MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 5, 2008

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-08275

Subject: **MHI's Responses to US-APWR DCD RAI No.92-1237 Revision 0**

Reference: 1) "Request for Additional Information No. 92-1237 Revision 0, SRP Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation," dated November 5, 2008

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

Enclosed are the responses to 32 RAIs contained within Reference 1. Of these RAIs, 2 questions #19-158 and #19-165 will not be answered within this package. These questions require additional time for internal discussions and computations, and will be answered by 19th of January 2009.

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 submittals. His contact information is below.

Sincerely,

Y. Ogata

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

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Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No. 92-1237 Revision 0 (proprietary version)

3. Responses to Request for Additional Information No. 92-1237 Revision 0 (non-proprietary version)

Attachment to Enclosure 2: CD 1 "MACCS2 input data". The files contained in CD 1 are listed in Attachment hereto.

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

Contact Information
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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-08275

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. 92-1237 Revision 0" dated 5 November 2008, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design 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 development of methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of modeling information.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 5th day of December 2008.

Y. Ogata

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

Enclosure 3

UAP-HF-08275
Docket Number 52-021

Responses to Request for Additional Information No. 92-1237
Revision 0

December, 2008
(Proprietary Information Excluded)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-151

On page 11-3 of MUAP-07030 (R0), it is stated that hydrogen control is to be achieved through twenty glow-plug type hydrogen igniters. Please identify the power source for these igniters.

ANSWER:

The design specification of the hydrogen igniter including its power source is described in Subsection 6.2.5.2 of the DCD as:

"6.2.5.2 System Design

...

- Power supply from two non-Class 1E buses capable of cross-connection and non-Class 1E alternate alternating current (ac) gas turbine generator backed

...

The hydrogen monitoring and control system is supplied by the non-Class 1E P1 and P2 power system, with alternate power capability. P1 and P2 buses are capable of cross-connection, providing power to both motor control centers (MCCs). Both P1 and P2 buses are backed by non-Class 1E alternate ac gas turbine generators. The power distribution to the monitor and igniters is designed to minimize the impact of the loss of any single power source. As noted above, the containment hydrogen concentration is indicated in the MCR. This system may also be actuated manually."

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-152

Please justify the statement on the last paragraph of Section 11.2.2.2 of the PRA that dismiss the potential for recriticality following firewater injection. Please provide the results of your calculations or other technical evidence that support these statements.

ANSWER:

It is fundamental nuclear physics that neutron moderator is necessary for criticality and water works as the neutron moderator for light water reactors. If boron as neutron absorber is contained in water, criticality can be controlled. As described in Section 11.2.2.2 of the PRA report, re-criticality may occur if molten debris drops into water with low boron concentration. Therefore, it is important to discuss how much boron concentration is expected in the reactor cavity water during the various severe accident conditions.

Chapter 14 of the PRA report describes the severe accident progression analysis, which includes discussions on the MAAP calculation results for various accident scenarios. One of the most anticipated situations that firewater injection is immediately required is the case that containment spray system (CSS) is not functional in any trains. Thus, the reactor cavity water mass during three typical initiation events (i.e. medium pipe break LOCA, small pipe break LOCA and transient) without CSS are discussed. Please see Figure 14-9, Figure 14-49 and Figure 14-97 of the PRA report for MBLOCA, LBLOCA and transient, respectively. In all these cases, more than 200~300 tons (= $4.4\times10^5 \sim 6.6\times10^5$ lbm) of water exists in the reactor cavity immediately after reactor vessel failure. This water is from primary system coolant, such as blow down water after LOCA and accumulator water, hence is borated water. (Note: boron concentration of primary system coolant varies depending on time of the cycle. Accumulator water is controlled with the boron concentration between 4000 and 4200 ppm.) The water mass gradually decreases due to evaporation; however, boron remains in the reactor cavity. Therefore, even if un-borated firewater is injected into the reactor cavity, boron concentration is maintained at the level consistent with that of the reactor cavity water immediately after reactor vessel failure. The volume of the reactor cavity is approximately 500 m^3; so that the borated water will be diluted by no more than one half of the original concentration if firewater is injected to fully fill the reactor cavity. The firewater injection is manually controlled by operators, thus it is possible to manage the water level necessary to cool down molten debris as well as not to occur re-criticality.

The residual gadolinium in molten fuel works as a preventive measure to preclude criticality as described in Section 11.2.2.2 of the PRA report. Gadolinia is generally used as a neutron absorber in the nuclear design because Gadolinium has large neutron absorption cross section. If Gadolinia is contained in the debris bed, it could prevent re-criticality. In the US-APWR core design, Gadolinia fuel rods are contained in most of the new fuel assemblies with high enrichment. Therefore, at the beginning of the cycle, Gadolinia is contained in the debris bed abundantly. Even though the average fuel enrichment of the core is the highest at the beginning of the cycle, the re-criticality could not occur because the amount of Gadolinia is also the highest and Gadolinia can work as strong neutron absorber. At the end of the cycle, Gadolinia, which is a burnable absorber, burns out and the average fuel enrichment of the core is the lowest. In addition, fission products contained in the fuel also work as neutron absorber.

And finally as it is mentioned in Section 11.2.2.2 of the PRA report, if the gap within the debris bed is smaller than the moderator's volume ratio to debris (Vm/Vd) required for criticality, re-criticality does not occur.

Based on the factors discussed above, i.e. borated water with a certain level of concentration, existence of Gadolinia in debris bed and the expected Vm/Vd, it is concluded that the likelihood of re-criticality is very low with the enrichment used in the US-APWR core design

In addition, it should be noted that the accident management actions to inject unborated firewater into containment is a backup measure if containment spray system with highly borated water does not function. The need to deploy firewater under these unlikely hypothetical accident scenarios is therefore extremely low. Because of the highly unlikely accident scenarios susceptible to re-criticality, the challenge to the containment integrity is negligible from risk perspective.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

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Mitsubishi Heavy Industries

Docket No.52-021

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

APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-153

The results of the Sandia tests (Table 15-17 of the PRA) geared to the U.S. designed PWR cavities may not be directly applicable to the assessment of high pressure melt ejection for U.S. APWR (i.e., it is not clear why the APWR cavity is considered to be similar to that of Zion). Please demonstrate either by presenting scaled test data or analyses that are supported by data prototypical of APWR reactor cavity configuration that support the discussions of Section 11.2.2.3, and the analysis approach of Section11.3.4.3, and Chapter 15 (15.6.3), specifically:

(a) Please justify the range of the RV breach size that is considered in Table 15-19. Why are larger breach areas excluded?

(b) Please discuss the applicability of TCE to APWR geometry and conditions? Please list all the TCE dispersal parameters that have been used in the analysis, and the basis for their selection.

In addition, on page 19.1-11 (under the heading Core debris trap), it is stated that "...the effect of this design feature is not explicitly addressed in the Level 2 PRA..." Please describe:

(c) The degree of trapping expected for the reactor cavity (and subsequently the degree of dispersal to adjacent compartments).

(d) The existence of any flow paths around the reactor vessel that could directly connect the reactor cavity to the upper containment compartments.

ANSWER:

(a) Considering the inherent uncertainty of the RV breach size, parametrical studies were performed. Table15-19 shows parametrical study cases. As shown in the table, RV breach size of 1.0ft was selected as the basic size, referring to the previous studies in NUREG/CR-5582 and NUREG/CR-6338. According to the analysis performed for the Zion type in NUREG/CR-6338, the RV breach size was reported as approximately 0.4 m. According to the eight experimental result measuring the breach size due to creep rupture reported in NUREG/CR-5582, RV breach size (full scale equivalent hole diameter) of five cases were measured within the range of 0.23-0.63m, two other cases were 1.66m and 3.58m, and the remaining one was reported as penetration weld failure. Considering these previously performed studies and experiments, 0.3 m (1.0ft) of RV breach size was determined as the basic size of sensitivity analysis, and sensitivity studies were performed.

Regarding the second question, the potential largest break is RV failure in its perimeter, and the diameter of RV is approximately 17ft for the US-APWR. This study includes the potential largest breach areas.

(b) Concerning the applicability of TCE model to the US-APWR, the assumptions for extrapolation mentioned in NUREG/CR-6075 is considered as the applicability criteria. Among the assumptions of the extrapolation mentioned in NUREG/CR-6075, it is considered that the items 1 through 3 pertain to the applicability of the TCE model and the items 4 through 6 pertain to the applicability of results of Zion DCH loads analysis. The extrapolation applied in the US-APWR analysis does not consider the Zion results but only the methodology of the TCE model. Therefore, if the items 1 through 3 are satisfied, the TCE model is applicable to evaluate the pressure rise for the US-APWR.

Assumption 1: There is an intermediate compartment between the reactor cavity and the upper dome that is large compared with the cavity but small compared with the main containment volume, and there is no direct pathway for debris transport from the cavity exit to the main containment volume.

Applicability to the assumption 1:
The US-APWR containment has partitioned space for SGs between the reactor cavity and the upper dome, and this area can be considered as an intermediate compartment. This SG area is large compared with the cavity but small compared with the main containment volume. There is no direct pathway for debris transport from the cavity exit to the main containment volume in the US-APWR containment since there is SG area between the reactor cavity and the upper dome. Therefore, this assumption is satisfied.

Assumption 2: The cavity is not expected to collect a significant amount of water such that energetic fuel-coolant interactions do not dominant the DCH phenomenon.

Applicability to the assumption 2:
The US-APWR is designed to fully fill the reactor cavity before RV fails, however, this evaluation conservatively considers the influence if postulated HPME happens when completely no water is available at the reactor cavity. Therefore, this assumption is satisfied.

Assumption 3: The containment is essentially adiabatic on the time scale of the DCH event (~15s).

Applicability to the assumption 3:
The containment of US-APWR is a large dry type, therefore, it can be considered as adiabatic. Therefore, this assumption is satisfied.

Regarding the list of parameters for TCE model's analysis, please refer to Table15-19 and Table15-20 in the PRA report. The basis for their selection of base case (case1) parameter is shown in Table 19-153-1.

Table 19-153-1 Basis for the TCE Model Input for Case 1 Calculation

(c) According to the Walker model, it is considered that large amount of melt is trapped by a flow path with large angle. For the US-APWR reactor cavity design, there are at least two flow paths with large angle between the reactor cavity and the adjacent comportment, and therefore it is considered that the degree of dispersal to adjacent compartments decreases considerably. In addition, the volume of the debris trap area provided for the US-APWR is more than twice of the amount of reactor core material and it is considered there is sufficient capacity to trap the blown out core material.

This design feature is incorporated in determining the split fraction of the CPET heading for DCH (Event DH) and relatively small value is assigned through engineering judgment.

(d) About the existence of any flow paths around the reactor vessel, the reactor vessel is surrounded with thick heat insulator and also there are various structures such as piping, etc. It is therefore considered that the narrow gap between the reactor vessel and the concrete structure is tightly packed with various materials so that direct path due to the gap is almost negligible.

Nevertheless in evaluating the pressure rise due to a postulated DCH, the pathway around RV is conservatively modeled to have wider space to allow core debris dispersion from the reactor cavity to upper compartments.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-154

In Section 15.3.3.1 of the PRA, it is not clear how the hydrogen is being partitioned between the in-vessel and ex-vessel phases (last paragraph before Section 15.3.3.1 talks about 1/3 released in one minute after RV failure as hydrogen generation by "breaking up" and 2/3 in 5 minutes after that as hydrogen due to MCCI). Please explain why this is considered as bounding and/or representative of severe accident conditions for APWR.

ANSWER:

10CFR50.44 requires considering the equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction. Therefore, it is assumed that the equivalent amount of hydrogen generated from 100% reaction of zirconium of the active fuel length cladding is released to the containment during in-vessel phase.

In the evaluation of the postulated severe accident for the US-APWR, in-vessel retention is not credited conservatively. Therefore, it is considered that reactor vessel (RV) failure occurs after core damage. As for the accident management treatment, the reactor cavity is fully filled with water before reactor vessel failure, the core debris will fall into the water after vessel melt through. It is therefore considered in the US-APWR evaluation that hydrogen is generated through the oxidization of the residual zirconium which is not oxidized during the in-vessel process, through two ex-vessel processes. One is the oxidization of molten debris due to fuel-coolant interaction while it is released from RV and drops in the reactor cavity, and the other is the zirconium reaction with the steam generated from concrete erosion.

Hence, in addition to the hydrogen released during the in-vessel phase, it is assumed that the equivalent amount of hydrogen generated from reaction of all residual zirconium (it means total zirconium in the core except active fuel length cladding) is released to the containment during the ex-vessel phase. Consequently, the total amount of hydrogen generation in this analysis is the equivalent amount of hydrogen generated from reaction of total zirconium in the core.

During the in-vessel phase, the rate of hydrogen release is manipulated based on MAAP analysis result so that its time-integral value of hydrogen release becomes equal to the amount from 100% zirconium of the active fuel length cladding reaction.

During the ex-vessel phase, first, it is assumed that hydrogen generation through the oxidization of the residual un-oxidized zirconium in the process that the fallen core debris is broken up in the water. In this debris broken up process, it is assumed that debris is broken up due to FCI and one-third of the residual zirconium reacts with water; accordingly equivalent amount of hydrogen is generated within one minute after RV failure. The duration of one minute is determined through engineering judge considering various postulated accident conditions.

After core debris accumulates on the reactor cavity floor, it is considered that hydrogen is generated through the reaction with the steam generated from molten core-concrete interaction (MCCI) although the concrete erosion is very slightly and the duration is limited. In this MCCI process, it is assumed that two-third of the residual zirconium reacts with steam and equivalent amount of hydrogen is generated within five minutes from one minute after RV failure. In the MAAP analysis results performed for the US-APWR severe accident progression analysis, it is observed in several cases that concrete erosion continues for a few minutes. It is therefore five minutes is determined as the representative time of concrete erosion continuation time.

It is assumed that the rate of hydrogen generation is constant during the ex-vessel phase.

Summary of above conditions are shown in Table 19-154-1.

Table 19-154-1 Bounding Condition of the Hydrogen Generation

Phase	Time Frame		Amount of hydrogen generation	Rate of hydrogen generation
	From	To		
In-vessel	Start of Analysis	RV Failure	100% reaction of zirconium of the active fuel length cladding	depend on MAAP analysis result
Ex-vessel (by break up)	RV Failure	RV Failure + 1min.	reaction of 1/3 residual zirconium	constant
Ex-vessel (by concrete erosion)	RV Failure + 1min.	RV Failure + 1min. + 5min.	reaction of 2/3 residual zirconium	constant

Note:
Residual zirconium means total zirconium in the core except active fuel length cladding.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-155

Please provide the technical basis for the concentration criteria of various hydrogen combustion modes (e.g., deflagration, DDT, etc.) in Section 15.3.3.4.2 of the PRA.

ANSWER:

In Subsection 15.3.3.4.2, the criteria related to hydrogen combustion are defined as below table.

	H_2 Molar Fraction	Steam Molar Fraction	O_2 Molar Fraction
Non-flammable (N)	LT 0.05	or GE 0.55	or LT 0.05
Flammable (B)	GT 0.05	and LT 0.55	and GT 0.05
Flammable and Hydrogen Molar Fraction above 10% (GB)	GT 0.1	and LT 0.55	and GT 0.05
Potential Detonation (DET)	GT 0.2	and LT 0.35	and GT 0.08

(Note: GT: greater than, LT: less than and GE: greater than or equal)

Technical bases for the concentration criteria are as follows.

(1) Non-flammable (N)

The criteria are determined as an antinomy to the criteria for (2) below.

(2) Flammable (B)

As for the hydrogen concentration limit for hydrogen burn, the value of approximately 0.04 to 0.05 is shown in Reference 1(p. 1-4, Figure 1-2) and Reference 2 (p. 43, Fig.3). It is considered that this value is influenced by test equipment and ignition energy. In Reference 3 (p. 66, Figure 25), it can be observed that the significant pressure rise due to combustion begins from about 0.05 of the hydrogen concentration. Therefore, the criterion of hydrogen burn is determined as the concentration of 0.05.

The steam concentration limit is distributed between about 0.5 and 0.6 in reference 1 (p. 1-4, Figure 1-2) and reference 2 (p. 43, Fig.3). The flammability limit by steam is set to 0.55 in reference 4 (A-20, Table A-2). Therefore, the criterion of steam concentration is determined as 0.55.

Since the air concentration limit is delineated from 0.2 to 0.25 in reference 1 (p. 1-4, Figure 1-2), the oxygen concentration limit is calculated from 0.04 to 0.05, assuming approximately 20% of air is oxygen. The flammability limit by oxygen is set to 0.05 in reference 4 (A-20, Table A-2). Therefore, the criterion of oxygen concentration for hydrogen burnable limit is set up with 0.05.

(3) Flammable and Hydrogen Molar Fraction above 10% (GB)

10CFR50.44 requires that hydrogen concentration is limited less than 10%. Considering this requirement, the criterion of hydrogen concentration is set with 0.10. The criterion of steam and oxygen is determined same as (2).

(4) Potential Detonation (DET)

As for the hydrogen concentration limit, the value from about 0.18 to 0.20 is delineated in Reference 1 (p. 1-4, Figure 1-2); hence the criterion of hydrogen concentration is set with 0.2. The steam concentration limit is about 0.35 in the same reference and the criterion of steam concentration is determined as 0.35. Regarding the oxygen concentration limit, approximately 0.42 of air concentration detonable limit can be read in the same reference, and it is simply converted into oxygen concentration of 0.08 assuming approximately 20% of air is oxygen. Hence, the criterion of oxygen concentration is set with 0.08.

In addition, it is considered through the recent studies that the detonation limit has a strong tendency to depend on the geometry of a compartment in which combustion generates, and turbulent flow due to combustion is developed, etc. rather than the simple gas composition. However the geometrical consideration is difficult to incorporate into the criteria determined in this study. The criteria of detonation due to gas composition are therefore approximately estimated through engineering consideration based on the references listed below.

References

1 S.R. Tieszen, et al., Detonability of H2-Air-Diluent Mixtures, NUREG/CR-4905, June 1987.

2 Douglas W. Stamps and Marshall Berman, High-Temperature Combustion in Reactor Safety Applications, Nuclear Science and Engineering, Vol. 109, pp.39-48, 1991.

3 William B. Benedick, John C. Cummings, Peter G. Prassinos, Combustion of Hydrogen: Air Mixture in The VGES Cylindrical Tank, NUREG/CR -3273, May 1984.

4 S. E. Dingman, A.L. Camp, et al., HECTOR Version 1.5 Use's Manual, NUREG/CR -4705, April 1986.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.92-1237 REVISION 0**

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

APPLICATION SECTION: **PRA**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO. : 19-156

Please provide justification for the scenarios considered for the evaluation of hydrogen generation and distribution in Section 15.3.3 of the PRA are bounding for APWR severe accidents. Please explain the impact of uncertainties in hydrogen release and transport on the overall conclusions.

ANSWER:

In the hydrogen mixing and combustion evaluations, as described in Section 15.3.3.1(2) of the PRA report, the hydrogen release rate from RCS to the containment before reactor vessel (RV) failure, calculated in MAAP analyses, is modified to increase the magnitude without changing the release duration so that its integrated amount of hydrogen is equivalent to the hydrogen generated from 100% of active fuel cladding (which is equivalent to about 75% of the total Zr in the RV). After RV failure, it is assumed that the residual unoxidized Zr (which is equivalent to about 25% of the total Zr in the RV) is oxidized in the reactor cavity through FCI or MCCI. The assumption in this evaluation is appropriately conservative because in-vessel Zr-water reaction ratio of 75% corresponds to about 80 percentile for low and high RCS pressure sequences and to about 100 percentile for medium RCS pressure sequences according to the expert judgment on the in-vessel hydrogen generation in NUREG/CR-4551. Thus hydrogen generated from all the Zr in the RV is used for hydrogen mixing and combustion evaluations considering the ex-vessel Zr-water reaction which is assumed to be within a short time such as several minutes.

As described above, uncertainties in hydrogen generation have very little impact on the conclusions because the amount of hydrogen used for the evaluation is the maximized value in Zr-water oxidation reaction although the uncertainty in the hydrogen release rate and duration remains.

Two sequences, one for the LOCA sequence and the other for the transient sequence with the depressurization valve open, are selected for the evaluation in order to consider the different release location of hydrogen generated before reactor vessel failure. In these analyses, containment spray operation is assumed, which is conservative assumption from the viewpoint of hydrogen combustion. Therefore, uncertainties in the accident sequences have no impact on the conclusion. Detailed discussions are described below.

In the LOCA sequences, hydrogen generated in the core is released through the broken portion to the SG compartment. Hydrogen also generates in the reactor cavity water after RV failure. Uncertainties in broken size have no impact on the conclusion because the hydrogen release path is common to almost all LOCA sequences with various broken size.

In the transient sequences, hydrogen generated in the core is released through the pressurizer safety valves and pressurizer relief tank to RWSP, and then is released through the depressurization valve to the containment dome compartment by the RCS depressurization operation. If RCS depressurization operation is failed, then following three subsequent accident situations are anticipated.

1) Hot leg creep failure
   The hydrogen release behavior is the same as in LOCA sequences.

2) Temperature induced SGTR
   Hydrogen does not accumulate in the containment because it is released through SG to the secondary system.

3) RV failure under high RCS pressure
   Hydrogen generated in the core is released through the pressurizer relief tank to RWSP before RV failure. It is expected that the hydrogen mixing occurs through the RWSP vents.

Thus uncertainties both in hydrogen generation and in accident sequences have no significant impact on the conclusion about the adequacy to the regulatory requirements for the hydrogen control.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-157

Page 19.2-8 of the FSAR lists a maximum pressure in the containment vessel under adiabatic isochoric complete combustion of 137 psia, whereas the results on page 15-15 of the PRA chapter (MUAP-07030 [R0]), lists a pressure ranging from 127 to 152 psia, depending on the extent of Zr oxidation. Please provide the source for the pressure of 137 psia that is listed in the FSAR. If this value is based on a revised analysis, please provide the details of the calculations and the results.

ANSWER:

The maximum pressure of 136 psia listed in the DCD page 19.2-8 is a typographical error. The pressures listed in the PRA report (MUAP-07030) page 15-15 are the correct calculation result. The analysis results described in the DCD and the PRA report are identical and no revised analysis has been performed for the DCD.

The erroneous description in the DCD, 136 psia, will be amended to 127 psia in the next revision.

Impact on DCD

The erroneous description will be amended in the next revision.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO. 19-1237 REVISION 0

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

APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-158

The hydrogen combustion evaluation in PRA Section 15.3.3.5 notes that deflagration to detonation transition is not expected when the hydrogen concentration does not exceed 10%.

(a) Please provide specific citations of experimental or analytical results in support of this statement.

(b) If some of the glow plug igniters were to become inoperable, please explain whether situations would arise where pressure waves may arise locally, propagate, and/or cause structural failures. Also, please explain how many igniters would be needed to fail for this to occur. Please provide the results of any sensitivity studies that have been performed to assess such situations.

ANSWER:

This question will be answered by 19 January 2009.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-159

The PRA Chapter 17, quantification of the probability of induced steam generator tube ruptures refers to the 1993 report NUREG/CR-4551. A remark is also made referring to unspecified "recent studies". Please identify these studies, and present an updated analysis of this phenomenon for the US ABWR using current technology and analyses (for example, NUREG-1570, and EPRI Technical Reports 100693 and TR-107623) in support of the numerical quantifications presented. This analysis should also include the impacts of depressurizing of the secondary side of the steam generators on SG tube integrity.

In addition, please provide and justify the maximum number of broken tubes that were assumed in the case of a temperature-induced SGTR, considering the fact that it is not clear that a single broken tube would necessarily depressurize the primary side of the system to an extent that would mitigate further tube failures.

ANSWER:

MHI has reviewed several studies on the temperature induced steam generator tube rupture (TISGTR) for designing the US-APWR, including NUREG-1570 as listed in Table 19.2-6 of the DCD, and referenced in Chapter 11 of the PRA report (MUAP-07030 Rev. 1), document numbers 11-54 to 11-57. As pointed by the NRC staff, the "recent studies" referred in Chapter 17 of the PRA report include all these studies although these studies are not listed as the references in Chapter 17 to avoid repeating many times in one document.

MHI agrees to the NRC staff pointed as "This analysis should also include the impacts of depressurizing of the secondary side of the steam generators on SG tube integrity." The previously performed study in NUREG/CR-4551, the split fractions for TISGTR, TI hot leg rupture and no rupture were set as 0.018, 0.72 and 0.262, respectively, under the condition with RCS pressure of 2500psi. However in that report it is not clearly discussed about the secondary side depressurization. Therefore the split fractions for these ones under high RCS pressure sequences were set as 1/3 each in the US-APWR PRA containment phenomenological event tree, regardless of the secondary side condition.

Regarding the second question on the number of broken SG tubes, for the US-APWR Level 2 PRA, evaluation of the LRF, the number of broken loop does not influence to the evaluation results. It is assumed that single SG tube failure causes large release of fission product to the environment.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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RAI NO.: NO. 19-1237 REVISION 0

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

APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-160

In Section 15.4.2.1, it is stated, "This condition is a conservative estimation in terms of the molten core spreading behavior." Please explain why the sub-cooled cavity water and high heat removal capability are considered to be conservative when evaluating melt spreading.

ANSWER:

For the debris spreading analysis, the most preferable evaluation result is that molten debris spreads thinly and widely so that the accumulation depth becomes below 25cm, mentioned in the Generic Letter No. 88-20 as the maximum coolable depth. If subcooled water with high heat capacity is available in the reactor cavity, the molten debris may be cooled down and temporarily solidified. The solidified debris does not widely spread but deeply accumulates under the broken reactor vessel, and forms thick debris bed. The temperature of this accumulated debris bed gradually rise due to its own decay heat, especially from inside of chunk, and will start eroding floor concrete. However it is difficult to cool it down again since the accumulated debris is too thick and coolant water does not reach deeply into the accumulated debris.

MHI considered that relatively low temperature subcooled reactor cavity water should be good for temporary molten debris cooling for solidification, and this may promote accumulation of debris within narrow location and prevent widely spread. This will subsequently cause difficulty for ultimate debris cooling, and enhance the challenge to the containment integrity. It is therefore considered that the subcooled cavity water with high heat removal capability is conservative condition for debris spreading behavior.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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RAI NO.: NO. 19-1237 REVISION 0

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

APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-161

The FLOW-3D code is used to assess core melt spreading in the cavity. Please present a brief discussion of the special capabilities of this code. Describe the method of qualifying the code for this application in terms of validation and verification. Describe special modeling, if any, added to the code for the US-APWR analysis. Please provide the experimental validation basis of this code for application to debris pool spreading. In addition:

(a) Provide references supporting the statements (Section 15.4.2.3 of the PRA) that it is "widely recognized" for its ability to evaluate the solidification behavior of molten materials, that the solidification model is considered having a good ability in its prediction, and that this model is expected to predict precisely the solidification form and spreading size of molten metal in the casting processes.

(b) Please describe what is meant by the volume of fluid (VOF) method in FLOW-3D code.

(c) In Section 15.4.2.3 please provide the references for "Previously performed studies show that the heat transfer coefficient of molten core and a reactor cavity....decreases to a value ranging from 1.76×10^2 to 4.40×10^2...as a result of formation of a crust layer."

ANSWER:

1. The special capabilities of the FLOW-3D code is described as follows,

Dr. Cyril W. Hirt, who was the main developer of the FLOW-3D code, performed research and development of the basic algorithm of the computational fluid dynamics energetically in the general manager times of numerical value fluid laboratory of the Los Alamos research institute. He sent an analytical technique (VOF: Volume Of Fluid method) in order to analyze the free surface behavior. He and his co-workers established the Flow Science Inc. in 1980 and announced the FLOW-3D code as the commercial code. New analysis function and physical model have been developed to be able to analyze various flow phenomena precisely, and it was verified by experiment collation, and calculation precision was improved. The FLOW-3D code calculates correctly the complicated flow field including the free surface that other fluid software can hardly predict preciously.

As for the FLOW-3D code, in the many fluid analysis fields in where the transformation behavior of the free surface is important, it becomes one of the most famous fluid analysis software in the world. It is utilized for the evaluations of the flow fields that comprise the transformation behavior of the free surface such as the molten solidification analysis, the sloshing analysis of the liquid fuel, ink-jet analysis, casting hot liquids flow solidification analysis, film coating analysis, etc.

2. To qualify the FLOW-3D code for the core melt spreading assessment is described as follows.

A function to show below is demanded to perform the core melt spreading assessment.

- To evaluate an interface movement of the core melt with high precision.
- To evaluate a molten metal solidification process with high precision.
- To predict temperature profile of the core melt with high precision.

Because these requirements mentioned above are as same as ones for a casting process assessment, it is possible; to refer the verification and validation results for the casting process field. The verification results of the FLOW-3D code for the casting process are shown in the following answer to the additional item (a) in detail.

According to the temperature profile evaluation, the temperature distribution predicted with the FLOW-3D code is compared with that of the MINI-ACOPO test results. (T. G. Theofanous, C. Liu, J. Scott, D. Williams, T. Salmassi, Natural Convection in Hemispherical Enclosures at Internal Rayleigh Number up to 7×10^{14}, DOE/ID-10460 APPENDIX D)

In this test, experimental model is a hemispherical enclosure with a radius of 0.22m. Initial wall temperature is 3°C. Freon 113 which initial temperature is 37°C is put into the cooled model. The temperature history is measured at the 9 points which is located in height direction and on a center line of the test model. The temperature profiles at each measurement times evaluated with the FLOW-3D code are agreement with that of test results.

3. Special modeling for the US-APWR analysis is described as follows.

A function to set a heat convection coefficient at the interface between the core melt and water is added to a FLOW-3D code in particular. This function is inspected in the following procedures.

The simple verification analysis is carried out. In this case, molten metal and water are stored in a container. There is the core melt layer in a bottom part of a container. On the other hand, water is distributed on the core melt layer. It is confirmed that the FLOW-3D code evaluates the heat flux depends on the heat transfer coefficient between the core melt and water and this heat flux is agreement with the value evaluated by a hand calculation result.

Answer to the additional item (a)

References describe the FLOW-3D abilities for the casting processing behavior, are shown as follows.

The Flow Science Inc., a developer of the FLOW-3D code, carried out the analysis verification of the casting process energetically. It introduced a lot of verification examples of the casting processing assessment in a CFD library of its web site. Representative examples are shown below.

(1) Al-alloy casting
The verification results are shown in Figure 1. In this simulation, melting Al-alloy is injected horizontally into the cast box. As for the melting Al-alloy temperature decreases with progress of time and Al-alloy is solidified.

In this figure, it is observed that the Al-alloy spreading area and the shape of the Al-alloy by the FLOW-3D code accord with experiment results.

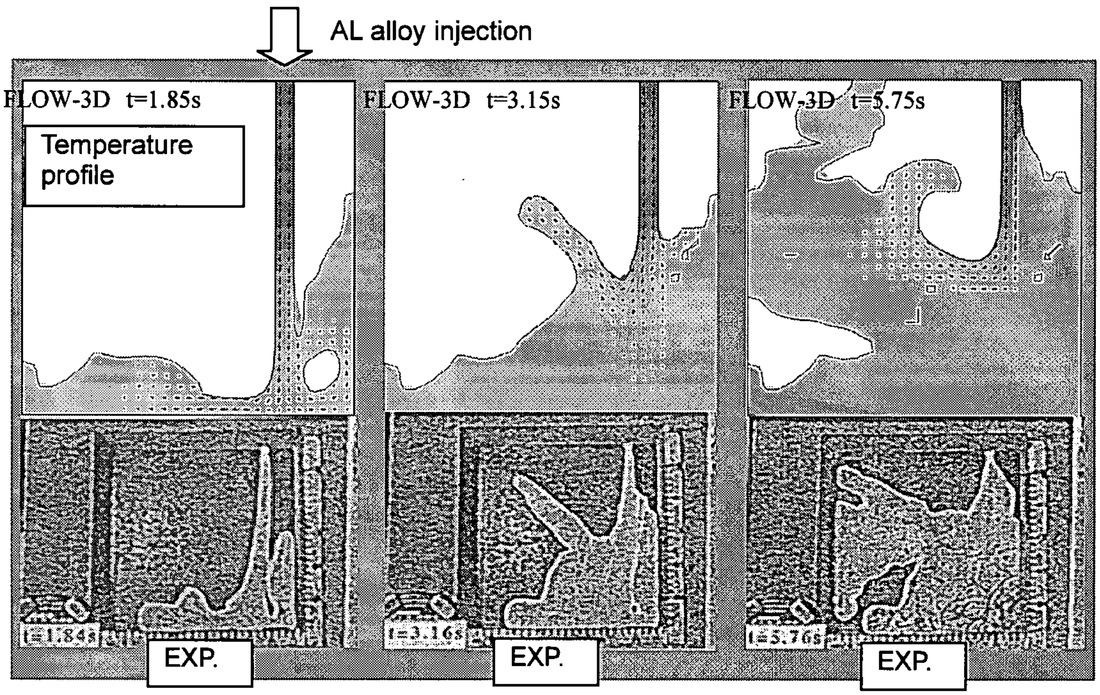


Figure 1 AL alloy casting processing simulation and experimental results from a CFD library in the CFD lab. Library.

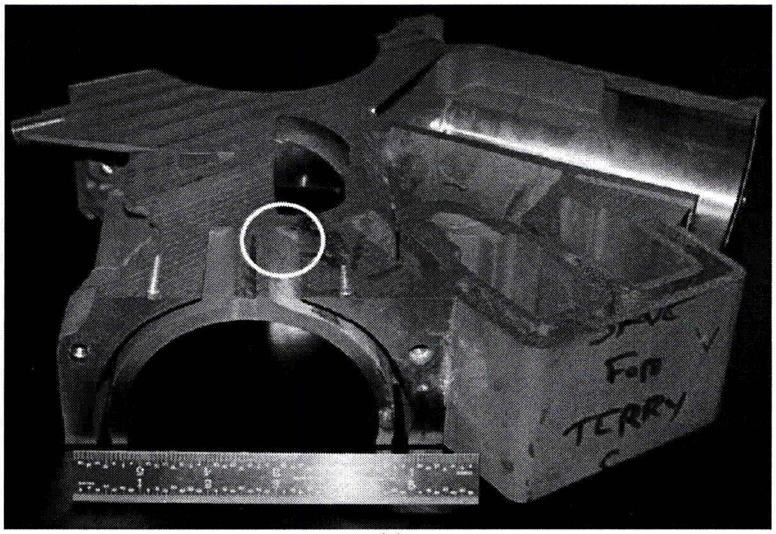
- H. Nieswaag and H. J. J. Deen, 57th World Foundry Congress

In addition, casting processing-related analysis verification is carried out by the FLOW-3D users who are researchers of universities and casting manufacturer, etc. The examples are shown below.

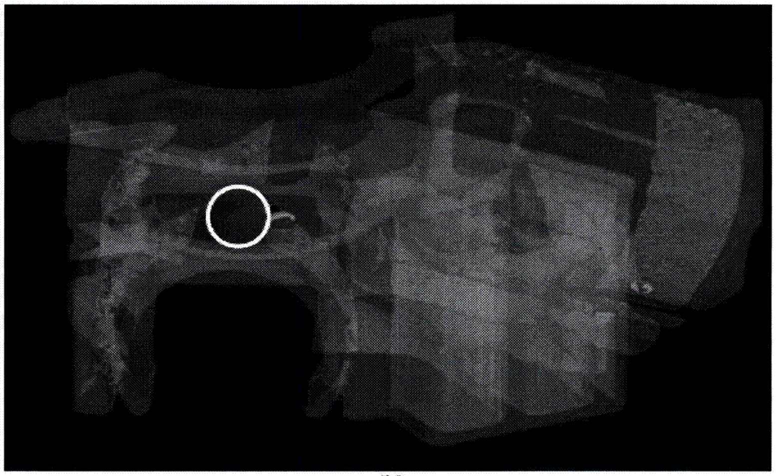
(1) P. Scarber, Jr., H. Littleton, Simulating Macro-Porosity in Aluminum Lost Foam Castings, American Foundry Society, © 2008, AFS Lost Foam Conference, Asheville, North Carolina, October, 2008

The result with the FLOW-3D code is shown in Figure 2.

In this paper, the authors confirmed that the FLOW-3D code can predict the temperature history of an engine head part and porosity location in an AL-alloy casting process very preciously.



(a)

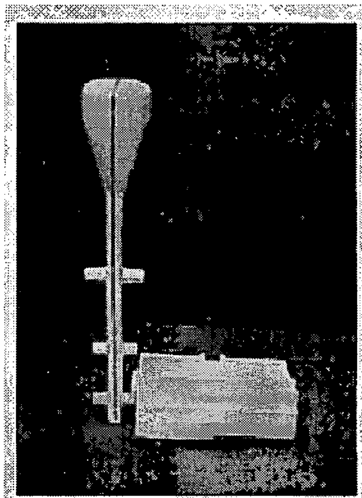


(b)

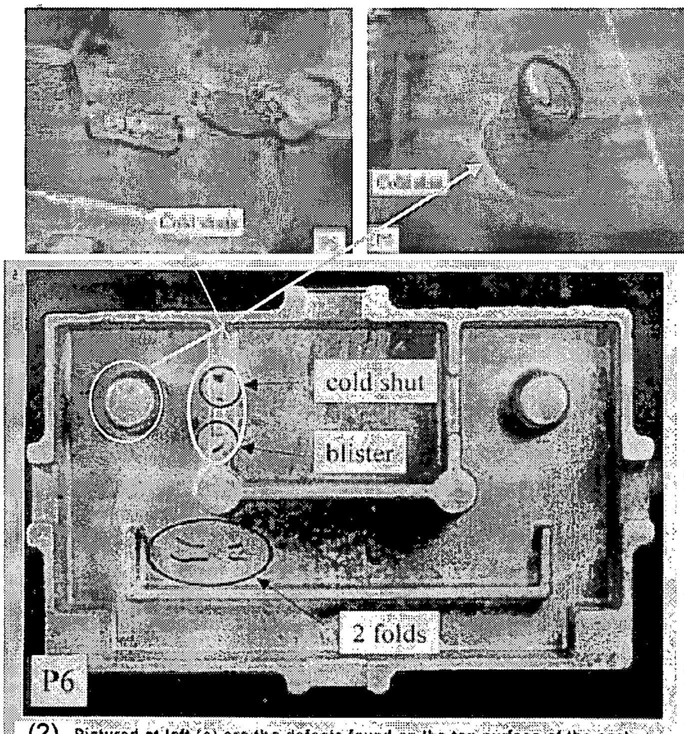
Figure 2 *Image of a casting slice from an LFC engine block (a) and a simulation image from the same area (b). The colored areas in the simulation image represent the predicted macro-porosity.*

(2) C.W. Hirt and M.R. Barkhudarov, *Predicting Defects in Lost Foam Castings*, Modern Casting, December 2002, pp 31-33
C. W. Hirt., *Predicting Defects in Lost Foam Castings*, December 13, 2001 in Flow Science Inc. CFD Library.

The results of the FLOW-3D code are shown in Figure 3.
An analysis object is shown in Figure3-(1). The Al is poured into this mold.
Figure 3-(2) and (3) show the experimental and FLOW-3D results. The red contour of a FLOW-3D result indicates the highest probability for the defect.
The red region is the last place to be filled with Al-alloy.
From these results, it is found that the defeat location of the FLOW-3D code is agreement with that of the experimental result.



(1) Pictured is the experimental box casting foam pattern with gating system attached.



(2) Pictured at left (a) are the defects found on the top surface of the cast component. At right (b) are the predicted locations of defects on the top surface. Cold shuts are indicated by arrows.

(2) Experimental result

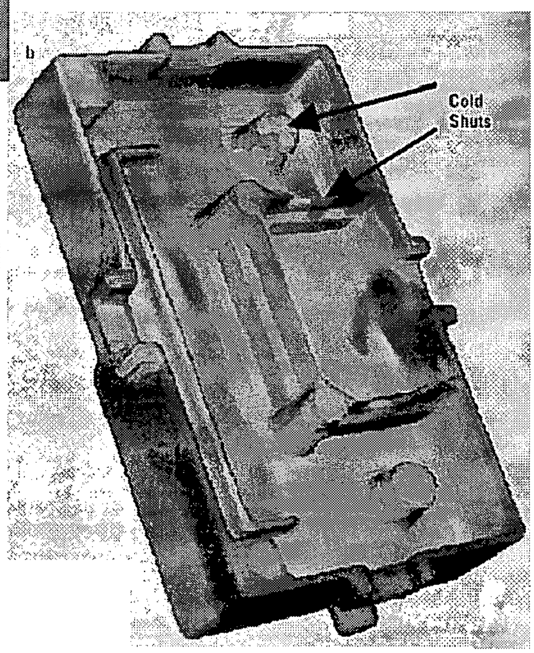


Figure 3 Experimental and CFD results

(3) Predicted location of defects (red). Cold shuts

(3) Flow-3D result

Answer to the additional item (b)

VOF (Volume of Fluid) is explained as follows.

The volume-of-fluid, VOF, method is used to track fluid free surfaces in the fixed grid. The fluid fraction function, F, is defined as equal to one inside the fluid, and to zero otherwise. Figure 4 shows the typical values of the VOF function.

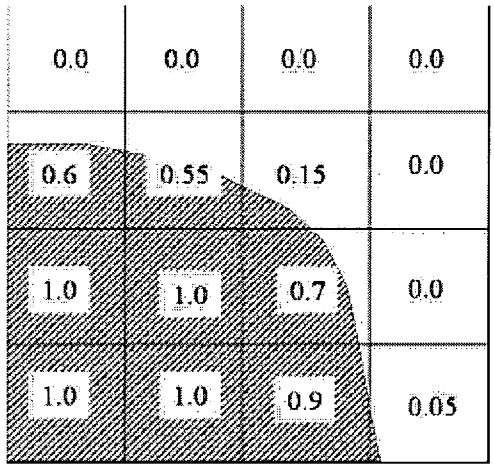


Figure 4 Typical values of the VOF function near free surface.

Averaged over a control volume containing free surface, the value of the fluid fraction falls into the range between zero and one, and, in general, will vary in time and space as the fluid moves through the computational domain. The kinematic transport equation for the VOF function is

$$\frac{\partial F}{\partial t}+\frac{1}{V_f}\nabla\cdot(F\vec{u}A_f)=0 \qquad \text{(eq.1)}$$

where Vf is volume fraction, u is velocity, Af is Area fraction.

At each time step, the equations of rigid body motion are solved if the object moves in a coupled fashion. Effects of hydraulic force (pressure and shear stress), gravitational force, non-inertial force and control force on coupled motion are considered. Control forces are extra forces and torques that can be applied to a moving object, for example, the engine thrust on a boat. Locations and orientations of all moving objects are tracked, and area and volume fractions are updated accordingly. Figure 5 illustrates how the volume and area fraction coefficients are computed in a two-dimensional rectangular mesh, as well as the source term on the right-hand side of Eq. (2).

$$-\frac{\partial V_f}{\partial t}=\vec{U}_{obj}\cdot\vec{n}S_{obj}\big/V_{cell} \qquad \text{(eq.2)}$$

where Uob j, n and Sobj are surface area, surface area normal vector and velocity of moving object boundary in a mesh cell respectively.

Within a control volume the solid boundary is approximated as a planar interface allowing for a simple evaluation of the unit normal and surface area for each such surface element. The FLOW-3D VOF method provides the means to impose accurate boundary conditions for fluid flow and heat transfer. Because of its high-order representation of the geometry, the total surface area and volume of a solid object approaches the exact value as the grid spacing is reduced.

Explicit approximations are used for the advective and viscous terms in the momentum equation. Eq. (3).

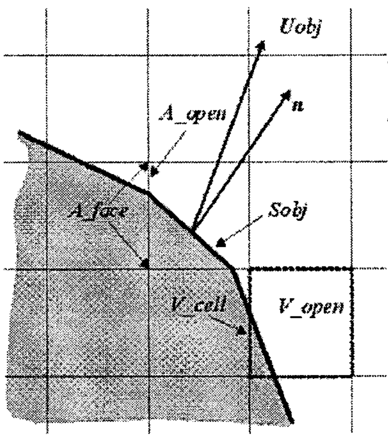
$$\frac{\partial \vec{u}}{\partial t} + \frac{1}{V_f}\left(\vec{u}A_f \cdot \nabla \vec{u}\right) = -\frac{1}{\rho}\left[\nabla p + \nabla \cdot (\tau A_f)\right] + \vec{G} \qquad \text{(eq.3)}$$

Here, p is pressure. τ is the viscous stress tensor and G is gravity.

The continuity equation, Eq. (4), is coupled with the momentum equation using the predictor-corrector method. An appropriate iteration scheme is then used to find the new pressure and velocity values.

$$\frac{V_f}{\rho}\frac{\partial \rho}{\partial t} + \frac{1}{\rho}\nabla \cdot (\rho \vec{u} A_f) = -\frac{\partial V_f}{\partial t} \qquad \text{(eq.4)}$$

Equation (1) is solved using a high-order interface tracking numerical scheme that employs geometric reconstruction of the interface to maintain accuracy during its advection



Volume fraction: $Vf = V\_open / V\_cell$

Area fraction: $Af = A\_open / A\_face$

Source term: $U_{obj} \cdot nS_{obj} / V_{cell}$

Figure 5. Schematic showing the calculation of area and volume fraction coefficients, A_f and V_f, for a solid object (shaded area) imbedded in a rectangular grid. The calculation of the source term for the continuity equation is shown for the central control volume.

Reference

- Flow Science Technical Note #63,
- M. Barkhudarov, and G. Wei, Modeling of the Coupled Motion of Rigid Bodies in Liquid Metal, Modeling of Casting, Welding and Advanced Solidification Processes - XI, May 28 - June 2, 2006, Opio, France, eds. Ch.-A. Gandin and M. Bellet, pp 71-78, 2006.)

Answer to the additional item (c)

The reference for "Previously performed studies…" is the following one.

B. D. Michel, B. Piar, F. Babik, J.-C. Latché G. Guillard and C. De Pascale, SYNTHESIS OF THE VALIDATION OF THE CROCO V1 SPREADING CODE,
OECD Workshop on Ex-Vessel Debris Coolability Karlsruhe, Germany, November 15-18, 1999.
Organised in collaboration with Forschungzentrum Karlsruhe (FZK) GmbH

In this material, it is reported that molten metal temperature distribution by assuming a heat transmission coefficient from 1.76×10^2 to 4.40×10^2 Btu/hr-ft^2-°F is agreement with that of experimental results such as a VULCANO VE-U1 tests.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-162

The conclusions based on the OECD MCCI experiments in Section 15.4.3.1 of the PRA on debris coolability need to be clearly tied to the specific experimental data. Please cite the exact data and experiments that have shown that debris was quenched and coolable, or identify and justify any other method that was used.

ANSWER:

As discussed in Section 15.4.3.1 of the PRA report, observations from two experiments, i.e. COTELS and OECD MCCI, are extracted below, as:

Extracts from COTELS experiment: (H. Nagasaka, et al., COTELS Project (3): Ex-vessel Debris Cooling Tests, Proceedings of OECD Workshop on Ex-Vessel Debris Coolability, Karlsruhe, Germany, 15-18 November 1999)

> *4. Conclusion*
>
> *...*
>
> *3) Debris, in which a decay heat generation was simulated, was cooled from about 20 minutes after water injection and MCCI was suppressed in all test cases.*
>
> *4) This favorable cooling was attributed to the remaining porosity of debris due to simulation of debris falling process, water penetration via eroded concrete side wall clearance and channels and the interruption of further concrete floor erosion due to the existence of accumulated pebble bed decomposed from concrete below the debris.*

Extracts from OECD MCCI experiment: (M.T. Farmer, et al. The Results of the CCI-2 Reactor Material Experiment Investigating 2-D Core-concrete Interaction and Debris Coolability, Proceedings of the 11th International Topical Meeting on Nuclear Reactor Thermal-hydraulics (NURETH-11), Avignon, France, 2-6 October 2005)

3. Results and Discussion

... (Page 9/11 last paragraph)
In general, this test exhibited a high rate of cooling in comparison to previous one-dimensional MACE tests conducted with LCS concrete. Although the reasons for this additional cooling are still under investigation, one mechanism that was clearly active in CCI-2 was that water was able to penetrate the interface between the corium and concrete sidewalls, thereby augmenting the overall debris cooling rate. This same mechanism was also observed in the COTELS reactor material tests, which investigated 2-D core-concrete interaction and debris coolability, albeit at very high specific power density relative to CCI-2 (Nagasaka et al., 1999). However, this additional cooling mechanism was not observed in MACE test M1b, which was conducted with inert refractory (MgO) sidewalls (Farmer et al., 2000).
...

... (Page 10/11 second paragraph)
In any case, the collection of information from this test (i.e., power supply response, steam formation rate due to quenching, and melt temperature thermocouple readings) all indicate a rapid corium cooling rate. In general, the data appear to indicate that late-phase cavity flooding may have the potential for significantly cooling the core material and terminating the accident sequence for LCS concrete. This effective cooling process thus minimized the fact that the insertable lance was not able to break the crust earlier in the test, thereby providing data on the transient crust breach cooling mechanism.

Nagasaka, H., *et. al.* 1999. COTELS project (3): ex-vessel debris cooling tests. *Proc. OECD Workshop on Ex-Vessel Debris Coolability*, Karlsruhe, Germany, November 15-18.

Farmer, M. T., Spencer, B. W., Kilsdonk, D. J., Aeschlimann, R. W. 2000. Results of MACE corium coolability experiments M0 and M1b. *Proc. 8th Int. Conf. on Nucl. Eng., ICONE-8175*, April 2-6.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-163

In Section 15.4.2.3 of the PRA, the film boiling heat transfer coefficient is listed as α = 8.81 × 10 Btu/hr-ft2-°F, please provide the missing power (exponent) of 10.

In addition, reference is made to the use of FLOW-3D; however, the submittal does not contain any information on the modeling features, solution methods, and the experimental validation and technical basis for the code. Please provide the technical details of the FLOW-3D computer code, including its experimental validation basis for application to debris pool spreading.

ANSWER:

The missing power (exponent) of 10 is one. That is $\alpha = 8.81 \times 10^1$ Btu/hr-ft^2-°F.

The technical details of the FLOW-3D code such as verification and validation are discussed in the answer to the question #19-161. The analytical technique is additionally described below.

In order to solve the core melt behavior such as the core melt velocity, core melt density and core melt temperature etc., the differential equations are solved. In the FLOW-3D calculation, differential equations consist of the mass continuity and the momentum equation, the energy equation and the volume fraction (α) continuity. When α is one, fluid is the core melt. On the other hands, when α is zero, fluid is the water. The VOF technique is mentioned in the answer to the question #19-161.

A simplified model for solid-to-liquid phase change can be constructed using the porous media drag concept. The solidification processes (i.e., the state of zero flow velocity) can be approximated by using a drag coefficient that is a function of the local solid fraction.

The drag should be effectively infinite when material is in the solid phase. At the intermediate states consisting of a mush, the drag should assume an intermediate value. In the FLOW-3D code, the drag coefficient is defined as

$$K = \frac{F_s^2}{(1-F_s)^3}$$

where F_s is the local solid fraction.

In a transition domain, ($I_{SC} < I_1 < I_{Sl}$), the local solid fraction F_s is defined as,

$$I_1 = I_{SC} + Fs \cdot hsc$$

where,I_1 is the specific internal energy of solid material(J/kg), I_{SC} is the specific internal energy from the Solidus curve , I_{Sl} is the specific internal energy from the liquidus curve and h_{sc} is the latent heat of solidification (= I_{Sl} - I_{SC}).

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-164

Please justify why the violation of acceptance criterion for basaltic concrete (case M1-2) in PRA Section 15.4.4 L/CS is not considered to be significant.

ANSWER:

Firstly, evaluation for the L/CS concrete case satisfies the determined acceptance criterion, so that there is no problem at all.

As for the basaltic concrete case, it is true that the evaluation result cannot satisfy the determined acceptance criterion; however the acceptance criterion is defined based on a very conservative assumption, i.e. containment failure is assumed when molten debris reaches the steel liner plate. In reality, containment does not immediately fail its role as a leak-tight barrier when molten debris reaches the steel liner plate. It is because there is very thick basemat concrete underneath the steel liner plate, which works as the barrier against the release of fission products to the environment. What is stated in SECY-93-087 as the recommendation by the staff for the containment performance is that "the containment should maintain its role as a reliable, leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products." In that sense, the containment including the basemat concrete still maintains its role as a leak-tight barrier against the uncontrolled release of fission products more than 24 hours, and it is therefore concluded that the slight violation of the conservatively determined acceptance criterion is not very significant.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO. 19-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-165

Please provide a list of various analysis cases that have been considered for ex-vessel steam explosions. Please include the following information for each case:

- · Debris pour composition
- · Lower head hole size
- · Pour temperature
- · Pour velocity
- · Cavity water temperature
- · Cavity water depth
- · Location of RV failure (middle or at the side)

For each case, please provide the peak pressure and the impulse load on the cavity wall.

ANSWER:

This question will be answered by 19 January 2009.

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QUESTION NO.: 19-166

The assessment of ex-vessel steam explosions considers two potential containment failure modes, namely (1) displacement of primary system loops and connecting steam generators challenging containment penetrations and (2) dynamic loads on the reactor cavity wall structure. Dynamic structural responses for both modes are assessed using the LS-DYNA code (a non-linear finite element analysis program developed by Livemore Software Technology Corp). Describe the method of qualifying the LS-DYNA code for this application in terms of validation and verification. Describe special modeling, if any, added to the code for the US-APWR analyses.

ANSWER:

MHI provides three kinds of verification reports for validation and verification of the LS-DYNA code as attachments shown below.

Attachments:

(1) Verification report of concrete material properties for the LS-DYNA code
(2) Verification report of the basic functions of the LS-DYNA code
(3) Modeling description of the dynamic structural response analysis by steam explosion

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

Attachment (1) to Question#19-166

Verification report of concrete material properties for the LS-DYNA code

Attachment (2) to Question#19-166

Verification report of the basic functions of the LS-DYNA code

Modeling description of the dynamic structural response analysis by steam explosion

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QUESTION NO.: 19-167

The probability of an ISLOCA is described to be negligible due to design provisions to mitigate this accident. Please provide a more quantitative basis in regard to the design and the probabilities in support of the omission of the ISLOCA accident from the PRA.

ANSWER:

Similar question regarding the probability of an ISLOCA was asked by the NRC staff before in the question #19-86 of RAI#40-610 as:

> *It is stated (DCD Section 19.1.3 "Special Design/Operational Features") that the residual heat removal system (RHRS) piping is designed to withstand a higher pressure than operating plants. This design feature was used in modeling interfacing systems LOCA in the PRA. It was assumed that following a break of the reactor coolant system (RCS) boundary at the RHRS suction or injection lines, the reactor coolant will flow to the refueling water storage pit (located inside the containment) unless a break occurs at the RHRS piping outside the containment. The probability that a break will occur at the RHRS piping outside containment, given RHRS over-pressurization, was based on a piping rupture rate of 1.5E-10/hr-ft. Even though the RHRS piping is designed to withstand higher pressures than operating plants, this rupture rate is not applicable to piping that may be pressurized above its design capability. Please provide information that justifies the assumed rupture rate for over-pressurized RHRS piping. Also, please summarize the interfacing systems LOCA risk evaluation in Chapter 19 of the DCD by including (1) important results, (2) risk insights regarding the design and operational features which contribute to the low risk associated with interfacing systems LOCA, and (3) key assumptions made in the analysis.*

MHI answered to this question as:

> *US-APWR is designed so that the residual heat removal system (RHRS) pressure does not exceed its critical pressure and RHRS break due to overpressure does not occur in case a*

break of the reactor coolant system (RCS) boundary at the RHRS suction or injection lines happens. It is also designed that pressure rise due to leakage from RCS boundary isolation valve can be mitigated by RHRS relief valves. It is therefore considered appropriate to apply a generic piping rupture rate to RHRS break. Even if the pipe rupture rate is assumed to be 100 times larger considering the severe condition, the frequency of an interfacing system LOCA (ISLOCA) is estimated to be 3E-10/yr. This simple evaluation can conclude that the sensitivity of the RHRS piping rupture rate is insignificant. Overall, the treatment of an ISLOCA for the US-APWR PRA can be summarized as following.

(1) The occurrence frequency of an ISLOCA is evaluated as negligible.

(2) A design Feature that RHRS piping withstands higher pressure than operating plants and that the reactor coolant following a break of the RCS boundary at the RHRS suction or injection lines flows into the in-containment refueling water storage pit (RWSP) can significantly reduce the risk due to an ISLOCA.

(3) One of the key assumptions is that the occurrence frequency of the RCS boundary break at the RHRS suction or injection lines is limited to the same level with operating plants.

Above discussions will be incorporated into the revised technical report 1) and summarized in the next revision of the DCD.

1) US-APWR Probabilistic Risk Assessment, MUAP-07030, Mitsubishi Heavy Industries, December. 2007.

These question and answer sufficiently cover the newly asked question #19-167.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

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APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO. : 19-168

The Accident Classes (ACLs) and plant damage states (PDSs) do not have specific identification of accident progression timing such as early or late. Please explain if it is the intent of MHI to include the Level 3 PRA reported in FSAR Chapter 19.2 as part of the PRA in Chapter 19.1 in a future revision.

ANSWER:

MHI understands the point of this question by the NRC staff is about the timing of the FP release to the environment, whether early release or late release. In that sense, it is more appropriate to discuss about the release categories defined in the US-APWR Level 3 PRA report (MUAP-080004-P Rev. 1), instead of the ACLs or PDSs asked in the question.

The definition of release categories (RC) for the US-APWR PRA is shown in Table 19-168-1. RC1 though RC 4 are classified as early containment failure, quantitatively it is defined as the containment failure within 24 hours after onset of core damage. RC5 is classified as late containment failure, i.e. containment fails at more than 24 hours after onset of core damage. RC6 is defined as intact containment, and FP release is expected with the design release rate.

MHI does not intend to include the Level 3 PRA as part of the PRA in Chapter 19 of the DCD in future revision. The DCD includes SAMDA evaluations required in 10 CFR 50.34(f). The complete SAMDA analysis is reported in the Applicant's Environmental Report - Standard Design Certification (MUAP-DC021, Rev.1). In addition, the supporting analysis is reported in the US-APWR Level 3 PRA report (MUAP-08004, Rev.1).

In addition, in case the MHI's understanding for this question is not the intention of the NRC staff, brief discussion on the timing related to the ACLs and PDSs are also provided. The ACLs and PDSs have specific identification for the timing. In the ACLs, for example, there is distinction on early or late core damage in comparison to containment failure, i.e. the identification "E" and "L" represent the accident sequences whose core damage occurs before and after containment failure, respectively (see the page 19.1-17 of the DCD, Rev.1). In the PDSs, for example, there is distinction between early and late flooding in the reactor cavity in comparison to reactor vessel (RV) failure, i.e. the identification "3", "6, "9" and "2", "5", "8" represent the flooding in the reactor cavity before and after RV failure, respectively (see the page 19.1-45 of the DCD, Rev.1).

Table 19-168-1 Definition of Release Categories

Release Category	Description
RC1	Containment bypass which includes both core damage after steam generator tube rupture (SGTR) and thermal induced SGTR after core damage
RC2	Containment isolation failure
RC3	Containment overpressure failure before core damage due to loss of heat removal
RC4	Early containment failure due to dynamic loads which includes hydrogen combustion before or just after reactor vessel failure, in-vessel and ex-vessel steam explosion, rocket-mode reactor vessel failure, and direct containment heating
RC5	Late containment failure which includes containment overpressure failure after core damage, hydrogen combustion long after reactor vessel failure, and basemat melt through
RC6	Intact containment in which fission products are released at design leak rate

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

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APPLICATION SECTION: **19**

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QUESTION NO.: 19-169

MAAP version 4.06 is used to support the U.S. APWR Level 2 PRA. Describe the method of qualifying the code for this application in terms of validation and verification. Also, please describe special modeling, if any, added to the code for the US-APWR analyses.

ANSWER:

MAAP for PWR is used for severe accident analysis for the US-APWR. It is described in the introduction of MAAP users' manual that MAAP/PWR is configured to analyze PWR designs built by several vendors including Mitsubishi. Fundamental equipment composition of the US-APWR is same as existing Mitsubishi PWR plants, and therefore it is judged that MAAP can be employed for the severe accident progression analyses of the US-APWR.

The US-APWR involves several evolutionary designs from the existing Mitsubishi PWR plants, such as elimination of the low head injection system and introduction of the advanced accumulator system. Japanese original design feature that is not considered in the original MAAP model also exists, i.e. the alternate containment cooling by containment fan cooler system. In order to model the advanced accumulators, plant parameters prepared in the MAAP code are adjusted. The alternate containment cooling is modeled to adjust the input parameters for the MAAP4 originally established fan cooler model.

- Advanced accumulators

The advanced accumulator system takes two important roles, one is ordinary accumulators, and the other is a substitution of the low head injection system. The injection flow rate from accumulators is automatically controlled by the flow damper according to the amount of water inside the accumulator, and it achieves these two roles. In the MAAP analysis, accumulator system is modeled by adjusting the resistance of flow path according to the water math in accumulator to realize the function of the flow damper.

- Alternate containment cooling

The alternate containment cooling is achieved by employing the containment fan cooler system. Hence, it is modeled as one of the fan coolers for the US-APWR MAAP model, i.e. correlation of inlet flow rate to the cooling coil and the corresponding heat removal capacity. The containment heat removal capacity through the fan cooler system under severe accident condition is incorporated into the MAAP4 originally established containment fan cooler model.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-170

GOTHIC7.2a-p5(QA) code is used for certain containment analyses in support of the Level 2 PRA. Please present a brief discussion on the special capabilities of this code, emphasizing the benefits and limitations of its application to complement and supplement MAAP. Please describe the method of qualifying the code for this application in terms of validation and verification, and describe special modeling, if any, added to the code for the US-APWR analyses.

When describing the capabilities of the code related to hydrogen burn in containment, please explain why MAAP 4.0.6 was not used in the accident progression analyses (see the assumptions in Section 14.2.1 of the PRA), and why the igniter model was not included in the developed MAAP model.

ANSWER:

(1) Discussion on the special capabilities of this code, emphasizing the benefits and limitations of its application to complement and supplement MAAP

Primarily MAAP code is used for the severe accident progression analysis for the US-APWR since it has extensive capability and great advantage to predict the severe accident phenomena. However MAAP code also involves a disadvantage in calculation accuracy that the fluid flow model assumes the quasi-static flow without solving a momentum equation in order to shorten the calculation time. On the other hand, the GOTHIC code evaluates the gas flow more accurately by solving momentum equations.

As for the hydrogen mixing and combustion analysis, it is considered that the prediction of transitional flow of hydrogen is one of the key to be evaluated. It is therefore decided that GOTHIC code is more suitable for hydrogen mixing and combustion analysis rather than MAAP code.

In the severe accident progression analysis employing MAAP code, the analysis is performed with the condition that the model of hydrogen burning is forced off. The potential maximum hydrogen concentration in various scenarios is evaluated from these analyses. The evaluation results are utilized to determine the probability of containment failure by hydrogen combustion under the conditions that igniters are not operable.

(2) Method of qualifying the code for this application in terms of validation and verification

Regarding the GOTHIC verification and validation against the hydrogen mixing and combustion analyses, appropriate benchmark analyses have been performed:

1) The NUPEC large-scale hydrogen mixing tests, Battelle-Frankfurt model containment tests, Heissdampfreaktor (HDR) full scale containment tests and Hanford Engineering Development Laboratory (HEDL) hydrogen mixing tests for hydrogen mixing phenomena

2) The Flame facility experiments and the Nevada Test Site experiments for hydrogen combustion

These results are described in the GOTHIC qualification report (Reference 1).

(3) Special modeling added to the code for the US-APWR analyses

There is no special modeling added to the code for the US-APWR analyses.

Reference

1 GOTHIC Containment Analysis Package Qualification Report Version 7.2a, NAI 8907-09 Revision 9, January 2006.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-171

A "Summary of Level 2 PRA Results from Internal Events at Power" is presented in PRA Table 17.3-9. It is not clear if internal floods and fires are included in these internal events. This question also applies to the results and tables in FSAR Section 19.1.

Please clarify this in the related sections of the PRA report.

ANSWER:

Please refer to the table of contents of the PRA report (MUAP-07030) stated as "PART 1 INTERNAL EVENTS RISK ASSESSMENT", this part includes chapters 2 to 21. And "PART 2 EXTERNAL EVENTS RISK ASSESSMENT", this part includes chapters 22 to 25, in which internal flood (chapter 22), internal fire (chapter 23), seismic (chapter 24) and other external events (chapter 25) are discussed.

Table of contents of the DCD chapter 19 also clearly states the topics of discussions, as the following:

19.1.4 Safety Insights from the Internal Events PRA for Operations at Power
- 19.1.4.1 Level 1 Internal Events PRA for Operations at Power
- 19.1.4.2 Level 2 Internal Events PRA for Operations at Power

19.1.5 Safety Insights from the External Events PRA for Operations at Power
- 19.1.5.1 Seismic Risk Evaluation
- 19.1.5.2 Internal Fires Risk Evaluation
- 19.1.5.3 Internal Flooding Risk Evaluation

19.1.6 Safety insights from the PRA for other modes of operation

...

The document structure of the DCD chapter 19 strictly follows the instructions of the Standard Review Plan (NUREG-0800 Revision 2) and the Regulatory Guide 1.206.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-172

In PRA Section 15.4.2, the FLOW-3D calculation of the corium spreading behavior on the US-APWR cavity floor is discussed. Previously performed studies are notes as showing that the heat transfer coefficient between the molten corium and the floor decreased as a result of crust formation. The PRA analysis however assumed no crust formation as conservatism.

(a) Please reference and summarize these previous studies.

(b) Discuss the conservatism of this assumption.

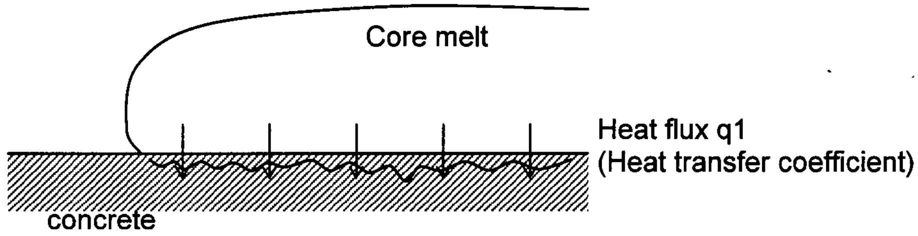
(c) Explain how this assumption is evaluated or included in the sensitivity studies performed for corium cooling and concrete erosion.

ANSWER:

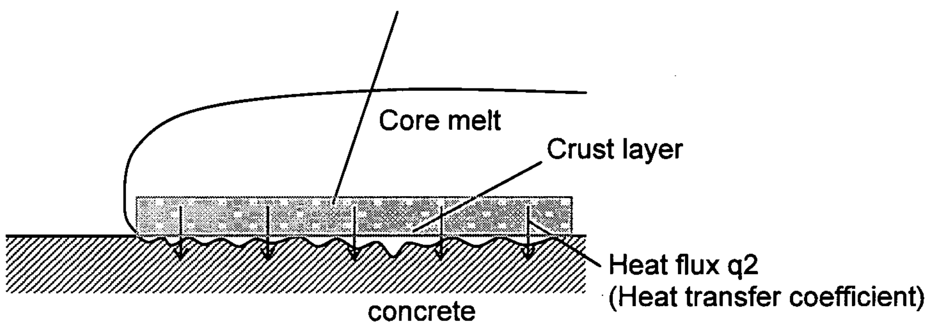
(a) Previous studies and their references are same as the ones answered to the question #19-161.

(b) Figure 19-172-1 shows schematic view of the core melt cooling behavior. When the crust layer is not formed, the core melt contacts directly with the concrete cavity floor. As a result, the cooling heat flux of the core melt is relatively large. On the other hands, when the crust layer is formed, the cooling heat flux of the core melt decreases. Because the thermal conductivity of the crust layer is smaller than that of the core melt.

In no crust formation case, the core melt is relatively rapidly cooled down and solidifies more quickly compared with that of the crust formation case. Therefore, as for the core melt coolability and spreading behavior, the PRA analysis assumed no crust formation for the FLOW-3D code is considered conservative evaluation.



(1) Heat flux of core melt to concrete without crust layer



(2) Heat flux of core melt to concrete with crust layer

The thermal conductivity of the crust layer is generally smaller than that of the core melt. Therefore the cooling heat flux of the core melt decreases when the crust layer is formed.

Figure 19-172-1 Consideration of core melt cooling with or without cavity floor concrete erosion

(c) As sensitivity analysis, a heat transfer coefficient between the concrete cavity and the core melt is set to 1760 Btu/hr-ft^2-°F. In this case, it is confirmed that the core melt spreads thinly in a cavity like a case of a designed heat transfer coefficient 792 Btu/hr-ft^2-°F.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-173

In PRA Section 15.4 "Core Debris Coolability and Molten Core Concrete Interaction", the capability of the US-APWR design to withstand the effects of molten corium release from the vessel is analyzed through calculations based on MAAP and FLOW-3D. A summary of early relevant studies is presented in support in Table 15-3. Since then, more experiments, analyses, and models have been developed. Please present an updated analysis of this phenomenon for the US-APWR using current technology and analyses in support of the numerical quantifications presented, or justify why no further analysis is needed.

ANSWER:

The summary table 15-3 widely covers the studies on debris cooling from the relatively early stage (middle of 80s) to the very recent studies (2007). MAAP code version 4.0.6 employed for the US-APWR severe accident evaluation was released at 2005 and it was the latest version of the MAAP4 series code when MHI started designing the US-APWR at 2006. FLOW-3D code version 9.0, utilized for analyzing the debris spreading behavior, was released at 2005, and this version was also the latest at 2006. The discussion on the phenomena related to the core debris coolability is presented in Subsection 15.4.3.1 of the PRA report as:

"Experiments performed relatively early stage of a research, such as SWISS and MACE, concluded that solid crust layer or rapidly reformed crust is formed and core debris is uncoolable. However, recently performed studies such as COTELS and OECD MCCI support the broken-up behavior of formed crust layer and concluded debris is quenchable and coolable. These experimental observations and conclusions are modeled in the computer codes. MAAP code supports the crust broken up behavior as long as coolant water sufficiently exists; on the other hand, MELCOR code supports the hard crust formation phenomenon and models uncoolable debris."

MHI considers that the experimental results of COTELS and OECD MCCI can be applicable to the US-APWR design. MHI supports the calculations obtained through MAAP code, and the results are incorporated into the plant design. On the other hand, MHI also agrees that these phenomena are still controversial and they may largely involve inherent uncertainty. Therefore several sensitivity studies were performed as described in Sections 15.4.3 and 15.4.4 of the PRA report.

The evaluation results on the core debris coolability and molten core concrete interaction for the US-APWR are derived in accordance with these achievements of the various studies from mid 80s to very recent 2007, and also based on the numerical evaluations utilizing the latest version of codes. MHI considers the overall analyses on the core debris coolability and molten core concrete interaction are performed applying the state-of-the-art techniques.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-174

(a) The "No" column of PRA Table 17.3-10 starts at Number two, and there is no Number one. Please explain or correct this error.

(b) The note to FSAR Table 19.1.50 says that the uncertainty sources are categorized into three types. Only two are listed, and "completeness" seems to have been omitted or deleted. Please explain or correct this error.

ANSWER:

(a) It is a typographical error. The erroneous numbers in the table will be amended to start from one in the next revision.

(b) It is also a typographical error and there are only two uncertainty sources for the Level 2 PRA. Uncertainty from completeness was identified only for Level 1 PRA but was not categorized for Level 2 PRA. The error will be amended in the next revision.

Impact on DCD

The erroneous description will be amended in the next revision.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-175

Please confirm whether the overall LPDS analyses were only for the Level 1 PRA model and did not include the CSETs or the CPETs.

ANSWER:

Correct. LPSD analyses for the US-APWR PRA do not involve Level 2 calculation. It is conservatively assumed that the containment is kept open during LPSD operation. In reality, this assumption is not always correct because the condition of containment isolation depends on the plant operation states (POSs). However, for the US-APWR Level 2 PRA, the detailed evaluation considering the POSs is eliminated for simplicity; and it assumes that core damage during LPSD regardless of the POSs immediately results in large release of the fission products to the environment. For the LPSD operation, the LRF is therefore equivalent to the CDF, i.e. the CCFP is unity.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-176

Some CET results are missing from PRA (MUAP-07030 R0) Table 17.3-7. Sheet 5 appears to be a duplicate of Sheet 4. Please provide corrected Sheet 5.

ANSWER:

It is a typographical error in the revision 0 of the MUAP-07030. The error was corrected in the revision 1 submitted to the NRC at September 2008.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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QUESTION NO.: 19-177

The information on the Level-3 PRA is provided outside of the FSAR (MUAP-08004-P). It is, however, pertinent to the evaluation of severe accident mitigation design alternatives. This analysis provides information on release categories, source terms and consequence analysis. For consequence analysis, the study primarily uses the data extracted from the MACCS2 manual for the majority of the site-specific input data, including the regional economic data.

(a) The analysis indicates consequence analysis was limited to releases occurred within 24 hours after the onset of core damage. Please provide technical bases for not considering releases after 24 hours, given that the release durations have been broken into four plumes, with the final plume release exceeding the 24 hours.

(b) For release category 5, late containment failure, since releases occur after 24 hours, no consequence analysis was performed. Please provide clarification on mitigation measures that eliminate the need for consequence analysis for this release category.

(c) It is not clear how the limiting 24 hour release duration affects various durations used in the MACCS2 sample problem. For example, the duration of emergency phase in the sample problem is one week. It is not clear if the analysis in this report used similar duration as that of the sample problem. In addition, release duration in MACCS2 should be less than 10 hours. The plume release duration data in the report indicates values in excess of 10 hours. Please provide MCCAS2 input files used for the consequence analysis.

(d) Please provide equilibrium core inventory source term given potential for high burnup.

(e) The property data used in the economic loss is essentially those of 1970 values. Please explain how the current property values would affect the results presented for offsite property damage.

(f) For release category 6, intact containment, the release would be continuous and in the order of technical specification limit. Please provide the details of how this release is modeled using MACCS.

ANSWER:

(a) In the level 3 PRA evaluation used in SAMDA of US-APWR DCD Chapter 19, the off-site exposure dose risk and off-site property damage risk 24 hours after core damage have been evaluated. That is, the evaluation period is determined to be 24 hours after core damage in accordance with the target stipulated in [ ]

If the evaluation period is extended from 24 hours after core damage to 72 hours after core damage, the off-site exposure dose risk increases by[ ]compared to that for 24 hours as shown in Table 19-177(1), while the off-site property damage risk increases also by[ ]. However, the proportion of off-site cost (for off-site exposure dose risk and off-site property damage risk) for the SAMDA maximum averted cost benefit is insignificant, with an increase by only[ ]as evaluated for 72 hours after core damage. It is thus estimated that the effect of off-site cost will also be small even for the SAMDA maximum averted cost benefit, even when taking one week as an evaluation period. Also the result of maximum averted cost benefit evaluated for 72 hours after core damage shows a sufficiently small value compared to $870 k in case of installation of a redundant containment spray system with the minimum capital investment. It is thus understood to reasonably take 24 hours after core damage as evaluation period.

Table 19-177(1) Effect of Evaluation Period on Maximum Averted Cost Benefit

( )

(b) The release category (RC) 5 in the case of late containment failure is characterized by the fact that containment failure occurs sufficiently late, causing late release of radionuclide into the environment after the onset of core damage. In the case of a typical sequence for US-APWR RC 5, release of radionuclide starts [ ] after core damage even without taking any mitigation measures, which means that no release occurs within 24 hours after core damage. This is the reason why no consequence analysis was performed, as mentioned in Item (a) above. Even when the evaluation period is extended to 72 hours after core damage, the contribution of RC 5 to the off-site exposure dose risk is as small as [ ] of the total risk as indicated in Table 19-177(2), while its contribution to the off-site property damage risk is [ ]. Still, as explained under Item (a), the extension of evaluation period to one week is considered to have negligible effect on SAMDA.

Table 19-177(2) Off-site Exposure Dose Risk and Property Damage Risk (Evaluation Period : 72 hours)

[ ]

(c)MHI provides MHI's limiting MAACS2 input decks for US-APWR at Power as the following files:

DOSDATA.INP	Dose Conversion factor data file
INDEXR.DAT	Decay chain data file
Maccs2.exe	Execution module
METSUR.INP	Meteorological data

(d) MHI provides fission product core inventory at the end of cycle for US-APWR in Table 19-177 (3).

Table 19-177 (3) Fission Product Core Inventory for US-APWR (Ci/core)

Nuclide	Core Inventory	Nuclide	Core Inventory
Co-58	0.00E+00	I-135	2.27E+08
Co-60	4.26E+05	Xe-133	2.44E+08
Kr-85	1.70E+06	Xe-135	6.89E+07
Kr-85m	3.04E+07	Cs-134	3.32E+07
Kr-87	5.79E+07	Cs-136	9.05E+06
Kr-88	8.14E+07	Cs-137	1.89E+07
Rb-86	3.33E+05	Ba-139	2.16E+08
Sr-89	1.11E+08	Ba-140	2.09E+08
Sr-90	1.36E+07	La-140	2.18E+08
Sr-91	1.38E+08	La-141	1.97E+08
Sr-92	1.50E+08	La-142	1.90E+08
Y-90	1.44E+07	Ce-141	1.99E+08
Y-91	1.44E+08	Ce-143	1.82E+08
Y-92	1.51E+08	Ce-144	1.61E+08
Y-93	1.76E+08	Pr-143	1.79E+08
Zr-95	2.07E+08	Nd-147	7.97E+07
Zr-97	2.10E+08	Np-239	2.54E+09
Nb-95	2.09E+08	Pu-238	7.32E+05
Mo-99	2.27E+08	Pu-239	5.53E+04
Tc-99m	1.99E+08	Pu-240	8.67E+04
Ru-103	1.90E+08	Pu-241	1.92E+07
Ru-105	1.32E+08	Am-241	2.59E+04
Ru-106	7.38E+07	Cm-242	6.42E+06
Rh-105	1.23E+08	Cm-244	7.80E+05
Sb-127	1.34E+07	U-230	1.17E-05
Sb-129	3.95E+07	U-231	3.55E-03
Te-127	1.33E+07	U-232	1.69E+00
Te-127m	1.77E+06	U-233	6.95E-04
Te-129	3.89E+07	U-234	3.34E+00
Te-129m	5.80E+06	U-235	3.38E+00
Te-131m	1.76E+07	U-236	4.93E+01
Te-132	1.71E+08	U-237	1.61E+08
I-131	1.21E+08	U-238	4.32E+01
I-132	1.74E+08	U-239	2.54E+09
I-133	2.43E+08	U-240	2.60E+03
I-134	2.66E+08	U-241	4.34E-03

(e) Economic effects are naturally influenced by variations in commodity prices. The consumer price index (CPI) is an indication of variations in the price of commodities and services that people daily purchase. The CPI in the United States is available in the following HP:

http://www.bls.gov/cpi/home.htm

The CPI assesses price variations with the expenditure base for 1982 to 1984 as 100, and on this basis, the CPI for September 2008 is about 220. The US-APWR level 3 RPA analysis used the sample input for MACCS2 that corresponds to the data for 1970 to 1980. On the expenditure base for 1982 to 1984 as 100, the CPI corresponding to this time period is assumed to be in the range of 40 and 80, which is about 1/5 to 1/3 of the current value. For example, the current off-site property damage risk will be about five times the maximum of the 1970 to 1980 data. As Table 19-177(4) shows, the increase in SAMDA maximum averted cost benefit is [ ], even when taking a five times off-site property damage risk. It is thus understood that the possible effect on SAMDA maximum averted cost benefit will be small enough, even when changing the 1970 to 1980 data for economic impact to that for 2008.

Table 19-177(4) Effect of Property Data on Maximum Averted Cost Benefit

[ ]

(f) The division of the plume used in MACCS2 is generated from the release fraction and the release rate of the radionuclide evaluated by the MAAP code in Level 2 PRA.
The release category (RC) 6 in the case of intact containment assumes the containment leakage of radionuclide. The source term analysis result of RC6 by the MAAP code has the following characteristics:

a) The release of the radionuclide is begun after the core damage at once.

b) The release of the radionuclide other than the noble gas will almost end in[ ] after the core damage.

c) The release of the noble gas is continued at constant after end of the release of the radionuclide other than the noble gas.

The source term data of MACCS2 set the start time and the end time of the plume from the above-mentioned character as Table 19-177(5).

If the evaluation period for RC6 is extended from 24 hours after core damage to 72 hours , the contribution of RC6 to the off-site exposure dose risk is as small as [ ] of the total risk as shown in Table 19-177(6). The contribution of RC6 to the off-site property damage risk is also as small as[ ]. That is, RC6 has only a negligible effect on SAMDA.

Table 19-177(5) Source Term Input Data for US-APWR RC 6 at Power

Plume No.	Start time	End time
1	Start time of release	About 3.5 hours after the core damage
2	End time of first plume	24 hours after the core damage
3	End time of second plume	24 hours after the start time of this plume
4	End time of third plume	24 hours after the start time of this plume

Table 19-177(6) Off-site Exposure Dose Risk and Property Damage Risk (Evaluation Period : 72 hours)

[ ]

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.92-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-178

The regulations in 10 CFR 52.47(b)(2) and 10 CFR 51.55(a) require the applicants to prepare an environmental report that includes a cost and benefits of severe accident mitigation design alternatives (SAMDA). The environmental report supporting the SAMDA analysis refers to a level 3 PRA that was performed to determine the overall risk perspective of the US APWR design (MAUP-DC201). The analysis uses a standard method (template) as provided in the NEI 05-01 for SAMDA analysis in support of license renewal. Review of the methods and assumptions has identified the following:

(a) Table 2 of the ER provides a list of SAMDA candidates. The table does not provide any details on assumptions and basis for screening/modification evaluation. Please provide the assumptions and basis for the screening of candidate SAMDAs.

(b) One screening criterion is labeled "not a design alternative." All except a few of these SAMDA items related to procedures and training. Please identify the screened out list that needs to be considered by COL holders of US APWR design. Also, please explain why the SAMDA items related to improving uninterruptable power supplies and enhancing control of combustible and ignition sources are screened under this criterion.

(c) NUREG/BR-0184 provides a range of doses and costs for occupational exposure, onsite clean up costs, and replacement power costs. The report used only the best estimate values with 3 and 7 per cent discount rates to estimate the range of potential averted costs. Given that these estimates are dated (1992 circa), please elaborate why the analysis did not consider potential uncertainties in these values, and evaluate the impacts on the cost benefit analysis.

(d) Table 12 of the ER provides a sensitivity analysis for the selected SAMDA items benefits. The report indicates that each SAMDA item benefit is calculated using ratio of each item contribution to decrease CDF or large release frequency (LRF). Please explain where these ratios are provided.

ANSWER:

(a) The screening criteria for "Not applicable", "Already implemented" and "Not a design alternative" are clearly identified. For instance, the design alternatives related to the ECCS switchover to the recirculation mode can be judged as no benefit because the US-APWR is designed involving in-containment RWSP and the necessity of ECCS switchover to the recirculation mode is eliminated from the design basis; and hence those design alternatives are screened out as "Not applicable". Regarding other three screening criteria, "Excessive implementation cost", "Very low benefit" and "Combined", design alternatives are evaluated in accordance with the following manners:

- Every group of Potential Enhancement (e.g. dc and ac power, core cooling, etc.) is examined whether it is related to the system which has meaningful influence to the currently evaluated risk or not. If the group is evaluated having no or little meaningful influence, then it is judges as "Very low benefit."

- If the group is evaluated having meaningful influence to the risk, the effectiveness of risk reduction by implementing the design alternative is examined. It is evaluated as the ones with good effectiveness are given priority to others. If one design alternative of the group is evaluated as potentially very good risk reduction although it is also evaluated as obviously "Excessive implementation cost", then it is screened out and subordinately effective design alternatives are evaluated. It is possible that individual design alternative is not very effective for risk reduction but combination of several design alternatives shows significant risk reduction effectiveness. Therefore it is necessary to consider the reasonable combinations of design alternatives, and the combination should be evaluated whether it is effective in risk reduction or not. For example, individual dc battery is not evaluated as significant; however it shows good effectiveness if it is combined with ac power source.

- For other design alternatives, relative evaluation is performed with ones with priority, and screened out to be assigned as "Combined", "Excessive implementation cost" or "Very low benefit."

(b) The list of candidate design alternatives screened out as "Not a design alternative" is shown in Table 19.178-1. Regarding the SAMDA items related to improving uninterruptable power supplies and enhancing control of combustible and ignition sources, the evaluation of the achievability, effectiveness of risk reduction, cost impact, etc. is considered highly dependent on the site specific conditions. Therefore it is determined as practically impossible to evaluate appropriately at the DC stage, and screened out as "Not a design alternative."

(c) SAMDA analysis for the US-APWR has been performed in accordance with the guidance provided in NEI 05-01 Rev A. In the chapter 8 of NEI 05-01 "Sensitivity analyses", six candidates are described and each item is examined if sensitivity analysis is necessary or not. As a result only the real discount rate is selected to be analyzed and other five candidates are excluded according to the following reasons:

8.1 Plant modifications

There is no significant modification such as power uprate because this is DC application phase and the standard design has not been approved yet.

8.2 Uncertainty

The CDF applied in this evaluation is 1.2E-6/ry (value for internal event at power) and the 95 percentile value from the uncertainty analysis is calculated as 2.9E-6/ry. Even if the evaluated cost is increased proportionally to the ratio of CDF, i.e. multiplied by 2.4, maximum averted cost does not exceed the cheapest design alternative of $870K.

8.3 Peer review findings or observations

There are no findings or observations from the peer review which causes significant increase in CDF greater than the one that maximum averted cost exceeds the cheapest design alternative of $870K. It is approximately three times greater CDF value than currently evaluated 1.2E-6/ry.

8.4 Evacuation speed

Evacuation is not taken into account in this SAMDA analysis, so that sensitivity analysis on evacuation speed is not applicable.

8.5 Real discount rate

A value of 7% real discount rate as the baseline analysis and values of 5% and 3% real discount rate as the sensitivity analysis are evaluated in accordance with the NEI 05-01 and NUREG/BR-0184 guide, respectively.

8.6 Analysis period

Because 60 years of a US-APWR plant life is assumed in SAMDA analysis, sensitivity analysis is not necessary.

In addition to these 6 potential candidates suggested in the NEI 05-01 guidance, the sensitivity for the monetary value of unit exposure is evaluated. The value of $2000 per person-rem is utilized for the basic evaluation as mentioned in NUREG/BR-0184, and additionally $3000 per person-rem is evaluated as the sensitivity case shown in the Table 19.2-9 of the DCD Rev. 1.

(d) The "ratios of each item contribution" questioned by the NRC are not described in any documents submitted for review, i.e. DCD, environmental report and PRA technical report. The selected design alternatives are examined whether it is effective in risk reduction for the highly ranked dominant sequences in PRA, and if the design alternative is evaluated as effective, then the contribution of the sequence is approximately evaluated as if the sequences can be eliminated from risk calculation in a conservative manner. And also if the combination of the design alternatives is evaluated as effective in risk reduction, then the combination is evaluated in a same manner with the individual design alternatives. The approximate evaluation results are shown in Table 19.178-2.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

Table 19.178-1 List of design alternatives screened out as "Not a design alternative" (1/2)

Phase II ID	Phase I SAMDA ID	Potential Enhancement (NEI-05-01)	Screening
dc and ac power			
	010	Revise procedure to allow bypass of diesel generator trips.	Not a design alternative
	016	Improve uninterruptible power supplies	Not a design alternative
	018	Develop procedures for replenishing diesel fuel oil	Not a design alternative
	021	Develop procedures to repair or replace failed 4 KV breakers	Not a design alternative
	022	In training, emphasize steps in recovery of off-site power after an station blackout (SBO)	Not a design alternative
	023	Develop a severe weather conditions procedure	Not a design alternative
Core Cooling			
	027	Revise procedure to allow operators to inhibit automatic vessel depressurization in non-ATWS scenarios	Not a design alternative
	036	Emphasize timely recirculation alignment in operator training.	Not a design alternative
	042	Make procedure changes for reactor coolant system depressurization	Not a design alternative
Cooling water			
	045	Enhance procedural guidance for use of cross-tied component cooling or service water pumps	Not a design alternative
	049	Enhance loss of component cooling water (or loss of service water) procedures to facilitate stopping the reactor coolant pumps	Not a design alternative
	050	Enhance loss of component cooling water procedure to underscore the desirability of cooling down the reactor coolant system prior to seal LOCA	Not a design alternative
	051	Additional training on loss of component cooling water	Not a design alternative
	061	Change procedure to isolate reactor coolant pump seal return flow on loss of component cooling water, and provide (or enhance) guidance on loss of injection during seal LOCA	Not a design alternative
	062	Implement procedures to stagger high pressure safety injection pump use after a loss of service water	Not a design alternative
Feedwater and Condensate			
	073	Proceduralize local manual operation of auxiliary feedwater system when control power is lost	Not a design alternative
Heating, Ventilation, and Air Conditioning			
Instrument Air and Nitrogen Supply			
	086	Modify procedure to provide ability to align diesel power to more air compressors	Not a design alternative
Containment			
	103	Institute simulator training for severe accident scenarios	Not a design alternative

Table 19.178-1 List of design alternatives screened out as "Not a design alternative" (2/2)

Phase II ID	Phase I SAMDA ID	Potential Enhancement (NEI-05-01)	Screening
Containment Bypass			
	117	Revise EOPs to improve ISLOCA identification	Not a design alternative
	118	Improve operator training on ISLOCA identification	Not a design alternative
	119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage	Not a design alternative
	123	Proceduralize use of pressurizer vent valves during steam generator tube rupture sequences	Not a design alternative
	127	Revise emergency operating procedures to direct isolation of a failure steam generator	Not a design alternative
	128	Direct steam generator flooding after a steam generator tube rupture, prior to core damage	Not a design alternative
ATWS			
Internal Flooding			
Seismic			
Fire			
	145	Enhance fire brigade awareness	Not a design alternative
	146	Enhance control of combustibles and ignition sources	Not a design alternative
Others			
	150	Improve maintenance procedures	Not a design alternative
	151	Increase training and operating experience feedback to improve operator response	Not a design alternative
	152	Develop procedures for transportation and nearby facility accidents	Not a design alternative
US-APWR Specific: Internal Flooding			
US-APWR Specific: Fire			

Table 19.178-2 Evaluation of risk reduction by the candidate design alternatives (1/2)

	Sequence Name	Sequence Freq. (/ry)	Percent Contrib.	DC battery	GT/G	Offsite power	HHIP	ESWP	Seal Inj. Pump	CCWP	EFWP	Filtered Vent	CV Spray
Internal at power													
1	LOOP_C-OPS-ADG-PRB-PRC-SEL	5.0E-07	42%	42%	42%	42%	42%		42%				
2	LOCCW-SCA-SEL	2.6E-07	22%				22%	22%	22%	22%			
3	RVR	1.0E-07	8%										
4	LOOP_A-CWR-SCO1-SEL	6.2E-08	5%			5%	5%		5%				
5	SLOCA-HIB-CSA-CRB	4.2E-08	4%										
6	LOCCW-EFA-SEL	4.0E-08	3%				3%	3%	3%	3%	3%		
7	LOFF-EFA-FBA	2.3E-08	2%								2%		
8	SLBO-MSO-BLA	1.7E-08	1.4%										
9	SLOCA-HIB-SRA	1.5E-08	1.3%										
10	TRANS-EFA-MFW-FBA1	1.4E-08	1.2%								1.2%		
11	LOOP_A-EFO-FBA2	1.2E-08	1.1%			1.1%					1.1%		
12	ATWS-ROD-MTC	1.0E-08	0.8%										
13	SLOCA-CXB-FNA2	8.6E-09	0.7%										
14	LOCCW-SRV	6.4E-09	0.5%										
15	SLOCA-HIB-CRB	6.2E-09	0.5%										
16	MLOCA-HIB-CSA-CRD	5.8E-09	0.5%										
17	PLOCW-SCK-SEL-CSA-CRB2-FNA7	5.1E-09	0.4%										
18	FWLB-EFD-BLA	4.3E-09	0.4%								0.4%		
19	SLOCA-CSA-CRB-FNA2	4.2E-09	0.4%										
20	MLOCA-HIB-CRD	3.8E-09	0.3%										
21	PLOCW-EFA-BLA	3.3E-09	0.3%								0.3%		
22	MLOCA-ACA	3.2E-09	0.3%										
23	ATWS-RPS-DAS-MTC	3.0E-09	0.3%										
24	SLOCA-HIB-CSA-CRB-FNA2	2.8E-09	0.2%										
25	LOOP_B-OPS-CWR-SCO1-SEL	2.4E-09	0.2%				0.2%		0.2%				
26	LOAC-EFA-MFW-FBA1	2.1E-09	0.2%								0.2%		
27	LOOP_D-OPS-ADG-EFO-PRB-SEL	2.1E-09	0.2%	0.2%	0.2%	0.2%	0.2%		0.2%				
28	SGTR-SGI-HT	1.9E-09	0.2%										
29	LOOP_D-OPS-ADG-EFO-CWR-SEL	1.9E-09	0.2%	0.2%	0.2%	0.2%	0.2%		0.2%				
30	SGTR-SGI-PZR	1.8E-09	0.2%										
31	SGTR-SGI-SRB	1.7E-09	0.1%										
32	PLOCW-SCK-SEL-HIC-SRA2	1.7E-09	0.1%										
33	PLOCW-EFA-HIC	1.6E-09	0.1%								0.1%		
34	MLOCA-HIB-SRA	1.3E-09	0.1%										
35	SLBI-EFD-BLA	1.3E-09	0.1%								0.1%		
36	MLOCA-CXC-FNA1	1.2E-09	0.1%										
37	LOOP_A-SRV-CWR	1.0E-09	0.1%			0.1%							
38	ATWS-SCF-DAS	1.0E-09	0.1%										
39	LODC-EFA-MFW-FBA1	9.0E-10	0.1%								0.1%		
40	PLOCW-SCK-SEL-HIC-CSA-CRB2-FNA7	8.6E-10	0.1%								0.1%		
			99%	42%	42%	49%	73%	25%	73%	25%	9%	0%	0%
	Total	1.2E-06		5.1E-07	5.1E-07	5.9E-07	8.7E-07	3.0E-07	8.7E-07	3.0E-07	1.1E-07	0.0E+00	0.0E+00
Internal fire													
1	Yard	1.2E-06	67%	67%	67%	67%	67%		67%				
2	FA6-101-01	1.0E-07	6%								6%		
3	FA6-101-04	8.4E-08	5%	5%	5%	5%	5%		5%				
4	FA4-101	4.6E-08	3%								3%		
5	FA2-205	4.6E-08	3%								3%		
6	FA2-202	4.4E-08	2%								2%		
7	FA3-104	3.7E-08	2%								2%		
8	FA2-205- M-05	3.7E-08	2%				2%	2%		2%			
			89%	71%	71%	71%	73%	2%	71%	2%	15%	0%	0%
	Total	1.8E-06		1.3E-06	1.3E-06	1.3E-06	1.3E-06	3.7E-08	1.3E-06	3.7E-08	2.7E-07	0.0E+00	0.0E+00
Internal flooding													
1	FA2-102-01 (Major flood at)	1.7E-07	12%					12%		12%	12%		
2	FA2-108-01 (Flood)	1.7E-07	12%					12%		12%	12%		
3	FA2-102-01 (Flood)	1.5E-07	11%					11%		11%	11%		
4	FA2-108-01 (Major flood)	1.5E-07	11%					11%		11%	11%		
5	FA2-414-01 (Major flood)	1.4E-07	10%								10%		
6	FA2-415-01 (Major flood)	1.3E-07	9%								9%		
7	FA2-414-01 (Spray)	7.3E-08	5%								5%		
8	FA2-501-03 (Flood)	3.7E-08	3%					3%		3%	3%		
9	FA2-501-01 (Flood)	3.7E-08	3%					3%		3%	3%		
10	FA2-415-01 (Spray)	3.3E-08	2%								2%		
11	FA2-102-01 (Spray)	3.1E-08	2%								2%		
12	FA2-108-01 (Spray)	1.3E-08	1%								1%		
13	FA2-112-01 (Major flood)	1.3E-08	1%					1%		1%	1%		
14	FA2-501-11 (Flood)	1.3E-08	1%					1%		1%	1%		
15	FA2-206-02 (Major flood)	1.2E-08	1%					1%		1%	1%		
16	FA2-407-04 (Flood)	1.2E-08	1%					1%		1%	1%		
17	FA2-501-11 (Major flood)	1.1E-08	1%					1%		1%	1%		
18	FA2-407-04 (Major flood)	1.1E-08	1%					1%		1%	1%		
19	FA2-201-02 (Major flood)	1.1E-08	1%					1%		1%	1%		
20	FA2-407-01 (Major flood)	1.0E-08	1%					1%		1%	1%		
			87.64%	0%	0%	0%	0%	58%	0%	58%	88%	0%	0%
	Total	1.4E-06		0.0E+00	0.0E+00	0.0E+00	0.0E+00	8.1E-07	0.0E+00	8.1E-07	1.2E-06	0.0E+00	0.0E+00
LPSD													
1	Loss of coolant accident	1.3E-07	65%				65%						
2	Loss of offsite power	3.8E-08	19%	19%	19%	19%	19%						
3	Loss of CCWS/ESWS	2.3E-08	12%					12%		12%			
4	Loss of RHRS caused by failing to maintain water level	6.1E-09	3%				3%						
5	Loss of RHRS caused by other failures	3.9E-09	2%				2%						
6	Loss of RHR due to over-drain	2.0E-09	1%				1%						
				19%	19%	19%	90%	12%	0%	12%	0%	0%	0%
	Total	2.0E-07		3.8E-08	3.8E-08	3.8E-08	1.8E-07	2.3E-08	0.0E+00	2.3E-08	0.0E+00	0.0E+00	0.0E+00

Table 19.178-2 Evaluation of risk reduction by the candidate design alternatives (2/2)

	Plant Damage State	PDS Freq. (/ry)	Percent Contrib.	DC battery	GT/G	Offsite power	HHIP	ESWP	Seal Inj. Pump	CCWP	EFWP	Filtered Vent	CV Spray
Internal at power(Level 2 PRA)													
1	3D	4.0E-08	38%									38%	
2	4K	1.8E-08	17%									17%	
3	4D	1.6E-08	15%										
4	3A	6.6E-09	6%										
5	4L	6.0E-09	6%										
6	4H	5.3E-09	5%										
7	9A	2.8E-09	3%										
8	6D	2.4E-09	2%									2%	
9	1K	1.6E-09	2%									2%	
10	5A	1.3E-09	1%										
11	5E	1.0E-09	1%										
12	4J	1.0E-09	1%										
13	1J	9.5E-10	1%										
14	3H	6.7E-10	1%									1%	
15	3C	5.1E-10	0%										
16	1D	2.1E-10	0%										
17	7D	1.6E-10	0%										
18	7J	1.1E-10	0%										
19	6H	8.9E-11	0%									0%	
20	3E	6.9E-11	0%										
				0%	0%	0%	0%	0%	0%	0%	0%	60%	0%
	Total	1.0E-07		0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	6.0E-08	0.0E+00

	Basic Event ID	Freq. (/ry)	FV Importance	DC battery	GT/G	Offsite power	HHIP	ESWP	Seal Inj. Pump	CCWP	EFWP	Filtered Vent	CV Spray
Internal at power(Level 2 PRA)													
1	ACWOO02FS	-	3.2E-01										
2	ACWOO02CT-DP2	-	3.1E-01										
3	OPS----PRBF	-	2.1E-01										
4	OPS----PRCF	-	2.1E-01										
5	OPSRSB	-	1.8E-01										
6	EPSCF4CBTD6H-ALL	-	1.5E-01										
7	NCCOO02CCW	-	1.4E-01										
8	RSAOO02FWP	-	1.4E-01										
9	CFAMVFCFSV2	-	8.7E-02										
10	CFAMVFCFSV5	-	8.7E-02										
11	RSSCF4MVOD114-ALL	-	8.1E-02										
12	MSROO02533A	-	5.3E-02										
13	CCWRSA	-	5.2E-02										
14	EPSCF4DLLRDG-ALL	-	5.1E-02										
15	EPSOO02RDG	-	4.9E-02										
16	EFWOO01PW2AB	-	4.9E-02										
17	CHIOO01CHIB	-	3.5E-02										
18	SWSTMPESWPD	-	3.1E-02										
19	OPSLOOP	-	3.0E-02										
20	SWSCF4PMBD-R-ALL	-	2.8E-02										
21	CHIPMBDCHPB-R	-	2.4E-02										
22	RSSTMRPRHEXC	-	2.2E-02										2.2E-02
23	HPIOO02FWBD	-	1.9E-02										
24	EPSCF2SLLRDGP-ALL	-	1.9E-02										
25	RSSCF4PMADCSP-ALL	-	1.9E-02										
26	RSSTMPICSPC	-	1.8E-02										1.8E-02
27	CWSTMRCCWHXD	-	1.8E-02										
28	EFWPTADFWP1A	-	1.7E-02										
29	EFWCF2TPADFWP1-ALL	-	1.6E-02										
30	CWSTMPCCWPD	-	1.5E-02										
31	CWSCF4PCBD-R-ALL	-	1.5E-02										
32	HPIOO02FWBD-S	-	1.5E-02										
33	SGNBTSWCCF	-	1.5E-02										
34	HITOO02-DP3	-	1.5E-02										
35	EPSDLLRDGP1-L2	-	1.4E-02										
36	RWSCF4SUPRST01-ALL	-	1.4E-02										
37	PZROO02PORV-DP3	-	1.4E-02										
38	MSPOO02STRV-SG-DP3	-	1.4E-02										
39	FDAMVFC58MC	-	1.3E-02										
40	FDAMVFC58RC	-	1.3E-02										
41	SWSTMPESWPB	-	1.1E-02										
42	EFWPTADFWP1B	-	1.1E-02										
43	EFWTMTAA	-	1.1E-02										
44	EPSDLLRDGP2-L2	-	1.1E-02										
45	EPSCF4DLADDG-ALL	-	1.1E-02										
46	MSPOO0250A1-DP2	-	1.0E-02										
47	MSPOO0250B1-DP2	-	1.0E-02										
48	MSPOO0250C1-DP2	-	1.0E-02										
49	CCWBTSWCCF	-	9.7E-03										
50	EPSCF4DLLRDG-134	-	9.4E-03										
51	CHICF2PMBD-ALL	-	8.8E-03										
52	EPSCF4DLSRDG-ALL	-	8.1E-03										
53	EPSTMDGP1	-	7.9E-03										
54	RSSOO02LNUP-SG-DP3	-	7.4E-03										
55	HPIOO02FWBD-S-DP4	-	7.4E-03										
56	MSPOO0250A2-DP2	-	7.0E-03										
57	MSPOO0250B2-DP2	-	7.0E-03										
58	MSPOO0250C2-DP2	-	7.0E-03										
59	EFWTMTAB	-	6.7E-03										
60	RSSPMADCSPC	-	6.5E-03										6.5E-03
61	MFWHARD	-	6.4E-03										
62	EFWPTSRFWP1A	-	6.1E-03										
63	BOSBTSWCCF	-	5.7E-03										
64	EPSTMDGP2	-	5.6E-03										
65	NCCOO02CCW-DP2	-	5.5E-03										
66	ACWTMPZCLTP	-	5.0E-03										
				0%	0%	0%	0%	0%	0%	0%	0%	0%	5%
	Total	1.0E-07		0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	4.7E-09

L1 All	4.6E-06		1.8E-06	1.8E-06	1.9E-06	2.4E-06	1.2E-06	2.2E-06	1.2E-06	1.6E-06	0.0E+00	0.0E+00
L2 internal	1.0E-07		0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	6.0E-08	4.7E-09
L1 percentage			40%	40%	41%	52%	25%	47%	25%	35%	0%	0%
L2 percentage			0%	0%	0%	0%	0%	0%	0%	0%	60%	5%
each benefit($k)	289		116	116	118	150	72	136	72	101	173	14

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: NO.92-1237 REVISION 0

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

APPLICATION SECTION: PRA

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO. : 19-179

Section 19.5 of the PRA provides the results of the uncertainty analysis for the LRF. It does not discuss how this calculation was carried out. Please provide a detailed discussion of the analysis including methods and data, especially with respect to the CPET portion of the LRF calculation.

ANSWER:

In the quantification of LRF, distributed split fractions were used in the Level 1 PRA event tree and in the CSET portion, while CPET analysis was carried out by using point-estimated split fractions. Uncertainty in the CPET portion was not considered in the US-APWR PRA. This is because, in the Level 1 PRA event tree and in the CSET portion, the component failures and human action failures are modeled as statistically random events, whereas the CPET deals with the deterministic physical processes (not random events). The uncertainty range of the total LRF is dominated by the higher LRF sequences and the dominant LRF sequences are evaluated as overpressure failure and bypass sequences. Those uncertainties are originated from the system unreliability and not influenced from the phenomenological uncertainty; accordingly the uncertainty originated from CPET is not very significant to the total LRF. Therefore, the CPET split fractions for the analysis are determined as numbers between 0 and 1.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-180

In Table 2.1-9, page 39 of the Level 3 PRA (MUAP-08004-P (R0)) and Figures 2.1-17 and 2.1-21, the release of CsI is significantly higher than that of CsOH. This larger release for CsI appears to be as result of late revaporization of previously deposited aerosols on the RCS structures (evident from Figure 2.1-17). Please explain the reasons for the absence of this revaporization contribution for CsOH resulting in about an order of magnitude lower release for CsOH as compared with CsI.

ANSWER:

About 20 hours after the initiation of accident, deposited aerosol on the RCS structures begins to evaporate, and releases to the environment due to containment isolation failure. At that time, saturation pressure of CsI is approximately equal to that of CsOH at the RCS surface temperature of this case. In the model of MAAP code, it is calculated that partial pressure of each fission product vapor is same as the saturation pressure. Therefore the number of CsI molecular released from the RCS surface to containment is also approximately equal to that of CsOH. However, because the molecular weight of CsI is larger than that of CsOH, the mass of CsI evaporated from the RCS surface is evaluated larger than that of CsOH. On the other hand, the initial mass of CsI is smaller than that of CsOH by a magnitude of approximately one order. Because the molecular weight of CsI is larger and initial mass of CsI is smaller, change rate in terms of the release fraction of CsI is much larger than that of CsOH.

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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

APPLICATION SECTION: 19

DATE OF RAI ISSUE: 11/5/2008

QUESTION NO.: 19-181

In the TMI-2 accident, it could be possible that the instrument tubes failed from oxidation of the Zircaloy cladding, causing steam, hydrogen, and fission products to be released to the containment building before the B-loop pump was restarted (see R. E. Henry's presentation to the MAAP Users' Group on May 7, 2008). Please explain what would be the consequences of such a development in the US-APWR for both high RCS pressure and low RCS pressure severe accident scenarios. In addition, please provide the location of hydrogen, steam, and fission product entry into the containment building, the hydrogen, steam, oxygen, and nitrogen mole fractions in the containment vs. time, the CsI, CsOH, and SrO fractions in the containment, and the containment pressure increase vs. time.

ANSWER:

High and low RCS pressure cases from each of the accident progression analyses and source term analyses are explained in this answer. The request on the discussion for the gas mole fractions and containment pressure are provided from the accident progression analysis cases, and the discussion for source term fractions are provided based on the source term analysis cases. The results of the accident progression cases are explained first, and followed by the results of the source term cases.

(1) Explanation of accident progression in high and low RCS pressure scenarios

(a) High primary system pressure case

Case AP201 in Chapter 14 of MUAP-07030-P (R1) is chosen as a representative high primary system pressure case.

Hydrogen, steam, and fission product are released from the pressurizer relief tank until reactor vessel failure. Hydrogen mole fraction is shown in Figure 14-95 (p.14-83), steam mole fraction is shown in Figure 14-96 (p.14-83). Oxygen and nitrogen mole fractions are not presented in the report; however they can be roughly estimated in the following manner. Concrete erosion in this sequence is not significant and therefore the gas generation due to MCCI is negligible. Assuming the gas composition except hydrogen and steam is maintained as the original state, approximately 20% of the remaining mole fraction can be considered as for oxygen; and 80% are for nitrogen. Containment pressure is shown in Figure 14-93 (p.14-82).

At first, main steam line break occurs. Primary system pressure rises and steam is released to the containment from the pressurizer relief tank. Containment spray and high head injection fail to operate. Primary system water level decreases, core is uncovered at 2.1 hours and damaged at 2.9 hours. After core uncovery, zirconium of clad metal reacts with steam and hydrogen is generated. Reactor vessel fails at 4.8 hours and molten core dropped into the reactor cavity. At this time, primary system is rapidly depressurized and accumulator water flows into the reactor cavity through the reactor vessel breach. Reactor cavity water gradually evaporates due to the decay heat of molten core, and hence the containment steam mole fraction rises, and hydrogen mole fraction decreases relatively. Containment fails at 54 hours because of overpressure.

(b) Low primary system pressure case

Case AP003 in Chapter 14 of MUAP-07030-P (R1) is chosen as a representative low primary system pressure case.

Hydrogen, steam, and fission product are released from the failure opening at horizontal part of cold leg until reactor vessel failure. Hydrogen mole fraction is shown in Figure 14-23 (p.14-47), steam mole fraction is shown in Figure 14-24 (p.14-47). Oxygen and nitrogen mole fractions are not presented in the report; however they can be roughly estimated as explained in above answer. Containment pressure is shown in Figure 14-21 (p.14-46).

At first, medium break LOCA occurs at horizontal part of cold leg. Primary system water and steam is released to the containment from the RCS failure opening. Containment spray system works successfully, but containment spray heat exchanger and high head injection fails to operate. After the end of accumulator injection, primary system water level started decreasing. Core is uncovered at 2.3 hours and damaged at 2.8 hours. After core uncovery, zirconium of clad metal reacts with steam and hydrogen is generated. Reactor vessel fails at 5.4 hours and molten core dropped into the reactor cavity. At that time, water in the reactor cavity flows back into the reactor vessel failure opening, subsequently the primary system water level rises again. Reactor cavity water gradually evaporates due to decay heat of molten core and containment spray heat exchanger fails to operate, the containment steam mole fraction rises, and hydrogen mole fraction decreases relatively. Containment fails at 48 hours because of overpressure.

(2) Explanation of source term in high and low RCS pressure scenarios

(a) High primary system pressure case

Case RC5 in Chapter 2 of MUAP-08004-P (R1) is chosen as a representative high primary system pressure case.

Hydrogen, steam, and fission product are released from the RCP seal until reactor vessel failure. Release fraction of CsI is shown in Figure 2.1-65 (p.56), release fraction of CsOH is shown in Figure 2.1-69 (p.57), release fraction of SrO is shown in Figure 2.1-67 (p.57).

At first, loss of ac power and RCP seal LOCA occurs. Primary system water and steam is released from the RCP seal failure. Because of the loss of ac power, containment spray system and high head injection fail to operate. Primary system water level decreases and cladding is damaged at 2.5 hours. At that time fission product in core begins to release to the containment. Reactor vessel fails at 5.3 hours, and molten core drops into the reactor cavity. Reactor cavity water gradually evaporates due to decay heat of molten core, then containment pressure rise and containment fails at 52 hours. At that time fission product begin to release to the environment. Because the containment spray system does not operate, deposition of aerosol fission product within the containment is not expected.

(b) Low primary system pressure case

Case RC6 in Chapter 2 of MUAP-08004-P (R1) is chosen as a representative low primary system pressure case.

Hydrogen, steam, and fission product are released from the failure opening at horizontal part of hot leg until reactor vessel failure. Release fraction of CsI is shown in Figure 2.1-77 (p.60), release fraction of CsOH is shown in Figure 2.1-81 (p.61), release fraction of SrO is shown in Figure 2.1-79 (p.61).

At first, large break LOCA occurs at horizontal part of hot leg. The containment keeps its integrity and design rate leakage from the containment to the environment is considered. Primary system water and steam is released to the containment from the RCS failure opening. Containment spray system works successfully, but high head injection system fails to operate. Primary system water level decreases quickly because the postulated break size is very large, and cladding is damaged at 20 minutes. At that time fission product in core begins to release to the containment through the RCS failure opening. Fission product is released to the environment at the design leak rate. Reactor vessel fails at 2.6 hours, and molten core fails to the reactor cavity. Molten core is cooled by reactor cavity water, and fission product release to containment and environment almost stops. Because the containment spray system works and the containment is intact, very little amount of fission product is released to the environment.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.: **NO.92-1237 REVISION 0**

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

APPLICATION SECTION: **19**

DATE OF RAI ISSUE: **11/5/2008**

QUESTION NO.: 19-182

Please provide the following design information related to the in-core instrument tubes:

(a) A schematic of a typical in-core instrumentation tube and associated guide tube geometry

(b) The outside guide tube diameter, material, and wall thickness.

(c) The material, the diameter of any casings where the thermocouples and/or neutron detectors are located, the wall thickness, and typical cross sectional drawing of the in-core detector and guide pipe.

(d) The material, its high-temperature yield strength as a function temperature, and its creep-rupture properties (e.g., Larson-Miller parameters) for the in-core instrument guide and other parts of the in-core detectors.

(e) A schematic of the in-core instrumentation guide tubes configuration, their location of entry into the reactor pressure vessel, and the location of entry into the instrumentation room inside the containment. This information should also show the actual location inside the containment (to be accompanied by a drawing of the containment marking the location of the instrumentation room).

(f) Description of instrumentation room compartment (room size), locations and opening areas for (flow) communication with adjacent compartments.

ANSWER:

(a) The schematics of the moveable detector and the thermocouple are shown in Figure 19-182-1 (The Movable Neutron Detector Guide Structure) and Figure 19-182-2 (The Core Exit Thermocouple Guide Structure).

(b) and (c) The typical cross sections of the in-core detector and guide pipes are shown in Figure 19-182-3.

(d) The physical-properties in ASME Section II are applied to the high-temperature yield strength. Based on the experience from the operating performance of domestic PWR plants, MHI considers that the creep-rupture does not occur.

(e) and (f) The actual location inside the containment is shown in Figure 19-182-4 (The location inside the containment). There is not the instrumentation room in the containment.

<u>Impact on DCD</u>

There is no impact on DCD from this RAI.

<u>Impact on COLA</u>

There is no impact on COLA from this RAI.

<u>Impact on PRA</u>

There is no impact on PRA from this RAI.

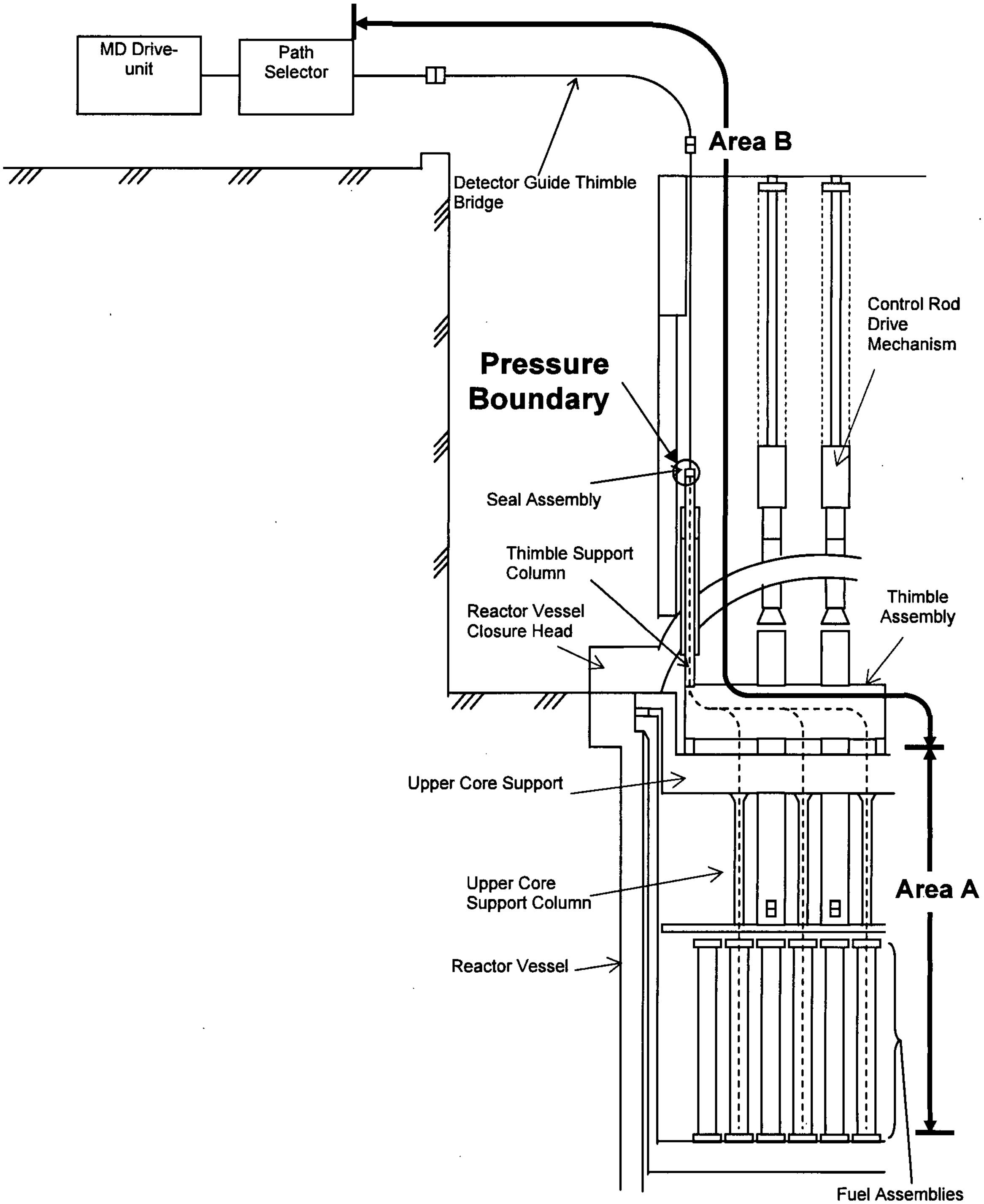

Figure 19-182-1 The Movable Neutron Detector Guide Structure

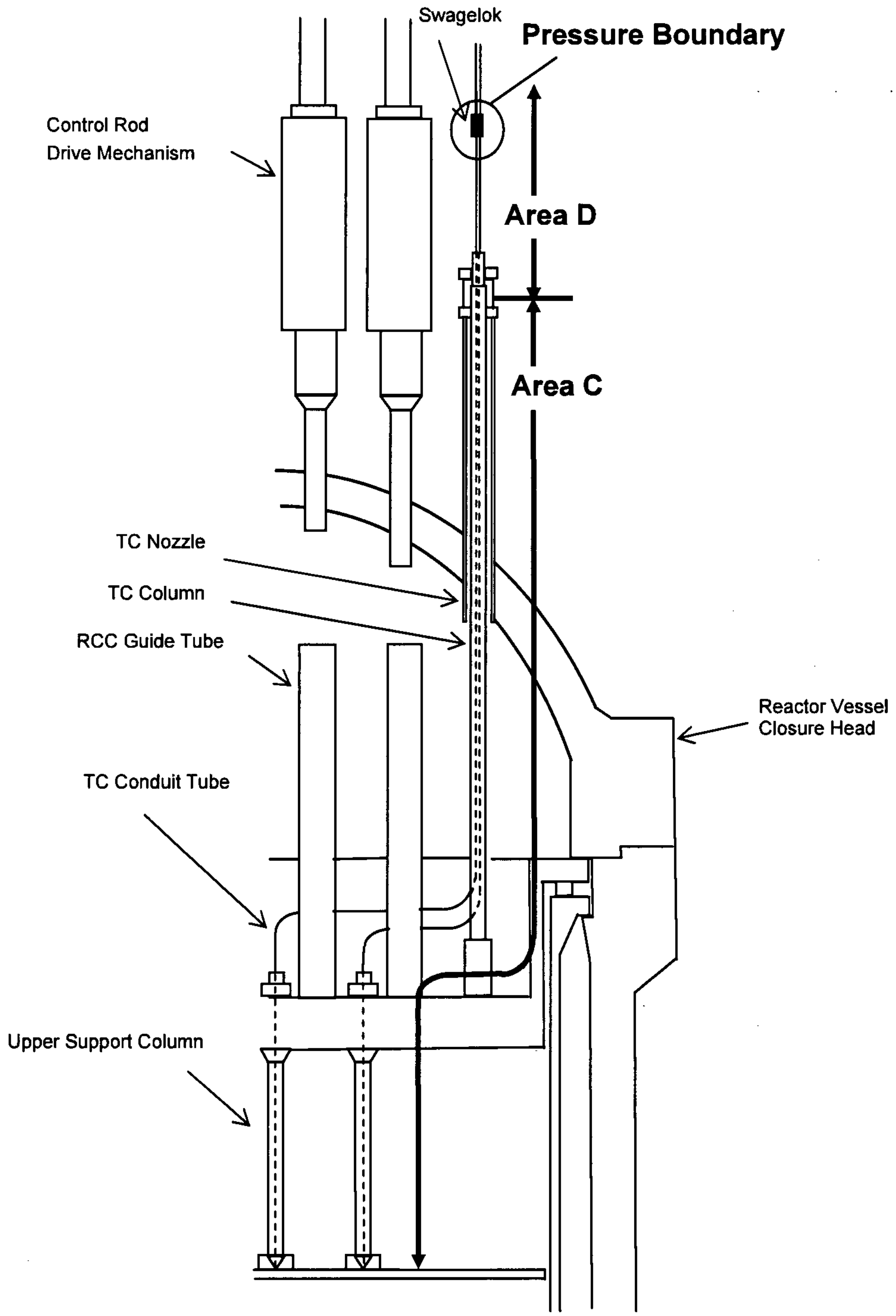

Figure 19-182-2 The Core Exit Thermocouple Guide Structure

Figure 19-182-3 The typical cross sections of the in-core detector

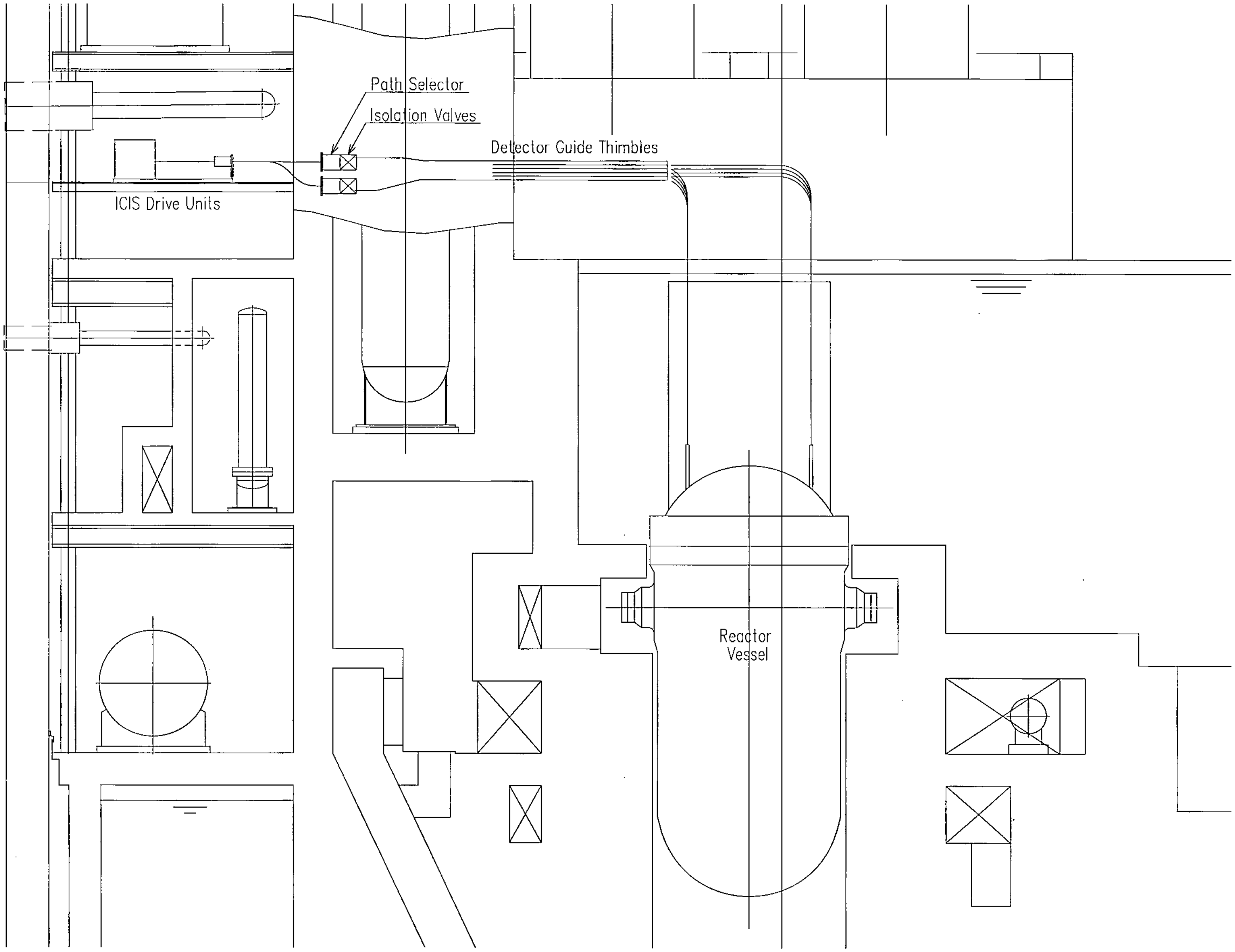

Figure 19-182-4 The location inside the containment (1/2)

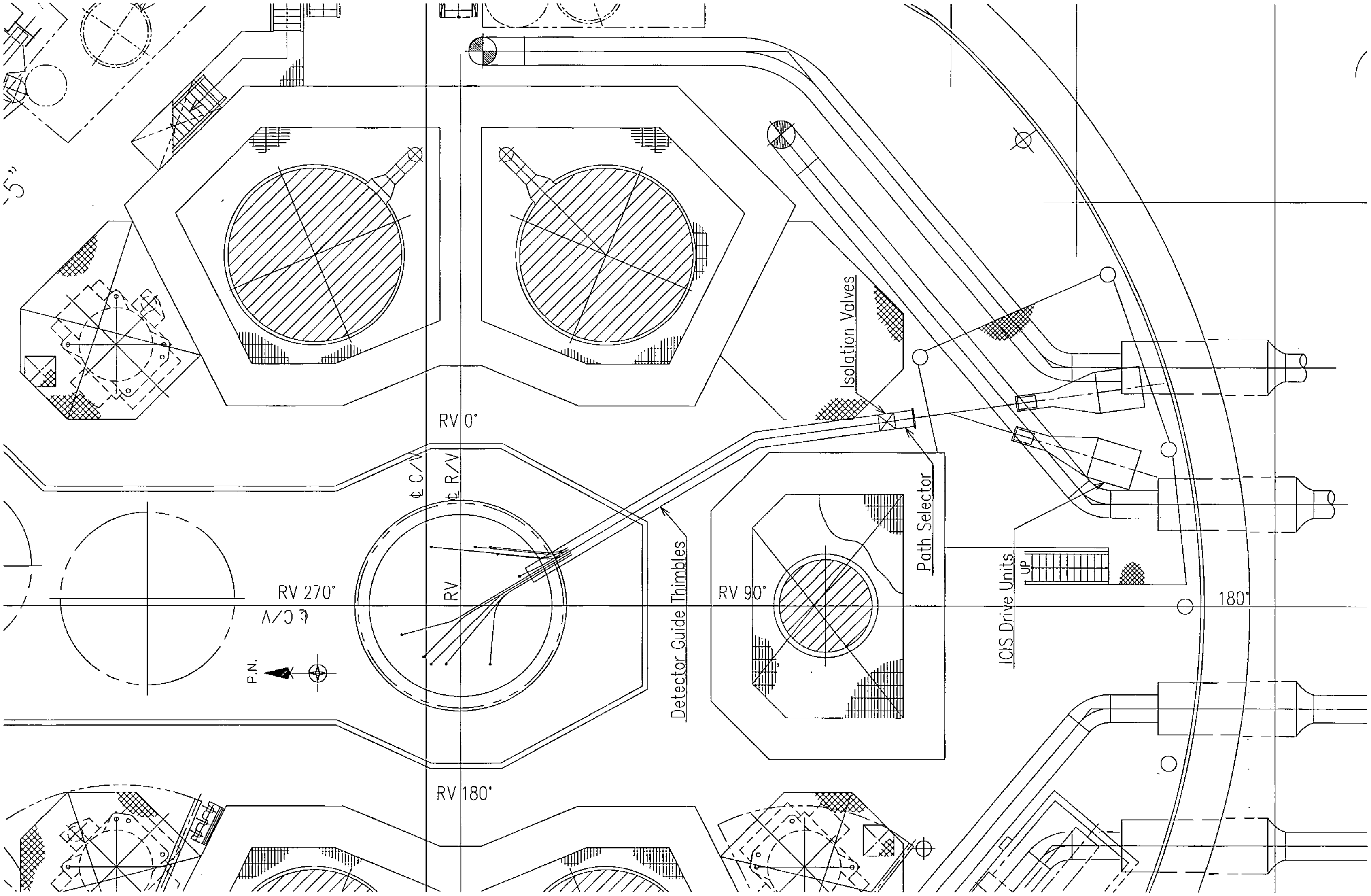

Figure 19-182-4 The location inside the containment (2/2)