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TOKYO, JAPAN

February 6, 2009

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-09043

Subject: MHI's Responses to US-APWR DCD RAI No. 148-1700 Revision 1

Reference: 1) "Request for Additional Information No. 148-1700 Revision 1, SRP Section: 19-Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19.1," dated January 9, 2009

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document as listed in Enclosure.

Enclosed are the responses to the RAIs contained within Reference 1. Of these RAIs, the RAI (19-275) will not be answered within this package. MHI will need additional analyses or surveys for the response to this RAI. The response to this RAI will be submitted by 10th March.

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

Sincerely,

Y. Ogata

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

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NRO

Enclosure:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.148-1700 Revision 1 (proprietary version)

3. Responses to Request for Additional Information No.148-1700 Revision 1 (non-proprietary version)

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

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-09043

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from 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 Request for Additional Information No. 148-1700, Revision 1", dated February 6, 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain 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 and methodology developed by MHI for performing the design of the US-APWR reactor.

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 the development of the unique design parameters.

B. Loss of competitive advantage of the US-APWR created by the benefits of the Control Rod Drive Mechanism operation.

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 6th day of February 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09043
Docket No. 52-021

Response to Request for Additional Information
No. 148-1700 Revision 1

February 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 148-1700 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-271

Please address the following questions related to the alternate containment cooling discussed in Attachment 6A.14.1 of Revision 1 of the US-APWR PRA report.

(a) Alternate containment cooling is credited in the PRA when the emergency containment cooling by the CS/RHR system is lost. In Appendix 6A.14.1.1.1 it is stated: "The alternate containment cooling is performed with heat removal from train A, B of the component cooling water (CCW) system to the containment vessel recirculation units A, B, C and D..." This statement implies that cooling water is used only from CCW surge tank A (the one which is associated with pumps (trains) A and B). However, it is also stated in the same paragraph that "... the surge tanks A, B should be pressurized ...," which implies that cooling water is used from both CCW surge tanks. On the other hand, the simplified system diagram (Figure 6A.14.1-1) shows a nitrogen supply system only for CCW surge tank A. Please clarify.

(b) No hardware failures are modeled. The simplified system diagram (Figure 6A.14.1-1) and the human actions listed in Section 6A.14.1.1.3 indicate the existence of several sets of valves which can fail to open or close due to common cause.

(c) The human actions listed in Section 6A.14.1.1.3 indicate that the operators have to open the recirculation unit inlet valves CH-5, CH-6, CH-7 and CH-8 for alternate containment cooling. However, the simplified system diagram (Figure 6A.14.1-1) shows valves CH-5, CH-6 and CH-7 as normally open during operation. Please clarify.

(d) The impact of the CCW system re-alignment, needed to provide alternate containment cooling, on the availability and reliability of other mitigating systems credited in the same accident sequences needs to be investigated and any dependencies be identified and modeled in the PRA. Additionally, the potential for introducing new accident sequences should be investigated. Please discuss.

(e) The probability of the operator failure to re-align the CCW system for alternate containment cooling (event NCCOO02CCW) was estimated to be 2.6E-2. This probability is based on several assumptions, such as a rule-based behavior and a moderately high stress level. However, there is no discussion about the bases of these assumptions. For example, what is the assumed time window for operators to complete the required actions? Is the assumed time window realistic? Also, there are no requirements or guidance for the combined operating license (COL) applicant/holder to ensure the implementation of what is assumed in the design certification. For example, the assumption of rule-based behavior assumes that the COL applicant/holder will develop an emergency operating procedure (EOP) for alternate containment cooling. A COL action item should be included in the design control document (DCD) to ensure that such an EOP will be developed.

ANSWER:

(a)
The containment fan cooler units are provided CCW by trains C and D of CCWS. Therefore, only CCW surge tank B is pressurized. In order to be consistent with the design, the simplified diagram and the description in the DCD and the PRA technical report will be revised in DCD Rev.2. Although it has been confirmed that the impact of this gap is negligibly small for internal events PRA, internal events PRA will be revised in DCD Rev.2 to eliminate the discrepancy between PRA and design. In addition, for external events PRA, the PRA model has been updated to be consistent with the design during DCD Rev.1.

(b)
Although the operation of several valves is needed for alternate containment cooling, hardware failures were not modeled since it was considered that the failure caused by human error will be the dominant factor rather than hardware failures including single failures and common cause failures.
To adequately evaluate the importance of the hardwares, the PRA will be revised in DCD Rev.2 to include the hardware failures of alternate containment cooling system.

(c)
The three valves CH-5, CH-6 and CH-7 are assumed to be normally opened during operation. However, the remaining one valve must be opened. Moreover, these normally open valves may close when the containment fan cooler fans stop and there is a possibility that the valves are closed at the time alternate containment cooling is initiated.

(d)
Alternate containment cooling is required when CV heat removal via CS/RHR heat exchanger has failed. In other words, CS/RHR heat exchanger is not operating when alternate containment cooling is established. CS/RHR heat exchanger is the highest heat load for the CCWS. Therefore, by isolating the CS/RHR heat exchanger that is not used, heat load of alternate containment cooling will not cause degradation of CCWS function nor cooling of other components, even when the CCWS has been re-aligned.

(e)
Operator actions credited in the PRA are listed in the DCD Table 19.1-115. The description will be revised in the future revision if necessary. The COL action item 13.5(6) is to describe the program for developing and implementing emergency operating procedures, and the related activity is described in the FSAR Subsection 13.5.2.1.

Regarding the assumptions for the HRA on operators' action after initiation of accident (type C human action), it is described in Section 9.3.3.1 of the PRA technical report. The operations for the alternate containment cooling will be developed as part of the EOP and hence rule-based behavior is assumed in the HRA. The stress level for the type C human action is considered either "Moderately high" or "Extremely high", and in this case moderately high stress level is applied. It is because extremely high stress level is applied to the operations under conditions of failure in multiple primary safety systems.

The alternate containment cooling is a backup function in case of the loss of CSS and it is anticipated that other safety functions for core cooling are properly functional when operations for the alternate containment cooling are required.

Regarding the time window for operators to complete the required actions, it is considered that there is sufficient time margin, more than several hours, to complete operations for the alternate containment cooling, and the reliability for the completion of activities is expected high because of the following considerations:

- Operators are required to start the related actions after confirming that CSS is not functional and the containment pressure continues rising higher than the threshold pressure to issue the containment spray signal (34psig).
- Operators are required to complete the related actions before the containment pressure reaches the design pressure (68psig). According to the accident progression analyses described in Chapter 14 of PRA technical report, it takes more than several hours that pressure rises from the containment spray signal pressure to the containment design pressure.
- In addition, the containment ultimate pressure capability (201psig) is much higher than the design pressure, and there is still a lot of time margin until containment failure even if operators failed to complete the related activities before reaching the design pressure.

Impact on DCD
DCD will be revised to address the information discussed for this RAI.

Impact on COLA
This RAI and its response will impact the COLA, which refers the DCD.

Impact on PRA
The PRA technical report will be revised.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 148-1700 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-272

Please address the following questions related to the emergency and alternate ac gas turbine generators (GTGs) discussed in Attachment 6A.11 of Revision 1 of the USAPWR PRA report.

(a) Section 6A.11.1.4 "Test and Maintenance" does not provide any information about the assumed testing and maintenance strategy for GTGs in the PRA other than the fact that the maintenance of the class 1E GTG is performed on line. Please provide all relevant information and assumptions regarding the testing and maintenance strategy on which the assessed GTG failure probabilities are based and verify the applicability of operating reactor experience to the US-APWR design.

(b) It is stated in Chapter 7 of the PRA report that "U.S. generic data of diesel generators are conservatively applied to gas turbine generators." However, this statement is not supported by the experience with non-safety GTGs used at some U.S. nuclear power plants and no results of studies or analyses are provided to support the assumption in the PRA that the US-APWR GTGs will be at least as reliable as diesel generators. Two sensitivity studies were performed, which are reported in Chapter 18 of the PRA report as well as in Chapter 19 of the US-APWR design control document (DCD), to investigate the sensitivity of the PRA results to the recognized uncertainty associated with GTG data. One sensitivity study was performed to investigate the impact of potentially higher failure rates than those considered in the baseline case (as the industry experience with non-safety related GTGs indicates). The other sensitivity study is based on common cause failure (CCF) parameters of "general components," which are smaller than the CCF parameters of diesel generators used in the baseline case. However, no basis is provided to justify why the CCF parameters for the GTGs used in the US-APWR design cannot be higher than the CCF parameters for "general components."

(c) From an examination of the reported minimum cutsets and risk importance ranking, it appears that not all combinations of two GTGs were considered in the CCF analysis. For example, explain the reason why the CCF of the sets AB and BD are not shown in the results while AC, AD, BC, and CD are shown.

(d) The CCF probability of all GTGs to fail to run at some time during the first hour (event EPSCF4DLSRDG-ALL) was estimated to be 1.6E-4. This probability is about an order of magnitude lower than what the staff estimates (1.5E-3) based on the reported GTG failure rate (2.9E-2 per demand) and the assumed CCF parameters for a diesel generator set of four (beta = 0.13, gamma = .7, delta = 0.57). Similarly, the CCF probability of both alternate ac (AAC) GTGs to fail to run at some time during the first hour (event EPSCF2DLSRDGP-ALL) was estimated to be 2.3E-4, which is about an order of magnitude smaller than what the staff estimates (2.2E-3) based on the reported GTG failure rate (2.9E-2 per demand) and the assumed CCF parameters for a diesel generator set of two (beta = 0.077). Please justify these CCF probabilities.

ANSWER:

(a)
Although maintenance of gas turbine generators (GTGs) can be performed online, the PRA assumes that scheduled maintenance of the GTGs will be performed during shutdown.

The generic unavailability of diesel generators (DGs) reported in NUREG/CR-6928 is considered applicable to US-APWR GTGs based on the following considerations.

- The gas turbine generators are tested monthly. This test frequency is the same with the surveillance test requirement for DGs in the standard technical specifications (STS). The reliability of gas turbine generators of US-APWR are expected to be higher than diesel generators as reported in US-APWR technical report "Qualification and Test Plan of Class 1E Gas Turbine Generator System" (MUAP-07024). Therefore, frequencies of unplanned maintenance of the US-APWR GTGs are expected to be less than that of DGs.
- Gas turbine is a simple rotational engine with rotor, stator and driving shaft composed by less quantity of composite parts compared to diesel engine. This characteristic of GTGs results in reduction of maintenance load compared to that of DGs, and therefore, expected mean outage time per maintenance is less than that of DGs.

(b)
The generic data of non-safety GTG reported in NUREG/CR-6928 is based on experience of two components installed in the same plant. The reliability data is supported by 267 demands with 6 failures. Failure rates derived from such limited experience are not generic. It should also be noted that the GTGs studied in NUREG/CR-6928 are limited to non-safety components.
MHI has reviewed industrial based field data of GTGs and estimated their reliability based on operating experience from over 300 GTG units. The failure probability on demand was estimated to be 3E-4 per demand. Furthermore, failure modes and effect analysis has been performed and the unreliability was estimated to be approximately 1.5E-3 per demand. These reliability studies are reported in the US-APWR technical report MUAP-07024. Comparison of reliability data of DGs and GTG are also provided in MUAP-07024 and the study indicates that the reliability of GTGs is expected to be higher or equal to DGs.
The basis for the CCF probabilities applied to US-APWR GTGs is discussed in the response to question 19-15 of RAI#1.

(c)
All combinations of GTGs are considered in the CCF analysis, but some combinations do not appear in the cutsets listed in the DCD nor the PRA technical report since the cutsets have low contributions to CDF.

The reason particular combinations of CCFs have low contribution to the CDF is due to the nature of US-APWR design, which is not completely symmetric among all four safety trains, and the asymmetric plant configurations the PRA assumes for the simplicity of modeling. Example of US-APWR design and PRA assumptions that result in asymmetries in cutsets regarding the CCF sets of GTGs are the followings:

US-APWR design
- Power supply to charging pumps. The two charging pump used for RCP seal injection are supplied power from Class 1E train A and B bus, respectively.

PRA assumptions
- Unavailability due to test and maintenance (TM) of essential service water system (ESWS) and component cooling water system (CCWS). For ESWS and CCWS, TM unavailability are only considered for train B and C, which are the trains assumed to be in standby condition.
- Asymmetric initiating events. LOCA events are assumed to occur in loop A or direct vessel injection (DVI) line A. Accordingly, train A of the safety injection system (SIS) is not credited depending on the size of the postulated LOCA event.
- Alternate ac (AAC) power source. The PRA assumes that the two non-safety GTG will be connected to Class 1E bus A and D respectively.

The reason why some combinations of CCFs have relatively low importance can be explained by the US-APWR design and the PRA assumptions. For instance, the top four cutsets involving CCFs of two GTGs are the followings:

a. (Small LOCA) x (Consequential LOOP after plant trip) x (CCF of GTG B and C) x (ESWS train D unavailable due to TM)

b. (LOOP) x (CCF of GTG A and D) x (single failure of GTG C) x (failure to connect AAC to Class 1E bus) x (ESWS train B unavailable due to TM)

c. (LOOP) x (CCF of GTG C and D) x (single failure of GTG A) x (failure to connect AAC to Class 1E bus) x (ESWS train B unavailable due to TM)

d. (LOOP) x (CCF of GTG A and C) x (single failure of GTG D) x (failure to connect AAC to Class 1E bus) x (ESWS train B unavailable due to TM)

Within the four cutsets described above, cutsets b., c. and d. are symmetrical, that means the cutsets are composed by equivalent combination of failures. Since the PRA assumes that ESWS train D can also be unavailable due to maintenance, an accident scenario similar to cutsets b., c. and d. but having ESWS train D unavailable may also occur. For example, the following cutset can be a candidate of a core damage scenario:

> (LOOP) x (CCF of GTG A and B) x (single failure of GTG C) x (failure to connect AAC to Class 1E bus) x (SWS train D unavailable due to TM).

However, in this accident scenario, RCP seal injection by charging pump B can be achieved if the operator succeeds to cool the charging pump by alternate CCW, since Class 1E bus D is available. Therefore, RCP seal LOCA can be prevented and accordingly cutset does not lead to core damage. Similar cutsets involving CCFs of GTGs BC and AC also do not result in core damage for the same reason.

Accident scenarios equivalent with cutest a. but involve different combinations of CCFs are not modeled in the PRA. This is because asymmetrical configuration of the initiating event (SLOCA) and the mitigations systems are assumed in the PRA as listed above.

(d)
Failure rates of GTGs applied in the US-APWR PRA are listed in Table 7.1-2 of chapter 7 in the PRA technical report, and are as follows:

- Fail to start on demand is 5.0E-3 per demand,
- Fail to run during the first hour is 8.0E-4 /hr, and
- Fail to run after the first hour of operation is 3.0E-3 /hr.

The basis of these failure rates is discussed in item (b) of this response.
Applying these failure rates and the CCF parameters pointed out in this question, the CCF probabilities used in the PRA can be calculated.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 148-1700 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-273

The modeling of Reactor Coolant Pump (RCP) seal LOCAs in the US-APWR PRA are discussed in Appendix 6A.14.2 of Revision 1 of the PRA report. It is stated: "New type O-ring which is improved for resistance to heat and pressure will be used in the USAPWR. But the RCP seal LOCA model for the US-APWR conservatively uses a mode[l] for the old type O-ring." However, the old O-ring model was modified by assuming that the maximum leak rate of 480 gpm per RCP will occur if cooling of the seals is lost for more than one hour. Although the assumed leak rate is very conservative (even according to the old O-ring model), the assumption that the RCP seal adopted by the US-APWR can keep its integrity for at least one hour without water cooling is not consistent with either the new or the old O-ring model. According to these models, there is a small probability (about 5E-3) that a leak rate of 480 gpm will develop within 10 to 30 minutes following loss of seal cooling. Please provide a more detailed discussion and the reason for using this "modified" old O-ring model in the US-APWR PRA as well as the basis for the assumption that the adopted RCP seal can keep its integrity for at least one hour without water cooling.

ANSWER:

The basis for assuming that the RCP seal can keep its integrity for at least one hour without water cooling is provided in the attachment to this response. As shown in the attachment, the RCP seal can keep its integrity for at least one hour without cooling. It is expected that the leak rate will develop with time after RCP seal starts to degrade. However, since the leak rate after RCP seal has degraded is uncertain at this time, the PRA conservatively applies the maximum leak rate of the old O-ring model. For this reason the US-APWR O-ring model assumes no leak during the first hour after loss of cooling, and a leak rate of 480 gpm per RCP beyond the first hour.

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.

Attachment to answer to question 19-273

Endurance of RCP shaft seal at SBO

At the SBO, the RCP will be stopped, and supply of the seal injection water and cooling water will be stopped. The low-temperature injection water stored in the pump will be drained to the outside of the pump as the leakage water from the seal, and the high-temperature RCS water will be fed into the pump. As described above, since the high-temperature water is fed into the pump, the temperature of the seal will rise. We have, therefore, examined whether the seal temperature rises to the dangerous point where the shaft seal function can be deteriorated. This document shows the result of this examination.

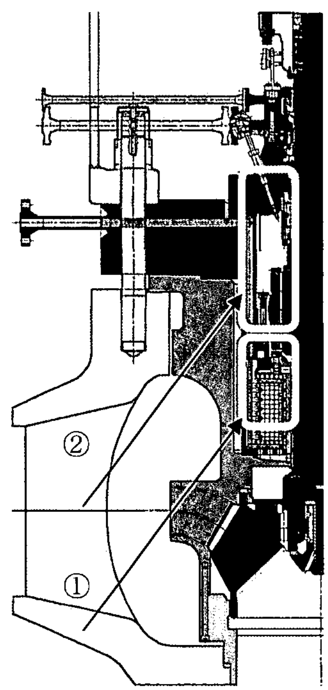
For the SBO conditions of the US-APWR, we evaluated the shaft seal temperature conditions.

1. EVALUATION CONDITIONS

- Seals No.1 through No.3 were manufactured by MHI.
- At the SBO, the seal injection water and the CCW cooling water should be stopped.
- When the SBO is detected, the seal injection water return line valve should be closed, and the total pressure should be applied to No.2 seal.
- When the SBO is detected, the pump should be immediately stopped. (According to the result data of the existing machines, the pump will be stopped in approximately 5 minutes.)
- We assumed that leakage from No.2 seal to the outside of the system (to the No.2 seal leak off line) would be 88.2lbm/hr(40L/hr) (reference value for total pressure application test during rotation in production).
- We assumed that at the time of SBO, the seal injection water temperature would be 130°F (54.4°C), and the RCS water temperature would be 580°F (304.4°C).
- For the seal, the upper limit temperature should be 235.4°F (113°C) (value specified in the instruction manual).
- Regarding the seal injection water capacity, we assumed 2 cases as described below to evaluate the temperature rise around the seal.

In the actual RCP, the high-temperature RCS water is fed into area ① shown in the right figure. The water is then mixed and fed into the upper area, and finally, the shaft seal inlet at the top of area ② will be heated.
Normally, after most of area ① is filled with the high-temperature water, the high-temperature water will be fed into area ②. For this evaluation, however, we set the condition on the safety side that the water should be completely mixed in areas ① and ② at the same time, and the seal temperature should rise comparatively quickly.

We set the seal injection water capacity (areas ①+ ② in the right figure) to 518lbm(235kg,235L), and assumed that the high-temperature RCS water fed into the seal injection water area would be immediately mixed with the seal injection water of the above capacity to uniform the temperature.



2. EVALUATION METHOD

Seal injection water temperature at SBO : 130°F (54.4°C)
RCS water temperature (temperature of water fed into seal injection water area) : 580°F (304.4°C)
Inner capacity of pump : 518lbm(235kg)
High-temperature water fed into pump (leakage to outside of pump) : 88.2lbm/hr,0.025lbm/sec (40L/hr, 0.011kg/sec)
We calculated the every-second difference, and completely mixed the water in the pump.
For evaluation, we assumed that the inner temperature of the pump would be T(t) when t sec passed, and we calculated the temperature of t+1 sec using the following formula:

$$T(t+1) = \frac{518 \bullet T(t) + 0.025 \bullet 580}{518 + 0.025} \quad \left(T(t+1) = \frac{235 \bullet T(t) + 0.011 \bullet 304.4}{235 + 0.011} \right)$$

We assumed that the temperature was 130°F (54.4°C) at first (T(0) = 130°F (54.4°C)), and we sequentially calculated the change in the temperature.

3. EVALUATION RESULT

Fig. 1 shows the temperature change at the seal inlet.
(For this evaluation, we ignored the heat transmitted from the outside (heat transfer from the outside). In the next section, therefore, we calculated the temperature rise caused by heat transfer from the outside.)

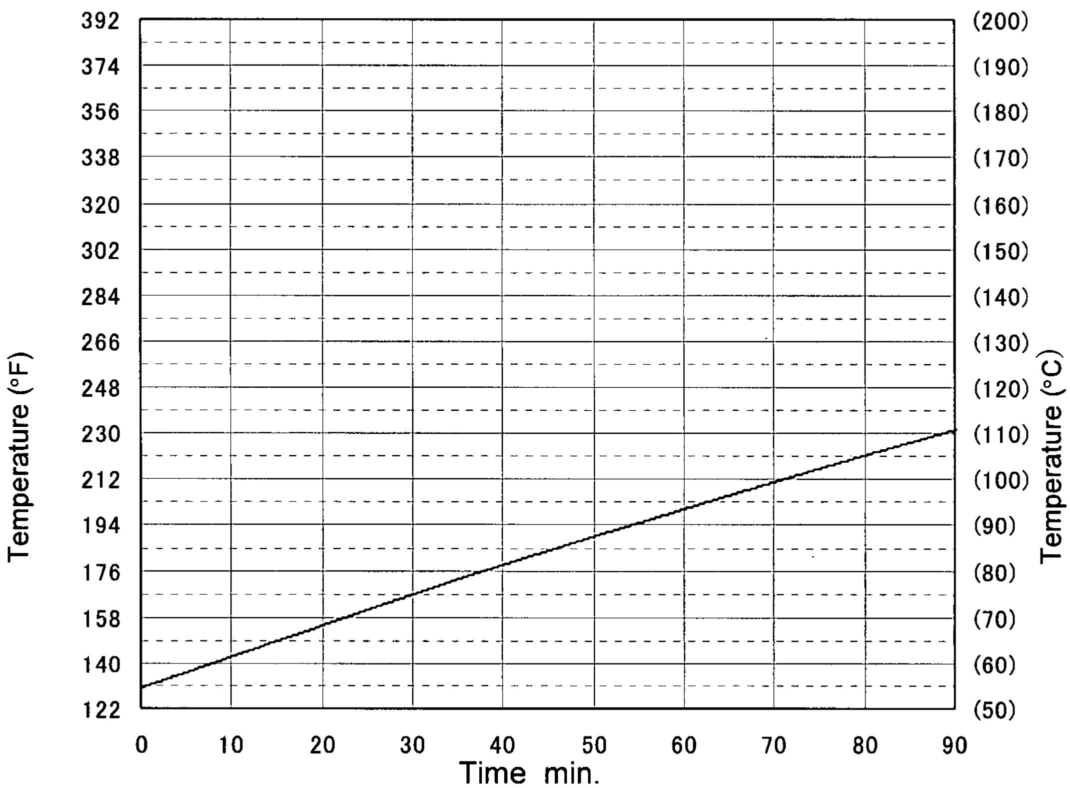


Fig. 1 Temperature change around shaft seal at SBO

4. TEMPERATURE RISE WHEN HEAT IS TRANSMITTED FROM OUTSIDE

As shown in the following figure, the external heat will be transmitted from the red line area. Assuming that this area is cylindrical and the length is the same, we calculated the transmitted heat (heat transfer) to determine the temperature rise.

(1)Evaluation conditions

On the safety side, the outer temperature was set to 580°F (304.4°C), and the inner temperature was fixed to 130°F (54.4°C). We ignored the heat transmission of water on the diffuser outer surface and the main flange inner surface (heat transmission from a fluid to a metal surface or from a metal surface to a fluid is infinite). In this way, we used the conditions of the safety side for evaluation.

(2)Evaluation method

Using the following formulas, we calculated the heat transfer from the heat resistance of the members, such as the diffuser. cvlindrical can. and thermal barrier.

$$q = \frac{\Delta T}{\sum_{k=1}^{n} \frac{\ln(r_{2k}/r_{2k-1})}{2\pi\kappa_1 L} + \sum_{k=1}^{n-1} \frac{\ln(r_{2k+1}/r_{2k})}{2\pi\kappa_2 L}} \qquad \delta T = \frac{q \times t}{Cp \times w}$$

Where,

ΔT=450°F (=580°F -130°F) (250°C (=304.4°C -54.4°C)) :Temperature difference between inside and outside

q Btu/hr (W) :Heat transfer to inside (heat transmitted into inside)

L=1.1ft (0.335 m) :Height of heat transmission area

κ1=9.82Btu/hr·ft·°F (17 W/m · K) :Heat conductivity of austenite-base stainless steel

κ2=0.40Btu/hr·ft·°F (0.7 W/m · K) :Heat conductivity of water

ΔT °F (°C) :Temperature rise in pump

Cp=1.22Btu/lbm·°F (5.1kJ/kg·K) :Specific heat of water

W=518lbm (235 kg) :Weight of incorporated water

t=3600(sec) :3,600 sec (We set the time to 3,600 sec to examine the temperature rise in 1 hour.)

(3)Evaluation result

We carried out calculation using the formulas shown above. As a result, the heat transfer was 14413 Btu/hr (4224W) (q = 14413 Btu/hr (4224W)), and therefore, the temperature rise caused by heat transmission from the water in the pump was 22.9°F (12.7°C) (ΔT = 22.9°F (12.7°C)).

5. CONCLUSION

As described in Sec. 4, the high-temperature water fed into the pump will raise the temperature. One hour after the SBO, therefore, the temperature of the seal inlet will rise to200.3°F (93.5°C). In addition, heat may be transmitted from the outside, and this heat transfer from the outside may be + 22.9°F (12.7°C). So the temperature may rise to 223.2°F (106.2°C) in total. This temperature is lower than 235.4°F (113°C) (235.4°F (113°°C) is the upper limit temperature of the heat non-resistant O-ring specified in the instruction manual). Therefore, the SBO will not deteriorate the seal function.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/6/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 148-1700 REVISION 1**

SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

DATE OF RAI ISSUE: **1/9/2009**

QUESTION NO. : 19-274

Please address the following questions related to the refueling water storage pit (RWSP) discussed in Attachment 6A.14.3 of Revision 1 of the US-APWR PRA report.

(a) It is stated that no potential common cause failures were identified for fault tree RWS. However, the containment sump strainers ST01A, B, C and D as well as the two motoroperated isolation valves (002 and 003) in the refueling water recirculation line can fail due to common cause. Also, the CCF (plug) of the containment sump strainers is highly risk significant. Please explain.

(b) It is stated that the list of single failures of components associated with fault tree RWS is shown in Table 6A.14.3-4. However, only a part of such failures is listed in Table 6A.14.3-4. For example the failure of motor-operated isolation valves 8820 and 9007 are listed with other systems in different tables. Please explain.

ANSWER:

(a)

Common cause failures of containment sump strainers are modeled in the PRA. The Multiple Greek Letter (MGL) parameters reported in NUREG/CR-5497 as "PWR Containment Sump Strainers Fail to allow flow" were applied. The MGL parameters applied to sump strainers are listed in table 8-5-3 of chapter 8 in the PRA technical report (MUAP-07030 R1). Attachment 6A.14.3 of the PRA technical report will be revised to state that CCFs of containment sump strainers are modeled.

The containment sump strainers are made of stainless steel and use perforated plates in a layered disc with 0.066 inch diameter hole, as discussed in Table 6.2.2-2 of DCD Chapter 6 and technical report "US-APWR Sump Strainer Performance, MUAP-08001-P, Rev. 2, December 2008". This design prevents buildup of debris at downstream locations in the refueling water recirculation line. The PRA therefore considers that the in-containment RWSP design has small impact on plugging probabilities in valves and operating reactor experience is applicable to the US-APWR. For the same reason, CCF due to plugging in valves cause by debris flowing from the RWSP is also considered to be unlikely to occur, and therefore is not modeled in the PRA.

(b)
In the PRA report, each basic events are basically listed the section of the system which they are allocated per the P&ID. Accordingly current tables of basic events in each section of Attachment 6A do not cover all basic events modeled in the fault tree described in the section. The list of basic events will be revised to list all basic events modeled in the fault tree (excluding support systems) of each section.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There PRA technical report will be revised to amend the list of basic events in Attachment 6A. However, there is no impact on the PRA results.

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QUESTION NO. : 19-276

The probability that one of the four pressurizer safety valves (PSVs) fails to recluse (stuck open) after opening for overpressure protection, thus leading to a loss-of-coolant accident (LOCA), is discussed in Appendix 6A.14.8 of Revision 1 of the US-APWR PRA. The failure rate of a stuck open PSV is taken to be 7E-5 per demand based on zero failures in a five year period from 1998 to 2003. This failure rate, which was taken from NUREG/CR-6928 "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," is significantly lower than most of the other failure rates reported in the literature and documented in Table 7.1-1 of the PRA report. For example, the failure rate recommended in the Advanced Light Water Reactor (ALWR) Utility Requirements Document for stuck open PSVs is 5E-3 per demand (which is almost two orders of magnitude higher that the failure rate used in the US-APWR PRA. NUREG/CR-6928 documents a study of the industry-average performance over a five year period including the years from 1998 to 2003. In that study, components from different systems with different operating conditions and maintenance policies were lumped together and actual failures that occurred before 1998 were not included in the database. For example, NUREG/CR-6928 reports in Table 8-1 two events of stuck open safety valves in U.S. pressurized water reactors (PWRs) of which the one that occurred at Fort Calhoun on July 3, 1992 involved a pressurizer code safety valve and resulted in high-pressure safety injection actuation, as discussed in NUREG/CR-5750. Furthermore, the frequency of a stuck open safety valve used in the US-APWR, which contributes to a small LOCA initiating event, is 3E-3/year and is taken from Table 8-1 of NUREG/CR-6928. This frequency is consistent with a stuck open PSV failure rate of 5E-3 and not 7E-5 as assumed in the US-APWR PRA. Please discuss.

ANSWER:

In the US-APWR PRA, it is assumed that the pressurizer safety valves will always open following a initiating event such as loss of offsite power (LOOP), partial loss of component cooling water (PLCCW) and total loss of CCW (LOCCW). Stuck open failure of the pressurizer safety valves following these initiating events that lead to LOCA is explicitly modeled applying the failure probability of safety valves reported in NUREG/CR-6928. It should be noted that the assumption that the pressurizer safety valvles will always open following such initiating events is conservative since it is highly likely that the RCS pressure will not reach the pressurizer safety valvle set pressure. Therefore, we judge that even if the stuck open failure probability of safety valves reported in NUREG/CR-6928 is low as pointed out by the staff, the US-APWR PRA does not provide artificially low frequency estimation of the accident scenario that may distort the results.

The basis of the consideration that RCS pressure will not reach pressurizer safety valve set pressure following LOOP, PLCCW and LOCCW events are shown below:

After LOOP event, the reactor will be instantly tripped due to loss of power supply to the reactor trip breakers. Since the reactor will be trip before SG heat removal function actually degrades, the RCS pressure is unlikely to increase above the pressurizer safety valve set pressure.
Time series variation of RCS pressure after LOOP event is shown in the PRA report Attachment 5A "Thermal/Hydraulics analysis for success criteria", section 5A.3.3. The results indicated that even under condition assuming no SG feed water, the RCS pressure will not reach the pressurizer safety valve set pressure during a certain period of time. From the comparison of the results with the analysis performed for loss of feed water initiating event, which is described in the same section 5A.3.3, the probability that the RCS pressure after LOOP reaches the pressurizer safety valve set pressure is considered to be very low.

In the event of loss of CCW, the operators will detect the symptom of the event, such as low pressure at pump outlet or high CCW temperature and manually trip the plant before losses of main feadwater would occur. In most cases, the reactor would be tripped before SG cooling ability degrades, and therefore, the RCS pressure is unlikely to reach the pressurizer safety valve set pressure.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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SRP SECTION: **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

APPLICATION SECTION: **19.1**

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QUESTION NO. : 19-277

The following statement is made in Attachment 6A.14.11 of Revision 1 of the US-APWR PRA report: "The [reactor coolant] pump is designed so that the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling." This design feature of the reactor coolant pumps (RCPs) is credited in the PRA and therefore must be listed in the appropriate Chapter 19 section (Table 19.1-115) of the US-APWR design control document (DCD).

ANSWER:

This design feature of the RCP cooling is described in chapter 5 of the DCD as follows:

5.4.1.3.4 Loss of Component Cooling Water
If loss of CCW should occur, seal injection flow continues to be provided to the RCP. The pump is designed so that the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling.

This design feature will be listed in Table 19.1-115 of DCD chapter 19.

Impact on DCD
Table 19.1-115 of DCD chapter 19 will be revised as follows:

Key assumptions
Design features j. The reactor coolant pump is designed so that the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.