity Floor, RWSP Floor and Ceiling	282	Stainless Steel Carbon Steel Concrete	0.118 0.197 22.6
(7) Lined Concrete (Carbon Steel, Thick) - Primary Shield Walls, Secondary Shield Walls, Header Compartment, C/V Reactor Coolant Drain Tank Room, Pressurizer Compartment, Deck Plates, Reactor Cavity Walls, SG Compartments	162,994	Carbon Steel Concrete	0.549 18.9
(8) Lined Concrete (Carbon Steel, Thin) - Deck Plates	162	Carbon Steel Concrete	0.311 7.08
(9) Component (Carbon Steel Thickness greater equals 2-inch) - Equipment Hatch, Air Lock, Accumulators, SG Supports, Level Switch	10,663	Carbon Steel	3.07
(10) Component (Carbon Steel Thickness between 2-inch and 1.2-inch) - Vents, Reactor Vessel Supports, Polar Crane, RCP Lower Bracket, RCP Supports	24,877	Carbon Steel	1.51
(11) Component (Carbon Steel Thickness between 1.2-inch and 0.4-inch) - Air Lock, Accumulator Column Supports, Excess Letdown Hx, Refueling Machine Rail, Fuel Transfer System, Piping Supports, Covering Steel, Ring Guarder, Vents, NIS Electrical Horn, ITV Instruments, SG Supports, Pressurizer Supports, RCP Upper Bracket, RCP Flame, Letdown Hx	186,943	Carbon Steel	0.472

Table 10-1 Passive Heat Sink for Minimum Containment Pressure (2/2)
(Same as Table 6.2.1-30 of the DCD)

Passive Heat Sinks	Heat Transfer Area (ft^2)	Material	Thickness (in)
(12) Component (Carbon Steel Thickness between 0.4-inch and 0.08-inch) - C/V Reactor Coolant Drain Tank Column Supports, Excess Letdown Hx Column Supports, Refueling Machine, Duct Supports, Duct Connection Flanges, HVAC Units, Fans, Connecting Boxes, I/C Piping Supports, Cable Tubes, Penetration Boxes, Electrical Boards, Trans, Motors, Luminaries, I/C Supports, Electrical Boxes, I/C Racks, Stairways, RCP Duct, RCP Air Coolers, RCP Flywheel Covers, NIS Source Range Detectors, Regenerative Hx Support	300,712	Carbon Steel	0.238
(13) Component (Carbon SteelThickness less than 0.08-inch) - Gratings, Ductings, Fans, HVAC Units, ICIS Boxes, Cable Trays, Duct Connecting Flanges, I/C Devices, ITV Instruments, NIS Air Horn	233,954	Carbon Steel	0.0496
(14) Component (Stainless Steel) - C/V Reactor Coolant Drain Tank, RCP Purge Water Head Tank, Fuel Transfer System, Refueling Machine, RMS Indicators, ICIS Instruments, DRPI Tube, Transmitters, Level Switch, Luminaries, Containment Rack, C/V Reactor Coolant Drain Pump, Containment Sump Pump, Piping Support in the RWSP	12,976	Stainless Steel	0.295
(15) Copper - Coils, Copper Tubes, Luminaries, Cooling Coil's Fins	250,972	Copper	0.0088
(16) Uninsulated Cold-Water-Filled Piping (Stainless Steel)	14,892	Stainless Steel Water	0.323 1.36
(17) Empty Piping (Stainless Steel)	982	Stainless Steel	0.126
(18) Uninsulated Cold-Water-Filled Piping (Carbon Steel)	663	Carbon Steel Water	0.197 0.630
(19) Empty Piping (Carbon Steel)	896	Carbon Steel	0.138
(20) Aluminum - NIS Power Range Detectors	59	Aluminum	0.118
(21) Web Plate	622	Carbon Steel	41.4

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 11

Annotate Slide 58 to show the plant elevation associated with each RWSP level depicted on this figure.

ANSWER:

The requested evaluation information has been added to Figure 11-1. It is noted that the volumetric relation is provided in Figure 3-11 of MUAP-08001 "US-APWR Sump Strainer Performance".

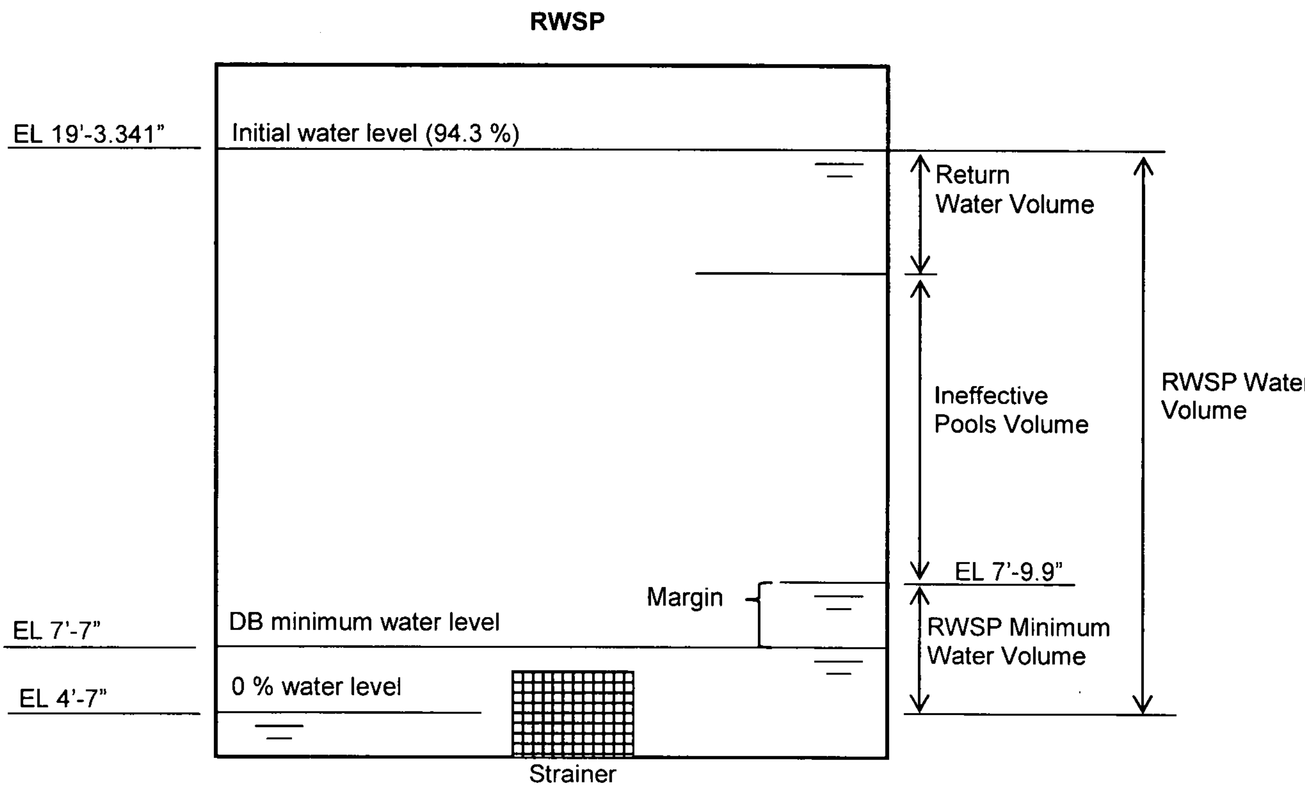

Figure 11-1 Plant Elevation Associated with Water Level for Minimum Water Level Calculation

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: 6

CHAPTER TITLE: ENGINEERED SAFETY FEATURES

DATE OF MEETING: 10/01/2013

QUESTION: Item 12

Supply an expansion of the plot given in Slide 61 over the first however long that time period is, 20 to 30 hours, to show in detail how the NPSH available versus NPSH required varies in a realistic analysis. Provide an expanded view of the plot on Slide 61 for the first 12 hours of the transient for both the realistic analysis and the limiting case.

ANSWER:

Figure 12-1 show the expanded view of the plot of NPSH available vs. time for the first time range. In addition, plots of best-estimate NPSH available and the containment pressure are added to this figure.

The best-estimate NPSH available is calculated without assuming that "the containment pressure equals to the saturation pressure of the RWSP water temperature". Therefore, this is the case that the pressure at the water surface is the containment pressure.

NPSH available (design-basis) decreases at approx. 30 minutes (1,800 seconds) due to the increase of the RWSP fluid temperature. When the RWSP water temperature exceeds approximately 212°F, the NPSH available for the design-basis is assumed constant because of the assumption that the containment pressure equals to the saturation pressure of the RWSP water temperature. After the RWSP temperature drops below approximately 212°F, NPSH available increases due to the RWSP temperature decrease. In this figure, there is a little margin between NPSH available (design-basis) and NPSH required during the RWSP temperature above 212°F, but there is more than 1 feet margin as shown in Figure 3-17 and Table 3-11 of MUAP-08001.

On the other hand, the increase in NPSH available (best-estimate) at 30 minutes corresponds to the increase in the containment pressure. Since the NPSH available (best-estimate) is much larger than the NPSH available (design-basis) during the period of time when the RWSP temperature is over 212 °F, it is concluded that MHI's current evaluation of NPSH available (design-basis) has sufficient margin and is sufficiently conservative.

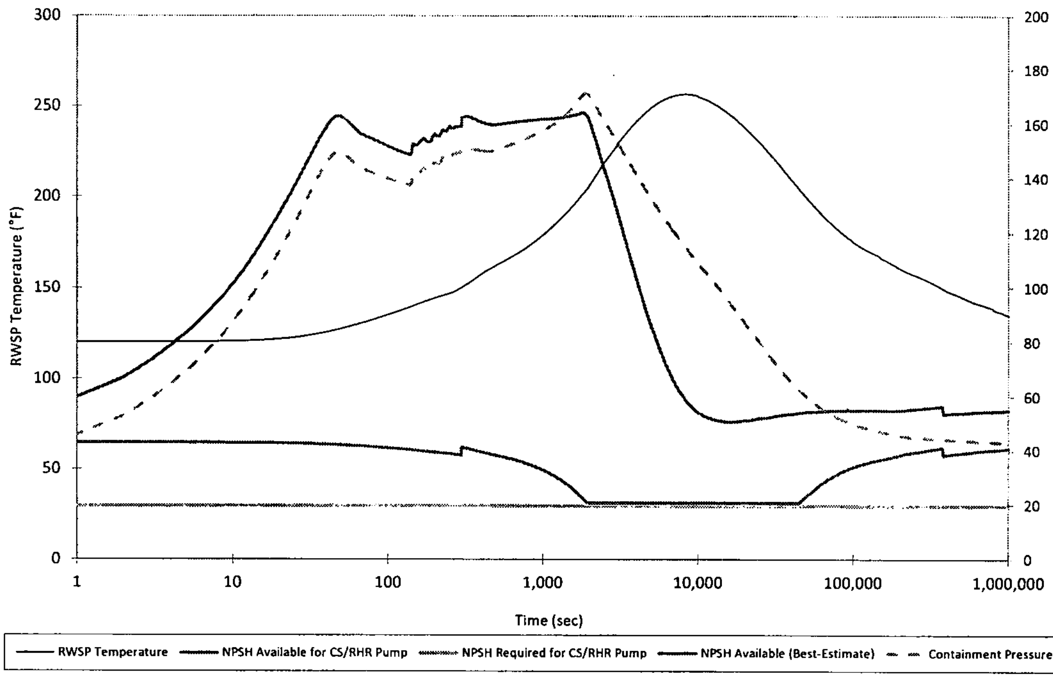

Figure 12-1 NPSH Available vs. Time

RESPONSE FOR ACRS SUBCOMMITTEE MEETING

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

CHAPTER: **6**

CHAPTER TITLE: **ENGINEERED SAFETY FEATURES**

DATE OF MEETING: **10/01/2013**

QUESTION: Item 13

The ACRS agreed with MHI that the core inlet blockage test results in MUAP-11002 and MUAP-12004 showed stable pressure conditions that were reasonably repeatable, but it was noted that the time to get to the stable conditions did not seem repeatable.

MHI agreed to double-check the time to get to the stable conditions in the CIB test results and provide justification for why the observed times are considered repeatable.

ANSWER:

The pressure drop criteria used to determine the introduction of chemical debris and termination of the tests were defined as [ ] pressure drop per an hour by MHI. The following table summarizes the pressure drop stabilization time to introduce chemical debris and to terminate measurements for both the base and repeatability tests. The detailed measured pressure drop test data corresponding to the table is provided in MUAP-11022 Figures 13-1 and 13-2.

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Figure 13-1 Sequence Data of HLB Tests

Figure 13-2 Sequence Data of CLB Tests