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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-10323

Subject: **MHI's Response to US-APWR DCD RAI No. 652-5194 Revision 0 (SRP Section 06.02.02)**

Reference: 1) "Request for Additional Information No. 652-5194 Revision 0, SRP Section: 06.02.02 – Containment Heat Removal Systems – Application Section: 6.2.2 and 6.3" dated November 1, 2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Responses to Request for Additional Information No. 652-5194 Revision 0".

Enclosed is the response to Question 06.02.02-62 that is contained within Reference 1.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

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

Enclosure:

1. Response to Request for Additional Information No. 652-5194 Revision 0

CC: J. A. Ciocco
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Enclosure 1

UAP-HF-10323
Docket No. 52-021

Responses to Request for Additional Information No. 652-5194
Revision 0

November 2010

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/30/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 652-5194 REVISION 0**

SRP SECTION: **06.02.02 – Containment Heat Removal Systems**

APPLICATION SECTION: **6.2.2 AND 6.3**

DATE OF RAI ISSUE: **11/1/2010**

QUESTION NO. : 06.02.02-62

Containment accident pressure is the pressure in containment during a postulated accident. RG 1.82 states that pump performance should be independent of the calculated increases in containment pressure caused by postulated LOCAs, and also states that sufficient available NPSH should be provided to system pumps assuming no increase in containment pressure from that present prior to the postulated LOCA. SRP 6.2.2 states that if "containment accident pressure is credited in determining available NPSH, an evaluation of the contribution to plant risk from inadequate containment pressure should be made." The US APWR design uses containment accident pressure in evaluating the net positive suction head (NPSH) for pumps that perform emergency core cooling and containment heat removal functions. Please perform a risk assessment and provide the results, along with a summary description of the methods used and assumptions made, to the staff for review. The risk assessment should address all plant accident conditions where CAP is credited for reliable operation of the ECCS and containment heat removal system pumps, and discuss the bases (e.g., results of thermal-hydraulic analyses) for determining whether CAP credit is needed. All accident initiating events (internal and external) and modes of operation modeled in the US APWR design-specific PRA must be addressed in assessing the risk associated with CAP credit. Qualitative arguments can be used to demonstrate that the risk associated with certain initiating events or accident sequences is insignificant or smaller than the risk associated with analyzed cases, as applicable. In particular, the risk analysis and its documentation should address the following items, as applicable:

1) Method, assumptions, and results for each LOCA initiating event category.
2) Method, assumptions, and results for non-LOCA accident initiating event categories which include feed-and-bleed operation, stuck-open safety valves, or any other means of providing heat to the in-containment refueling water storage tank.
3) Investigate any potential adverse interaction among the operator actions credited in the PRA for accident mitigation and the need to prevent human actions that could lead to inadvertent opening of the containment isolation valves or to containment depressurization.
4) Investigate the risk impact of operating emergency core cooling and containment heat removal systems with impaired containment integrity (e.g., undetected preexisting containment opening) or operation of containment heat removal systems at too high a rate.

In addition, describe the monitoring program that demonstrates that the actual performance of plant equipment is consistent with the performance assumed in the engineering and probabilistic analyses used to justify CAP in determining NPSH available.

Answer:

Introduction

Containment accident pressure is the pressure in containment during a postulated accident. In the evaluation of net positive suction head available ($NPSH_A$), MHI does not credit the containment accident pressure. Instead, MHI assumes that the containment pressure is equal to the refueling water storage pit (RWSP) fluid vapor pressure for high sump fluid temperatures (greater than 204°F). At these high sump temperatures, this assumption does credit an increase in containment pressure above that originally present prior to a postulated loss of coolant accident (LOCA).

This assumption is shown elsewhere to be a physically appropriate assumption (i.e., necessary to ensure that the RWSP water remains in the liquid state), and to be conservatively bounded by the actual containment accident pressure during all design basis conditions. This methodology is discussed in the response to RAI 4750 and upcoming revisions to MUAP-08001. This discussion demonstrates that events that could reduce containment accident pressure (e.g., impaired containment integrity or operation of heat removal systems at too high a rate) would also accordingly reduce the peak RWSP fluid temperature, such that the containment accident pressure still bounds the RWSP fluid vapor pressure.

Therefore, there is no increase to plant risk from this methodology, with respect to achieving and/or reaching the peak RWSP temperature. However, a rapid depressurization of containment during these periods of high RWSP temperature could potentially cause a loss of sufficient NPSH margin (e.g., due to flashing).

An evaluation of the contribution to plant risk from inadequate containment pressure during periods of high sump temperatures during post-LOCA operation has been performed. This evaluation was performed to address the five key principles of risk-informed decision making of RG 1.174 Rev. 1.

First, an exhaustive search of initiating events and accident scenarios that can result in high RWSP temperature was performed. Secondly, events that can cause loss of containment pressurization were identified. Finally, a quantitative risk assessment was performed to evaluate the risk from loss of sufficient NPSH margin caused by rapid containment depressurization.

The first section of this RAI response identifies the potential events which may lead to high RWSP temperature and the maximum time that the RWSP temperature is above 212°F, which is the saturated temperature of water under the containment pressure assuming the most severe post accident containment depressurization. The second section identifies potential events which may cause a loss of containment pressurization. The third section evaluates the contribution to plant risk from the most limiting event, a failure of containment isolation. The last section discusses monitoring programs to ensure the performance of plant equipment used in the risk assessment herein.

The risk assessment is performed for high temperature periods above 212°F, while the $NPSH_A$ evaluation uses 204°F to define periods of high temperature. This arises from the use of 212°F as the saturation temperature corresponding to nominal atmospheric pressure (14.7 psia), while 204°F is the saturation temperature corresponding to a conservative minimum US-APWR tech spec initial containment pressure (plus margin). This change is made for simplification and is judged to have a negligible impact within the order of accuracy of the risk assessment.

1. Identification of Events Leading to High RWSP Liquid Temperature

In the evaluation of NPSH, containment pressure is assumed to be equal to the sump (RWSP) fluid vapor pressure for high sump fluid temperatures. Therefore, containment pressure is important when the RWSP liquid temperature is above 212 °F.

The RWSP contains a minimum of 583,340 gallons of borated water with temperature equal to or below 120 °F (US-APWR Technical Specification) during operation at power. Due to the large heat capacity of the RWSP fluid, causes of significant increase in RWSP liquid temperature are limited to events involving high energy release from the Reactor Coolant System (RCS) or the main steam system inside the containment. Events that may result in RWSP liquid temperature above 212 °F are discussed below.

- **High energy release from the RCS** – LOCA events result in high energy release inside the containment that lead to an increase in RWSP liquid temperature. Table 1 shows the periods when RWSP liquid temperature exceeds 212 °F during a LOCA, which are categorized into three divisions by break size. These break sizes are consistent with the LOCA break size categorization applied to the PRA. In 2-inch diameter small break LOCAs (SBLOCAs), the maximum RWSP liquid temperature was maintained below 212 °F under all conditions. Therefore, SBLOCA events are excluded from events leading to high RWSP liquid temperature. For medium break LOCA (MBLOCA) and large break LOCA (LBLOCA) events, the RWSP liquid temperature can exceed 212 °F and there is a potential for loss of significant NPSH margin if the event is followed by rapid reduction in containment pressure. The duration for which the RWSP liquid temperature exceeds 212 °F ranges from 1,500 seconds to a maximum of 86400 seconds (24 hours) for LBLOCA. Because LBLOCA events provide bounding results with respect to MBLOCA events for the durations of high RWSP liquid temperature, these values are also conservatively applied to MBLOCA events.

  Initiation of feed and bleed operation also involves high energy release in the containment. The safety depressurization valves opened during feed and bleed operation are 4 inches in diameter, and therefore, RWSP liquid temperature increase after feed and bleed would be similar to MBLOCA events. Feed and bleed is considered in the events leading to an increase in RWSP liquid temperature above 212 °F.

  Stuck-open safety valves result in LOCAs that may or may not cause automatic or manual actuation of safety injection systems and spray systems. According to the NUREG reports (NUREG/CR-5750, NUREG/CR-6928), there have been two pressurizer safety valve stuck-open events experienced in the US nuclear industry between the period of 1988 and 2002. The leak rates for the two events were 200 gpm and 25 gpm during shutdown. Taking into consideration the relatively low leak rates caused by the stuck-open pressurizer safety valve events compared to those of MB-LOCA and LB-LOCA events, stuck-open safety valve events are categorized as SBLOCA events in the PRA. Simultaneous stuck-open safety valve events involving multiple safety valve failures that result in an energy release equivalent to MBLOCA or LBLOCA were screened out from evaluation. This is because there has been no industrial experience of multiple stuck-open events. Even if such events were to occur, it is likely that leak rates from the valves would not be significant, so that the consequence would not be as severe as MBLOCA or LBLOCA.

- **High energy release from the main steam system** – The bounding event of this type in terms of energy release in containment is a main steam line double-ended guillotine break (DEGB). Thermal hydraulic analysis performed for Section 6.2.1.4 of DCD Chapter 6 confirms that the RWSP liquid temperature will not exceed 212 °F if the intact steam lines can be isolated from the faulted line. Should failures in both the main steam isolation valves (MSIVs) and main steam check valves (MSCVs) occur following a main steam line break (MSLB), the main steam from multiple steam generators (SGs) will be released from the faulted steam line. In this case,

the consequence can be worse than the Section 6.2.1.4 analysis. However, since the main steam isolation valves (MSIVs) and the (MSCVs) are independent, the probability of both valves failing is low. Given that the frequency of steam line break in containment is 1.0×10^{-3} /reactor-year (RY), and failure probabilities of MSIVs and MSCVs respectively 1.2×10^{-3} and 1.0×10^{-4}, the frequency of a steam line break followed by failures of both the MSCV and the MSIV of the faulted SG is 1.2×10^{-10} /RY. The frequency of main steam release from multiple SGs would be lower than this value since the closure of MSIVs of the intact main steam lines can also prevent main steam release from intact SGs. Thus, the frequency of a MSLB event resulting in energy release from multiple SGs in the containment is estimated to be more than three orders of magnitudes lower than the core damage frequency of the plant, and the contribution to plant risk is negligible.

Within the potential sources for high RWSP liquid, MBLOCA event, LBLOCA event and initiation of feed and bleed are considered to be the limiting events that may result in the loss of sufficient NPSH, when followed by rapid containment depressurization.

Table 1. Duration of RWSP Temperature > 212 °F after LOCA

LOCA Type	Break Diameter	Duration of RWSP Liquid Temp. above 212 °F after LOCA
LBLOCA	Over 8 inch	1500 sec to 1 day (86400 sec)
MBLOCA	2 inch to 8 inch	1500 sec to 1 day (86400 sec)
SBLOCA	Under 2 inch	N/A

2. Sources for Reduction in Containment Pressure

Potential sources for a reduction in containment pressure are listed below, along with the bounding event sequence from the existing PRA and / or other justification for screening out the source from additional consideration. Sources which require further evaluation are detailed in Section 3.

- Containment Structural Failure – Even for the worst case large LOCA event, the peak containment pressure is 59.5 psig according to the design-basis accident analysis (See DCD Table 6.2.1-1). The US-APWR design applies a pre-stressed concrete containment vessel (PCCV) and its ultimate pressure is 201 psig (DCD Section 19.2.4). This provides sufficient margin against pressure increase after LOCA events. Therefore, a containment structural failure that will result in rapid depressurization is unlikely to occur, and the contribution to plant risk is negligible.
- Operator Error (Isolation) – There is no operational procedures that allow operators to open containment isolation valves (CIVs) that will result in containment bypass during an accident. The only possibility that the operator will allow containment bypass is by operator error. In a condition where a LOCA has occurred, the containment isolation (CI) signal is initiated and the alarm annunciated. A feature of the US-APWR design to prevent human errors that could lead to inadvertent opening of containment isolation valves is that the operators cannot manually control the CIVs without resetting the CI signal. It is unlikely that the operator will inadvertently open any CIVs in such a situation. Therefore, an operator error that opens an isolation valve and results in significant depressurization of the containment is unlikely to occur, and the contribution to plant risk is negligible.
- Operator Error (Heat Removal) – There is a possibility for the containment to experience depressurization through the actuation of additional containment spray trains or the heating, ventilation, and air conditioning (HVAC) system inside containment. Actuation of additional containment spray trains is analyzed explicitly in the design basis analysis of MUAP-08001, as discussed in the Introduction above. As shown in this design basis analysis, containment pressure is maintained above the RWSP saturation pressure after actuation of additional containment spray trains, which implies that this event will not lead to loss of sufficient $NPSH_A$. Operators inadvertently restarting HVAC systems in the containment is considered unlikely to

occur, because the HVAC system inside containment is isolated by the containment isolation signal. The operator would not be able to initiate containment cooling by the HVAC system unless containment isolation signal is reset. Therefore, the contribution to plant risk is negligible.

- Containment Isolation Valve (CIV) Failure – Failure of a CIV was considered to be the most likely event which may result in a rapid depressurization due to the number of valves and penetration lines. This event is discussed in detail in Section 3 below.

Within the potential sources for a reduction in containment pressure, failure of a containment isolation valve is considered to be the most likely event that may result in the loss of sufficient $NPSH_A$. The contribution of this event to plant risk is discussed in Section 3.

Pre-existing containment isolation failures are excluded because such failure will not cause rapid containment depressurization. Such failures will relax containment pressurization throughout the accident rather than causing rapid depressurization after reaching an elevated RWSP temperature and, therefore, will not result in significant loss in $NPSH_A$ margin.

3. Result of Probabilistic Assessment for the Loss of the Containment Isolation

Probabilistic analysis of significant depressurization due to loss of containment isolation during the period of high RWSP liquid was performed. The frequency of the RWSP liquid temperature being above 212 °F was estimated from frequencies of MB-LOCA events, LB-LOCA events, and accident sequences involving feed and bleed operation. The LOCA events, and feed and bleed operation resulting from fire and flood accident scenarios, have been considered as well as internal events. In the evaluation of fire and flood scenarios, fire and flood that potentially impact the availability of containment isolation valves have been assessed. Quantification of risk from inadequate containment pressure was performed by first estimating the frequency of accident sequences that leads to high RWSP temperature using the Probabilistic Risk Assessment (PRA) results for internal and external events, and then multiplying the probability of having failure in the penetration line or isolation valve that leads to containment bypass. Assumptions applied in the risk quantification are the following:

- The probability of an occurrence of rapid containment depressurization was evaluated assuming the only likely depressurization event from Section 2, the failure of a containment isolation valve.
- One hundred (100) lines were conservatively assumed for the number of penetrations, and 24 hours was conservatively assumed as the period of high RWSP liquid temperature. Although failures of isolation valves (depending on the size) do not all result in rapid containment depressurization, it was conservatively assumed that failures in any of the penetration lines during the 24 hour period will cause a loss of sufficient NPSH margin.
- For fire scenarios leading to feed and bleed operation, the PRA conservatively assumed that one train of the isolation valves are inoperable, since there is a high probability that one train of the isolation valve is affected by the fire event. Similarly, for flood events accompanying LOCA, one train of the isolation valve is conservatively assumed to be unavailable since there is a high chance the isolation valve is affected by the flood.

With these conservative assumptions, the total frequency of the sequences was determined to be two orders of magnitude lower than the core damage frequency described in Chapter 19 of the DCD. Dominant accident scenarios that lead to loss of sufficient NPSH are the following:

- Fire events followed by failures in the emergency feed water system (EFWS), and containment isolation.

  The plant is tripped after a fire event. Containment pressure and temperature increases due to the energy released from the RCS as a result of feed and bleed. One train of the containment isolation valves is inoperable due to the fire that has affected the cables. Mechanical failure occurs in the isolation valve of that intact train or any containment penetration line piping during the period where the RWSP is above 212 °F. The containment experiences a rapid depressurization, and safety

injection and containment spray pumps become inoperable due to loss in NPSH margin. Feed and bleed is no longer operable, and eventually the core is damaged.

- MBLOCA events followed by failure of containment isolation

  A MBLOCA event occurs and the plant is tripped. Random and fire induced failures occur in the EFWS, resulting in degradation of the heat removal function from SGs. Operators initiate feed and bleed. Containment pressure and temperature increases due to the energy released from the RCS via the breach, and the RWSP temperature exceeds 212 °F. Mechanical failure occurs in the isolation valves or any containment penetration line piping, and the containment experiences a rapid depressurization. Safety injection and containment spray pumps become inoperable due to loss in NPSH margin. Safety injection functions are no longer operable and the plant fails to mitigate the MBLOCA event.

The evaluated core damage risk from inadequate containment pressure is two orders of magnitude lower than the core damage frequency described in Chapter 19 of the DCD, even with the conservative assumptions as mentioned above. The contribution to plant risk from inadequate containment pressure is therefore considered negligible.

4. Monitoring Programs

DCD, Chapter 16, Technical Specification, Section 5.5.8 Inservice Testing Program commits the US-APWR to the use of the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and to its applicable Addenda for the testing of ASME Code Class 1, 2, and 3 Components.

The ECCS pumps are ASME Code Class pumps and will therefore be tested in accord with the ASME OM Code. ASME OM ISTB requires that Group A pumps be pre-service tested to confirm design basis capability. In accordance with the OM Code, once in-service they will be tested quarterly to detect degradation and bi-annually (Comprehensive Pump Test) to confirm the ability to meet design basis requirements. The use of the ASME OM Code is required as per 10CFR50.55a.

The containment isolation valves, including interlocks, are ASME Code Class valves and will therefore be tested in accord with the ASME OM Code Section ISTC. Containment isolation valves are considered Category A valves and will be tested and monitored in accord with program requirements. The US-APWR is committed to 10CFR50, Appendix J. and the COL Applicant is committed to a monitoring and testing program as per DCD Chapter 6, COL 6.2(8). The use of the ASME OM Code and Appendix J is required as per 10CFR50.55a. Additional details regarding the containment isolation valve monitoring and testing can be found in DCD Chapter 6 Engineered Safety Features, Sections 6.2.4 and 6.2.6; Chapter 14 Verification Programs; and Chapter 16 Technical Specifications, Section 5.5.16.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.