MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

August 22, 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-08149

Subject: MHI's Responses to US-APWR DCD RAI No.38

References: 1) "Request for Additional Information No.38 Revision 1, SRP Section: 15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors, Application Section: 15," dated July 24, 2008.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Request for Additional Information No.38 Revision 1."

Enclosed are the responses to 23 RAIs contained within Reference 1.

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 CFR § 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 "[ ]".

And one version includes certain information, designated pursuant to the Commission guidance as sensitive unclassified non-safeguards information, referred to as security-related information ("SRI"), that is to be withheld from public disclosure under 10 CFR § 2.390. The information that is SRI is identified by braces "{ }". On the other hand, another version omits the SRI and is suitable for public disclosure. In the public version of the DCD, the SRI is replaced by the designation "{Security-Related Information - Withheld Under 10 CFR § 2.390}".

This letter includes a copy of the proprietary and SRI included version (Enclosure 2), a copy of the non-proprietary and SRI excluded 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 CFR § 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,

D081
NRO

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional information No.38 Revision 1 (proprietary and SRI included version)

3. Responses to Request for Additional information No.38 Revision 1 (non-proprietary and SRI excluded version)

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

Contact Information
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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-08149

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 CFR § 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 "Response to Request for Additional Information No.38 Revision 1" dated August, 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 CFR § 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 of the safety analysis, developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and the performance of detailed hardware design and software development extending over several years.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive

position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with development of the safety analysis methodology. Providing public access to such information permits competitors to duplicate or mimic the methodology without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced plant safety, and reduced operation and maintenance costs associated with the safety analysis.

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 22nd day of August, 2008.

Y. Ogata

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

Enclosure 3

UAP-HF-08149
Docket No. 52-021

Responses to Request for Additional Information No.38 Revision 1

August, 2008
(Non-Proprietary and Security-Related Information Excluded)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/22/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO.38 REVISION 1**
SRP SECTION:	**15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
APPLICATION SECTION:	**15**
DATE OF RAI ISSUE:	**7/24/2008**

QUESTION NO. : 15.00.03-1

Please identify the dose conversion factors (DCFs) from FGR-11 Table 2.1 that were used to calculate dose equivalent iodine-131 (DEI-131) nuclide-specific reactor coolant concentrations; committed effective dose equivalent (CEDE) or thyroid. Justify which DCFs were used.

ANSWER:

In accordance with Regulatory Guidance (RG) 1.183 Section 4.1.2, MHI uses DCFs of CEDE to estimate radiological consequences. Therefore, MHI uses DCFs of CEDE to calculate dose equivalent iodine-131 (DE I-131). The column labeled "effective" in FGR-11 Table 2.1 corresponds to the CEDE DCFs used by MHI.

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.

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QUESTION NO. : 15.00.03-2

Provide the calculation of the iodine appearance rates listed in DCD Table 15.0-11, including the basis for the inputs and assumptions. Explain why the iodine appearance rate varies between the three accidents identified in the table.

ANSWER:

The amount of iodine released from the primary system is determined by assuming that all of the water released from the primary system during the transient initiated iodine spike period contains the equilibrium concentration of iodine. The release rate when this amount is released at a constant flow for 8 hours is the release rate corresponding to the iodine concentration at the equilibrium value. According to RG 1.183 Appendix E and Appendix F, and SRP 15.6.2, a multiplier of 500 (steam system piping failure and failure of small lines carrying primary coolant outside containment) or 335 (steam generator tube rupture) should be applied to this release rate when analyzing a transient initiated iodine spike. This multiplied release rate is the iodine appearance rate. This calculation is mathematically described below with the specific values used for the various parameters summarized in Table 15.00.03-2-1 on the next page. Since both break flow and break flow end time vary with accident, the iodine appearance rate varies depending on the accident.

$$(500 \ or \ 335) \times [\{(Letdown \ flow) + (RCS \ leak \ rate) + (Break \ flow)\} \times (Break \ flow \ end \ time)$$
$$+ \{(Letdown \ flow) + (RCS \ leak \ rate)\} \times (Remaining \ period \ after \ Break \ flow \ end \ time)]$$
$$\times (Equilibrium \ concentration) / (8 \ hours)$$

Table 15.00.03-2-1 Parameters for the Calculation of Iodine Appearance Rate

	Steam system piping failure	Steam generator tube rupture	Failure of a small line carrying primary coolant outside containment
Letdown flow (gpm)	180*1	180*1	180*1
RCS leak rate (gpm)	11*2	11*2	11*2
Break flow (gpm)	-*3	319*4	97*4
Break flow end time (h)	-*3	1.17	0.75
Remaining period after break flow end time (h)	8	6.83	7.25
Iodine appearance rate			
I-131	2.67×10^2	2.22×10^2	2.79×10^2
I-132	1.43×10^2	1.19×10^2	1.50×10^2
I-133	4.60×10^2	3.82×10^2	4.82×10^2
I-134	9.80×10^1	8.15×10^1	1.03×10^2
I-135	3.01×10^2	2.51×10^2	3.16×10^2

*1 This flow rate is In DCD table 9.3.4-2.
*2 The sum of ildentified and unidentified leakage in technical specification 3.4.13.
*3 In the steam system piping failure, break flow from primary coolant is not taken into account.
*4 These break flow are at a density of 62.4 lb/ft^3.

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.

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QUESTION NO. : 15.00.03-3

Identify the dose conversion factors from FGR-12 Table III.1 (effective dose equivalent or organ dose equivalent) which were used to calculate dose equivalent Xe-133 (DE Xe-133) nuclide-specific reactor coolant concentrations. Justify which DCFs were used.

ANSWER:

In accordance with Regulatory Guidance (RG) 1.183 Section 4.1.4, MHI uses DCFs of effective dose equivalent (EDE) to estimate radiological consequences. Therefore, MHI uses DCFs of EDE to calculate dose equivalent Xe-133 (DE Xe-133). The column labeled "effective" in FGR-12 Table III.1 corresponds to the EDE DCFs used by MHI.

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.

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QUESTION NO. : 15.00.03-4

Provide the assumed initial reactor coolant system fluid volume for the design basis accidents (DBAs) that model leakage to the secondary system.

ANSWER:

For the DBAs that model leakage to the secondary system, MHI assumes an initial reactor coolant system fluid weight of 646,000 lb corresponding to a volume of 77,400 gallons.

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.

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QUESTION NO. : 15.00.03-5

Justify the moisture carryover particulate partition coefficient of 1000 used in the reactor coolant pump (RCP) rotor seizure accident and steam generator tube rupture (SGTR) dose analyses. Provide the amount of moisture carryover assumed in these analyses and the basis for the assumption. Please ensure to describe the calculation used.

ANSWER:

According to Regulatory Guide (RG) 1.183 Appendix E Section 5.5.4, the retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators. The US-APWR steam generator is designed for a maximum moisture carryover of 0.1%. The value of moisture carryover is described in DCD Table 5.4.2-1. Therefore, MHI assumes that particulate partition coefficient is 1000 as the moisture carryover for the RCP rotor seizure accident and SGTR events.

MHI estimates noble gas, iodine and particulate transport separately. Moisture carryover is applied only to the particulate transport. A flow from the secondary coolant equal to 1/1000 of the steam flow rate for moisture carryover is directly discharged to the environment.

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.

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QUESTION NO. : 15.00.03-6

Provide the basis for assuming 0.25% of the core fuel melts as a result of the rod ejection accident.

ANSWER:

In the safety design of a plant, it is desirable to maintain dose consequences as low as possible. This target is highly satisfied in the US-APWR design, in which the core fuel melting does not occur during the rod ejection accident. However, from the standpoint of confirming that dose consequences remain acceptably low even when the core fuel melts, the dose analysis conservatively assumes that 0.25% of the fuel melts.

In the design experience of MHI so far, core fuel melting has not occurred in the rod ejection accident analysis. Consequently, even if the fuel centerline melting should occur in some part of the fuel rods experiencing DNB conditions, its extent would be negligible. Considering low dose consequences under the unlikely condition ending in fuel melting, the rod ejection dose evaluation conservatively assumes that 0.25% of the fuel experiences melt conditions as a reasonably small but significant value for engineering evaluation. This value is, for example, corresponds to the case that the fuel centerline melting consequent on the film boiling occurs in a 10 percent area of the fuel rods in DNB, which is also conservatively assumed as less than 10 percent of the whole number of rods in the core in safety analyses, and the melting area around the fuel centerline reaches to the half of the fuel pellet diameter (a quarter of the volume). The product of 10%, 10%, and 25% gives the assumed value of 0.25%.

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.

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QUESTION NO. : 15.00.03-7

On page 15.4-74, it states that SPR 4.2, Appendix B, was considered in determining the transient fission gas release for the rod ejection accident. Provide the calculation of the transient fission gas release from the fuel rods for the rod ejection accident.

ANSWER:

Transient fission gas release (FGR) from the fuel rod to the gap for a reactivity initiated accident is based on guidance provided in SRP 4.2 Appendix B Section D. The following correlation between the gas release and the maximum fuel enthalpy increase was used to determine the transient fission gas release:

$$\text{Transient FGR} = [(0.2286*\Delta H) - 7.1419]$$

Where:
FGR = Fission gas release, % (must be > 0)
ΔH = Increase in fuel enthalpy, Δcal/g

The transient FGR of 11% is obtained from the above equation when the maximum increase in fuel enthalpy of 79 cal/g is used as ΔH. This value of ΔH corresponds to the difference between the initial fuel enthalpy value of 18.7 cal/g and the peak radial average fuel enthalpy value of 97.5 cal/g described in DCD Section 15.4.8.3.3.

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.

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QUESTION NO. : 15.00.03-8

For the rod ejection accident, 10 percent of the fuel in the core experiences departure from nucleate boiling and is assumed to have cladding failure with a gap fraction of 10% for iodines and noble gases. Assuming cladding failure with a gap fraction of 10% for iodines and noble gases, and given the calculated transient release of 11% of the iodines and noble gases, compare the total release to a total release from the damaged non-melted fuel of 21% of the iodines and noble gases.

ANSWER:

MHI assumes the total gap fraction of damaged non-melted fuel to be 21%. This 21% is the sum of the gap fraction of "10%" and the transient fission gas release of "11%". MHI believes that the DCD analysis is the same as the comparison requested in this RAI.

During the conference call on July 10, 2008 the NRC asked MHI to clarify whether the two rod ejection accident leakage pathways were added into a single dose response. This response confirms that the Chapter 15 dose analysis does assume that the release from containment (including ESF recirculation leakage) and the release from the secondary system occur at the same time and are thus added together. As was discussed during the July 10, 2008 conference call, this is a very conservative assumption since the two events will not occur simultaneously.

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.

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QUESTION NO. : 15.00.03-9

Iodine and aerosol removal in containment by spray and natural deposition is discussed in some detail in Section 6.5.2 of the DCD. For clarity, refer to this information in DCD 15.4.8.5 as it relates to the rod ejection accident.

ANSWER:

DCD Section 6.5.2 provides a detailed discussion of iodine and aerosol removal in containment by spray and natural deposition. DCD Section 15.4.8.5 discusses the manual actuation of the containment spray system, but does not discuss the iodine and aerosol removal aspects of the spray system. MHI will revise the discussion of the radiological consequences of the rod ejection event in DCD Section 15.4.8.5 to include a reference to the more detailed spray removal information found in DCD Section 6.5.2.

Impact on DCD

MHI will revise DCD Section 15.4.8.5 to add the necessary cross-reference to the existing discussion of iodine and aerosol removal by containment spray and natural deposition in DCD Section 6.5.2.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO. : 15.00.03-10

Provide the basis for the assumed break flow rate of 97 gallons per minute (gpm) for the sample line break dose analysis.

ANSWER:

The break flow rate of 97 gpm at a density of 62.4 lb/ft^3 is based on the results of a calculation of the reactor coolant flow from the reactor coolant system (RCS) hot-leg piping through the sample line at rated power operation.

The break flow rate at the break point is evaluated by using the RCS pressure and the resistance of the path through which the reactor coolant flows.

The rate of flow of reactor coolant into the sample line is restricted by a flow restrictor installed at the sample line connection to the RCS. Note that the flow resistance of the sample line piping, valves, etc. is conservatively ignored in the calculation of the resistance of the flow path.

In evaluating the break flow rate from the sample line, the following conditions are assumed:

- RCS temperature (hot leg) = 614.7°F
- RCS pressure (hot leg) = 2248 psig

- Pressure at break point = 0 psig (atmospheric pressure)
- Inner diameter of flow restrictor = 9.5 mm

From the above assumptions, the break flow rate is calculated to be 48,500 lb/hr (22,000 kg/hr). At a density of 62.4 lb/ft^3, the resulting break flow rate is 97 gpm.

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.

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QUESTION NO. : 15.00.03-11

Iodine and aerosol removal in containment by spray and natural deposition are discussed in additional detail in Section 6.5.2 of the DCD. For clarity, refer to this information in DCD Sections 15.6.5.5.1.1, "LOCA Consequence Model" and 15A.1.2, "Airborne Radioactivity Removal Coefficients."

ANSWER:

DCD Section 6.5.2 provides a detailed discussion of iodine and aerosol removal in containment by spray and natural deposition. DCD Sections 15.6.5.5.1.1 "LOCA Consequence Model" and 15A.1.2 "Airborne Radioactivity Removal Coefficients" both mention iodine and aerosol removal in containment by spray and natural deposition, but do not reference the more detailed information which is stated in DCD Section 6.5.2. MHI will add a reference to the more detailed information found in DCD Section 6.5.2 to the appropriate locations in DCD Sections 15.6.5.5.1.1 of the DCD "LOCA Consequence Model" and 15A.1.2 "Airborne Radioactivity Removal Coefficients".

Impact on DCD

MHI will revise DCD Sections 15.6.5.5.1.1 and 15A1.2 to include a cross-reference to the existing discussion of iodine and aerosol removal by containment spray and natural deposition in DCD Section 6.5.2.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO. : 15.00.03-12

Provide the values of the elemental iodine and particulate spray removal coefficients used in the LOCA dose analysis. Provide the time at which the containment spray is no longer operating. Confirm that the spray removal is only credited in the assumed sprayed region of containment.

ANSWER:

The values of the elemental iodine and particulate spray removal coefficients in the containment, the regions in containment where spray removal is credited, and the time at which the containment spray is no longer operating are described below.

<u>Elemental Iodine</u>
As described in DCD Section 6.5.2.3.1, no credit is taken for removal of elemental iodine by the containment spray; only removal by natural deposition is considered in both the sprayed and unsprayed regions of containment. The removal coefficient by natural deposition for elemental iodine, λ_w, is based on SRP 6.5.2. DCD Section 15A.1.2.1 states that the value of λ_w is 0.376 hr^{-1}. Credit for elemental iodine removal is assumed to continue until the decontamination factor (DF) reaches a value of 200 in the containment atmosphere, which occurs 15 hours after initiation of the event. Note that the DF is a time dependent function of the total iodine concentration (elemental and particulate) in containment, therefore, the DF is affected by radiological decay, containment leakage, natural deposition, and containment spray even though spray is not credited for the removal of elemental iodine from the containment atmosphere.

Particulate

Particulate are removed from the containment atmosphere by natural deposition and containment spray. Removal of particulate by the containment spray is considered only in the sprayed region of containment. Particulate removal in the unsprayed regions is only by natural deposition. The containment spray removal coefficient for particulate iodine, λ_P, is also based on SRP 6.5.2. The value of λ_P is 7.32 hr^{-1}. The removal coefficient is assumed to decrease by a factor of 10 after the decontamination factor (DF) reaches 50. When considering removal by radiological decay, containment leakage, natural deposition, and containment spray, a DF of 50 is achieved in 3.23 hours. In the dose analysis, continuous containment spray operation is assumed.

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.

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QUESTION NO. : 15.00.03-13

Inputs and assumptions for the sprayed volume of containment, mixing rate between the sprayed and unsprayed regions of containment, elemental iodine and particulate spray removal coefficients, elemental iodine deposition coefficient, time when elemental removal ends, time when containment spray operation ends, and whether elemental iodine removal and/or aerosol removal are credited in the sprayed and unsprayed regions of containment should be added to DCD Table 15.6.5-4 for completeness.

ANSWER:

DCD Section 6.5.2.2 and DCD Table 6.5-4 provide information regarding the sprayed versus unsprayed volume of containment and mixing between the sprayed and unsprayed regions. DCD Appendix 15A.1.2 provide discussions of spray removal coefficients including details of how the removal mechanisms are modeled (value of the removal coefficient, applicable timing, etc.) in the LOCA dose consequence analysis.

DCD Table 15.6.5-4 summarizes the major input parameters used in the LOCA consequence analysis for the US-APWR. As requested by the NRC, for completeness, MHI will modify DCD Table 15.6.5-4 to also include inputs and assumptions for the sprayed volume of containment, mixing rate between the sprayed and unsprayed regions of containment, elemental iodine and particulate spray removal coefficients, elemental iodine deposition coefficient, time when elemental removal ends, time when containment spray operation ends, and whether elemental iodine removal and/or aerosol removal are credited in the sprayed and unsprayed regions of containment.

Impact on DCD

MHI will add the necessary descriptions of input, assumptions, and values to DCD Table 15.6.5-4.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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DATE OF RAI ISSUE:	**7/24/2008**

QUESTION NO. : 15.00.03-14

In regard to the LOCA dose analysis engineered safety feature (ESF) recirculation system leakage, you state that the ESF systems leakage is taken as two times the sum of the simultaneous leakage from all ESF recirculation systems components above which the Technical Specifications would require such systems inoperable. The Technical Specifications do not appear to include such ESF systems leakage requirements. Please provide the basis for the assumed ESF leakage rate of 17.6 pounds per hour (lb/hr).

ANSWER:

Per the requirements of Regulatory Guide (RG) 1.183, the DCD Chapter 15 dose evaluation assumes a leakage rate that is twice the estimated ESF system leakage rate. The estimated ESF system leakage rate for the US-APWR is 0.018 gpm (4000 cc/hr), which is conservatively estimated based on the following assumptions:

- The portions of the Safety Injection System (SIS) and the Containment Spray System (CSS) that circulate water outside the containment are assumed to leak during their intended operation. Components, fittings and valves in these systems are assumed to leak outside their systems.

- The estimated ESF system leakage rate is the sum of design leakage rates of all devices. The devices assumed to leak are as follows:

    - Pumps (flange and mechanical seal leakage)

- Heat exchangers (flange leakage)

- Instruments (flange leakage)

- Valves (flange, valve stem and seat leakage)

The DCD Chapter 15 dose evaluation assumes a leakage rate of 17.6 lb/hr, which is based on a density of 62.4 lb/ft^3 and corresponds to 0.035 gpm (8000 cc/hr), or twice the estimated ESF system leakage rate as required by RG 1.183.

The assumed ESF leakage rate of 0.018 gpm (4000 cc/h) is not based on ESF leakage requirements in the Technical Specifications but is instead based on the summation of the design leakages for various system components. Therefore, the Section 15.6.5.5.1.1 “Release Paths” text which states that ESF leakage requirements are in the Technical Specifications is deleted.

Impact on DCD

The description in Chapter 15 Section 15.6.5.5.1.1 which states that ESF leakage requirements are in the Technical Specifications is deleted in DCD revision 1.

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

8/22/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO.38 REVISION 1**
SRP SECTION:	**15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
APPLICATION SECTION:	**15**
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QUESTION NO. : 15.00.03-15

In regard to the LOCA dose analysis for ESF recirculation system leakage, you state that all fission products released from the fuel to the containment are assumed to mix in the refueling water storage pit (RWSP) water at the time of release from the core. Confirm that recirculation water mass in DCD Table 15.6.5-4 includes water from the reactor coolant system (RCS) released through the break.

ANSWER:

MHI takes RCS water into account as part of the recirculation water. Therefore, the recirculation water mass in DCD Table 15.6.5-4 includes water from the RCS released through the break.

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

8/22/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

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RAI NO.:	**NO.38 REVISION 1**
SRP SECTION:	**15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
APPLICATION SECTION:	**15**
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QUESTION NO. : 15.00.03-16

The LOCA dose analysis states that control room unfiltered inleakage is assumed to be 120 cubic feet per minute (cfm) due to unexpected ingress/egress. Identify the amount of unfiltered inleakage expected due to leakage through the control room envelope, and the associated TS testing requirements. If the testing requirement for control room envelope inleakage is 120 cfm, then an additional amount should be assumed for ingress/egress through the doors, as discussed in SRP 6.4. The staff considers 10 cfm to be a reasonable estimate for ingress/egress through doors without a vestibule. Further information on control room habitability and control room envelope inleakage may be found in RG 1.196, "Control Room Habitability at Light-water Nuclear Power Reactors," and RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

ANSWER:

The total amount of unfiltered inleakage into the control room is 120 cfm. This total value includes the inleakage through the control room envelope and unexpected inleakage through the control room envelope such as through ingress to and egress from doors without a vestibule. MHI assumes an ingress/egress rate through doors without a vestibule of 10 cfm, which is consistent with NRC staff guidance. The description in DCD Section 15.6.5.5.1.2 associated with MCR unfiltered inleakage will be revised to clarify that the 120 cfm is a total unfiltered inleakage that consists of the described inleakages.

Impact on DCD

MHI will clarify the description in DCD Section 15.6.5.5.1.2 and Table 15.6.5-5 which are related to MCR unfiltered inleakage.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI NO.:	**NO.38 REVISION 1**
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QUESTION NO. : 15.00.03-17

DCD 15.6.5.5.1.3 states that the technical support center (TSC) dose consequences are represented by the control room dose consequences and there are some qualitative statements about the relationship of the TSC model to the control room model. Please provide the details of the TSC dose consequence model, sufficient for the NRC staff to determine independently if the control room dose consequence model is bounding for the TSC. Include information on the TSC intake and unfiltered inleakage receptor locations sufficient for the NRC staff to determine independently that the TSC atmospheric dispersion factors are bounded by the control room atmospheric dispersion factors for all DBAs.

ANSWER:

MHI provides the following basis for their statement in DCD Section 15.6.5.5.1.3.

The technical support center (TSC) dose calculation models are the same as the main control room (MCR) dose calculation models. The relationship among the potential release points, the TSC air intake points, and inleak points are shown in Figure 15.00.03-17-1, which is identical to DCD Figure 15A-1, except for the addition of the TSC intake and inleak receptors.

Major input parameters for the MCR and TSC consequence analysis are compared in Table 15.00.03-17-1. Data in this table allows the verification of the qualitative statement in DCD Section 15.6.5.5.1.3 that the ratio of ventilation flow rate to room volume for the TSC and MCR are the same.

The TSC atmospheric dispersion factors (χ/Q values) are compared to the MCR values for all DBAs in Tables 15.00.03-17-2 through 15.00.03-17-7. Since the access building in which TSC is located does not have any exterior doors, the TSC inleak for all the DBAs, except for the failure of small lines carrying primary coolant outside of the containment, is assumed to be equal to the TSC intake, which is located on the roof of the auxiliary building.

The χ/Q values shown for the RCS sample line are used for the failure of small lines carrying primary coolant outside containment shown in Table 15.00.03-17-5. Based on the layout provided in Figure 15.00.03-17-1, the reactor coolant is assumed to spill in the auxiliary building, be discharged to the atmosphere from the plant vent, and be transferred to the TSC via the TSC HVAC system. However, the TSC dose consequence evaluation uses χ/Q values based on the conservative assumption that radioactive materials leaked in the auxiliary building are transferred directly to the TSC through the access building door. Therefore, the failure of small lines carrying primary coolant outside containment dose consequence assessment uses χ/Q values for TSC intake and inleak at the access building door.

Table 15.00.03-17-1
US-APWR Major Input Parameters Used in the MCR and TSC Consequence Analysis

Parameters	Value	
	MCR	TSC
Control room envelope volume (ft^3)	140,000	46000
Occupancy frequency		
0 to 24 hrs	1.0	1.0
24 hrs to 96 hrs	0.6	0.6
96 hrs to 720 hrs	0.4	0.4
Unfiltered inleakage (cfm)	120	40
Control room HVAC system		
Time delay to switch from normal operation to emergency CRE air filtration mode (s)	180	180
Unfiltered air intake flow during normal operation (cfm)	1800	1000
Filtered air intake flow (cfm)	1200	400
Filtered air recirculation flow (cfm)	2400	1400
Filter efficiency		
• Elemental iodine (%)	95	95
• Organic iodine (%)	95	95
• Particulates (%)	99	99

Table 15.00.03-17-2
MCR and TSC χ/Q for Steam System Piping Failure Analysis
(MCR values are taken from DCD Table 15A-18.)

Accidents		Steam system piping failure							
Sources		Steam line break releases				Main steam relief valve and safety valve releases			
Receptors		MCR		TSC		MCR		TSC	
		Intake	Inleak	Intake	Inleak	Intake	Inleak	Intake	Inleak
		MCR HVAC intake	MCR HVAC intake	TSC HVAC intake	TSC HVAC intake	Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		17	17	67	67	24	24	60	60
Vertical Distance (m)		0	0	0	0	22	22	13	13
χ/Q (s/m^3)	0-8 hr	1.9×10^{-2}	1.9×10^{-2}	1.4×10^{-3}	1.4×10^{-3}	5.3×10^{-3}	5.3×10^{-3}	1.7×10^{-3}	1.7×10^{-3}
	8-24 hr	1.1×10^{-2}	1.1×10^{-2}	8.4×10^{-4}	8.4×10^{-4}	3.1×10^{-3}	3.1×10^{-3}	9.9×10^{-4}	9.9×10^{-4}
	24-96 hr	7.1×10^{-3}	7.1×10^{-3}	5.3×10^{-4}	5.3×10^{-4}	2.0×10^{-3}	2.0×10^{-3}	6.3×10^{-4}	6.3×10^{-4}
	96-720 hr	3.1×10^{-3}	3.1×10^{-3}	2.3×10^{-4}	2.3×10^{-4}	8.7×10^{-4}	8.7×10^{-4}	2.8×10^{-4}	2.8×10^{-4}

Table 15.00.03-17-3
MCR and TSC χ/Q for RCP Rotor Seizure Analysis
(MCR values are taken from DCD Table 15A-19.)

Accidents		RCP rotor seizure accident			
Sources		Main steam relief valve and safety valve releases			
Receptors		**MCR**		**TSC**	
		Intake	Inleak	Intake	Inleak
		Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		24	24	60	60
Vertical Distance (m)		22	22	13	13
χ/Q (s/m^3)	0-8 hr	5.3×10^{-3}	5.3×10^{-3}	1.7×10^{-3}	1.7×10^{-3}
	8-24 hr	3.1×10^{-3}	3.1×10^{-3}	9.9×10^{-4}	9.9×10^{-4}
	24-96 hr	2.0×10^{-3}	2.0×10^{-3}	6.3×10^{-4}	6.3×10^{-4}
	96-720 hr	8.7×10^{-4}	8.7×10^{-4}	2.8×10^{-4}	2.8×10^{-4}

Table 15.00.03-17-4
MCR and TSC χ/Q for Rod Ejection Accident Analysis (Sheet 1 of 2)
(MCR values are taken from DCD Table 15A-20 Sheet 1 of 2.)

Accidents		Rod ejection accident							
Sources		Plant vent				Ground level containment release point			
Receptors		MCR		TSC		MCR		TSC	
		Intake	Inleak	Intake	Inleak	Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Auxiliary building HVAC intake	TSC HVAC intake	TSC HVAC intake	MCRHVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		56	55	55	55	32	27	44	44
Vertical Distance (m)		52	43	43	43	32	33	22	22
χ/Q (s/m^3)	0-8 hr	1.1×10^{-3}	1.4×10^{-3}	1.4×10^{-3}	1.4×10^{-3}	2.2×10^{-3}	2.4×10^{-3}	1.9×10^{-3}	1.9×10^{-3}
	8-24 hr	6.6×10^{-4}	8.0×10^{-4}	8.0×10^{-4}	8.0×10^{-4}	1.3×10^{-3}	1.4×10^{-3}	1.1×10^{-3}	1.1×10^{-3}
	24-96 hr	4.2×10^{-4}	5.1×10^{-4}	5.1×10^{-4}	5.1×10^{-4}	8.3×10^{-4}	9.1×10^{-4}	7.2×10^{-4}	7.2×10^{-4}
	96-720 hr	1.9×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	3.6×10^{-4}	4.0×10^{-4}	3.2×10^{-4}	3.2×10^{-4}

Table 15.00.03-17-4
MCR and TSC χ/Q for Rod Ejection Accident Analysis (Sheet 2 of 2)
(MCR values are taken from DCD Table 15A-20 Sheet 2 of 2.)

Accidents		Rod ejection accident			
Sources		Main steam relief valve and safety valve releases			
Receptors		MCR		TSC	
		Intake	Inleak	Intake	Inleak
		Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		24	24	60	60
Vertical Distance (m)		22	22	13	13
χ/Q (s/m^3)	0-8 hr	5.3×10^{-3}	5.3×10^{-3}	1.7×10^{-3}	1.7×10^{-3}
	8-24 hr	3.1×10^{-3}	3.1×10^{-3}	9.9×10^{-4}	9.9×10^{-4}
	24-96 hr	2.0×10^{-3}	2.0×10^{-3}	6.3×10^{-4}	6.3×10^{-4}
	96-720 hr	8.7×10^{-4}	8.7×10^{-4}	2.8×10^{-4}	2.8×10^{-4}

Table 15.00.03-17-5
MCR and TSC χ/Q for Failure of Small Lines Carrying Primary Coolant Outside Containment and SGTR Analyses
(MCR values are taken from DCD Table 15A-21.)

Accidents		Failure of small lines carrying primary coolant outside containment				SGTR			
Sources		Auxiliary building (RCS sample line)				Main steam relief valve and safety valve releases			
Receptors		**MCR**		**TSC**		**MCR**		**TSC**	
		Intake	Inleak	Intake	Inleak	Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Reactor building door	Access building door	Access building door	Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		52	34	23	23	24	24	60	60
Vertical Distance (m)		7	0	0.9	0.9	22	22	13	13
χ/Q (s/m^3)	0-8 hr	2.2×10^{-3}	4.9×10^{-3}	1.0×10^{-2}	1.0×10^{-2}	5.3×10^{-3}	5.3×10^{-3}	1.7×10^{-3}	1.7×10^{-3}
	8-24 hr	1.3×10^{-3}	2.9×10^{-3}	6.2×10^{-3}	6.2×10^{-3}	3.1×10^{-3}	3.1×10^{-3}	9.9×10^{-4}	9.9×10^{-4}
	24-96 hr	8.4×10^{-4}	1.8×10^{-3}	3.9×10^{-3}	3.9×10^{-3}	2.0×10^{-3}	2.0×10^{-3}	6.3×10^{-4}	6.3×10^{-4}
	96-720 hr	3.7×10^{-4}	8.1×10^{-4}	1.7×10^{-3}	1.7×10^{-3}	8.7×10^{-4}	8.7×10^{-4}	2.8×10^{-4}	2.8×10^{-4}

Table 15.00.03-17-6
MCR and TSC χ/Q for LOCA Analysis
(MCR values are taken from DCD Table 15A-22.)

Accidents		LOCA							
Sources		Plant vent				Ground level containment release point			
Receptors		MCR		TSC		MCR		TSC	
		Intake	Inleak	Intake	Inleak	Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Reactor building door	TSC HVAC intake	TSC HVAC intake	MCR HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		56	37	55	55	32	27	44	44
Vertical Distance (m)		52	60	43	43	32	33	22	22
χ/Q (s/m^3)	0-8 hr	1.1×10^{-3}	1.3×10^{-3}	1.4×10^{-3}	1.4×10^{-3}	2.2×10^{-3}	2.4×10^{-3}	1.9×10^{-3}	1.9×10^{-3}
	8-24 hr	6.6×10^{-4}	7.7×10^{-4}	8.0×10^{-4}	8.0×10^{-4}	1.3×10^{-3}	1.4×10^{-3}	1.1×10^{-3}	1.1×10^{-3}
	24-96 hr	4.2×10^{-4}	4.9×10^{-4}	5.1×10^{-4}	5.1×10^{-4}	8.3×10^{-4}	9.1×10^{-4}	7.2×10^{-4}	7.2×10^{-4}
	96-720 hr	1.9×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-4}	3.6×10^{-4}	4.0×10^{-4}	3.2×10^{-4}	3.2×10^{-4}

Table 15.00.03-17-7
MCR and TSC χ/Q for Fuel Handling Accident Analysis
(MCR values are taken from DCD Table 15A-23.)

Accidents		Fuel handling accident in the containment				Fuel handling accident in the fuel handling area			
Sources		Air lock of containment				Fuel handling area			
Receptors		MCR		TSC		MCR		TSC	
		Intake	Inleak	Intake	Inleak	Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake	MCR HVAC intake	Class 1E electrical room HVAC intake	TSC HVAC intake	TSC HVAC intake
Horizontal Distance (m)		35	29	80	80	82	76	100	100
Vertical Distance (m)		5.2	5.8	0.6	0.6	8.5	8.5	17	17
χ/Q (s/m^3)	0-8 hr	4.7×10^{-3}	6.4×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	9.9×10^{-4}	1.1×10^{-3}	6.7×10^{-4}	6.7×10^{-4}
	8-24 hr	2.8×10^{-3}	3.8×10^{-3}	6.2×10^{-4}	6.2×10^{-4}	5.9×10^{-4}	6.7×10^{-4}	3.9×10^{-4}	3.9×10^{-4}
	24-96 hr	1.8×10^{-3}	2.4×10^{-3}	3.9×10^{-4}	3.9×10^{-4}	3.7×10^{-4}	4.3×10^{-4}	2.5×10^{-4}	2.5×10^{-4}
	96-720 hr	7.7×10^{-4}	1.1×10^{-3}	1.7×10^{-4}	1.7×10^{-4}	1.6×10^{-4}	1.9×10^{-4}	1.1×10^{-4}	1.1×10^{-4}

Sources

1. Containment shell to class 1E electrical HVAC intake (as diffuse area source)
2. Containment shell to main control room HVAC intake (as diffuse area source)
3. Containment shell to auxiliary building HVAC intake (as diffuse area source)
4. Main steam line
5. Main steam relief valve and safety valve
6. Fuel handling area
7. Plant vent
8. Sampling system line
9. Air lock

Receptors

a. Main control room HVAC intake
b. Reactor building door
c. Auxiliary building HVAC intake
d. Class 1E electrical room HVAC intake
e. Technical support center HVAC intake
f. Access building door

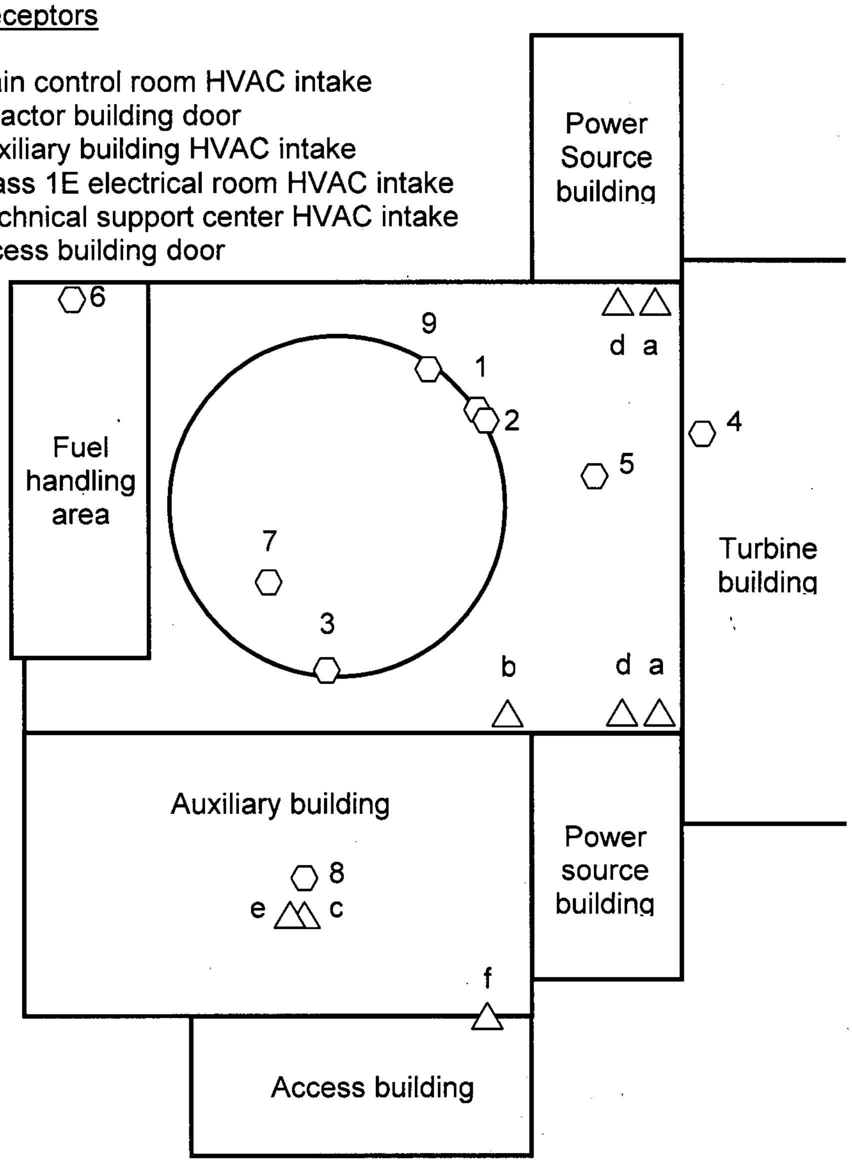

Figure 15.00.03-17-1 Site Plan with Release and Intake Locations

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

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Mitsubishi Heavy Industries

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QUESTION NO. : 15.00.03-18

Provide the control room direct dose calculations and justify the assumptions and inputs. Indicate the receptor location for the operator used in the control room direct dose model.

ANSWER:

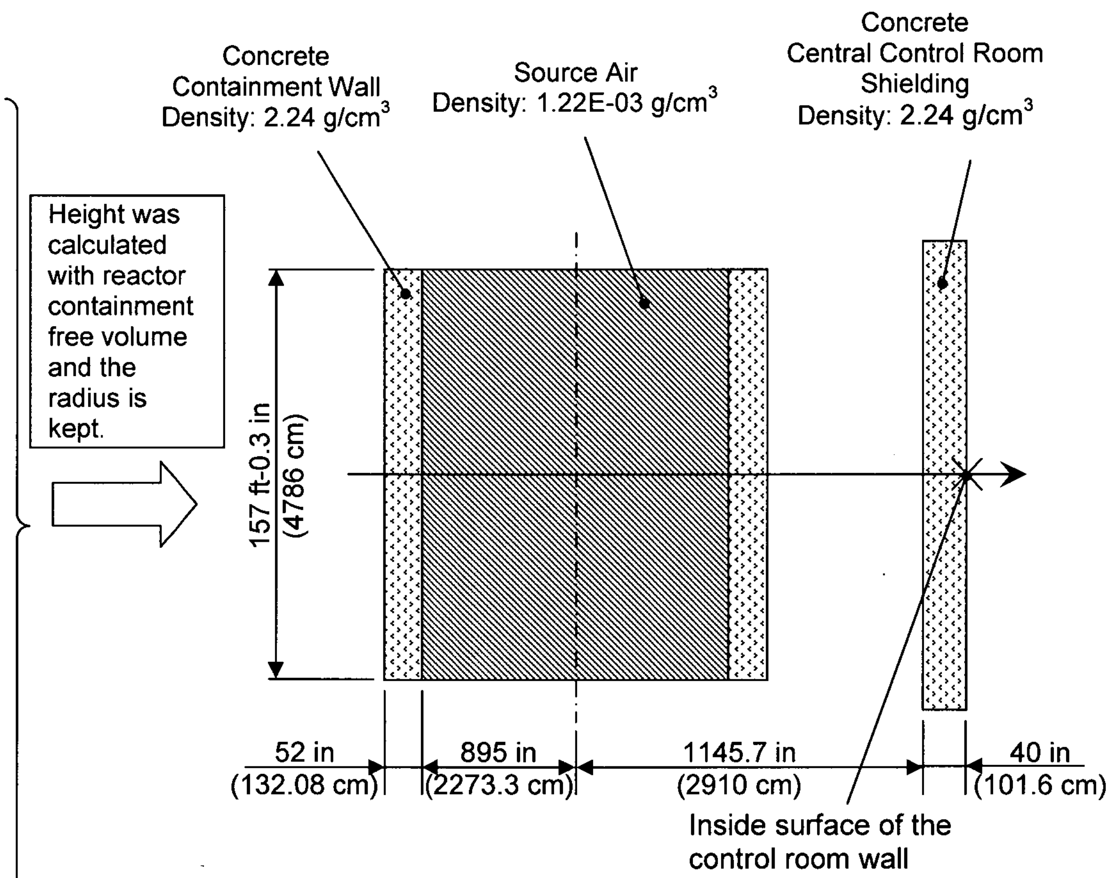
The central control room direct dose calculations are based on the source in the reactor containment, the central control room filter source and the outdoor cloud source. The modeling method for each source and the operator's position (dose evaluation point) are as follows:

(1) Source in the reactor containment
Figure 15.00.03-18-1 proves a schematic description of the containment source model. The source term is modeled as a cylinder whose radius and height are determined from the radius and free volume of the reactor containment. The operator's position is on the inside surface of the control room wall.

(2) Central control room filter source
Figure 15.00.03-18-2 provides a schematic description of the central control room filter source model. The filter is modeled as a point source located on the surface of the filter room floor and the operator's position is conservatively set to 79 in. (2 m) above the central control room floor.

(3) Cloud source
Figure 15.00.03-18-3 provides a schematic description of the cloud source model. The radioactive cloud is modeled as a line source located on the outside surface of the reactor building ceiling. The operator's position is set to 79 in. (2 m) above the central control room floor. When modeling sources located near the shielding surface, as in this case, the radiation travels through the shield (concrete) at an angle resulting in a path through the shield that is longer than the shield thickness. This effect results in a lower dose on the opposite side of the shield unless scattering within the shield is properly accounted for. In order to assure a conservatively calculated dose, the effect from the slant distance through the shield was estimated with another model and multiplied by the result of the dose calculation model shown in Figure 15.00.03-18-3.

{Security-Related Information - Withheld Under 10 CFR 2.390}



Layout (SRI) Model Schematic

Figure 15.00.03-18-1 Modeling method for evaluation of direct dose from reactor containment at the time of the accident

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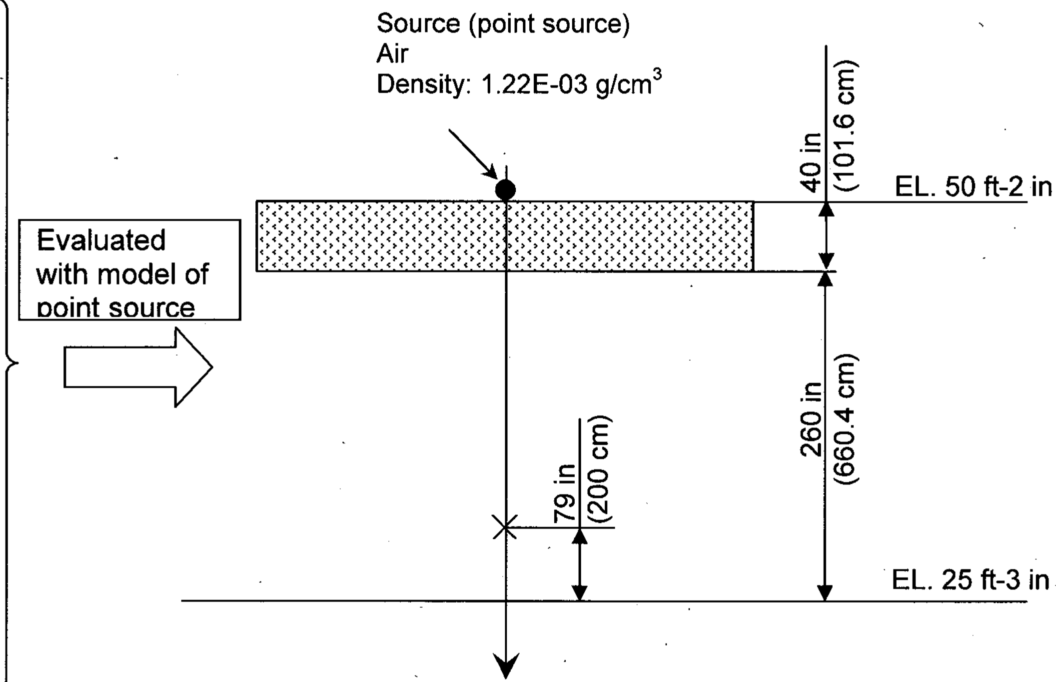
Layout

(SRI)

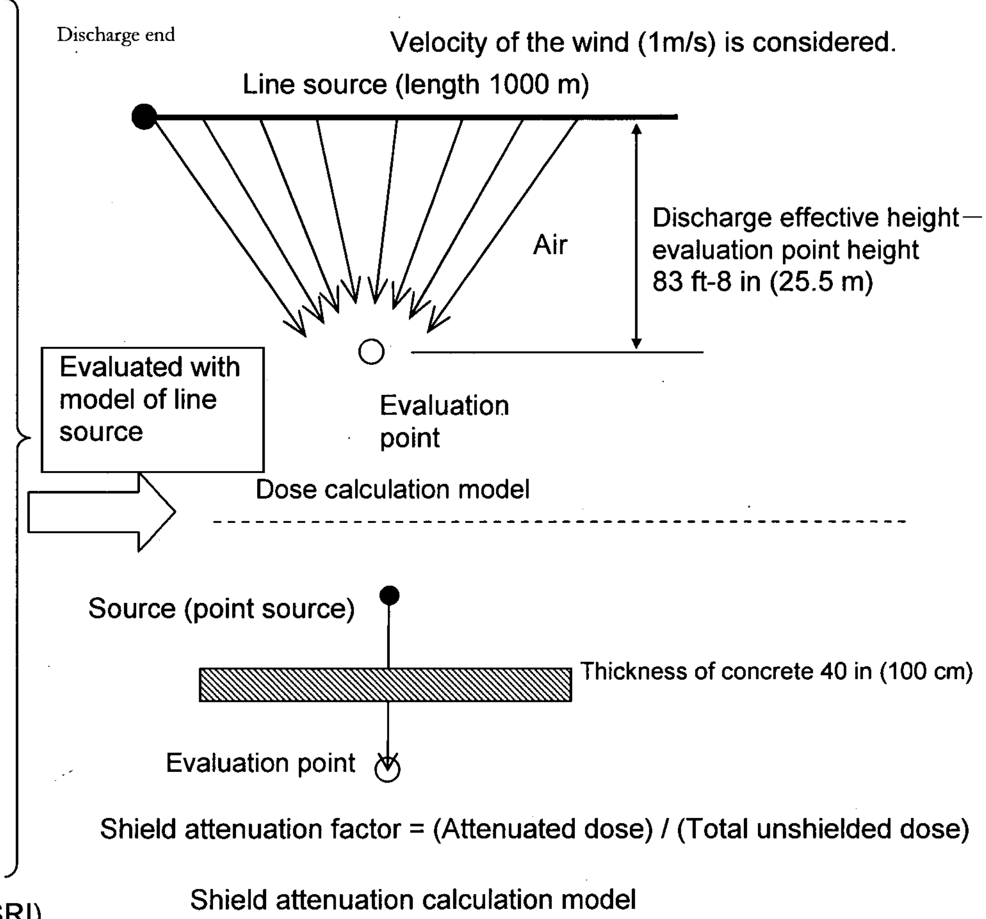


Model Schematic

Figure 15.00.03-18-2 Modeling method for evaluation of direct dose from central control room filter at the time of the accident

{Security-Related Information - Withheld Under 10 CFR 2.390}

(SRI)



Layout

Model Schematic

Figure 15.00.03-18-3 Modeling method for evaluation of direct dose from cloud at the time of the accident

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

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QUESTION NO. : 15.00.03-19

The non-LOCA fission product gap fractions given in RG 1.183 Table 3 were used as input to the fuel handling accident (FHA) analysis. Confirm that the US-APWR fuel meets the burnup and peak rod average linear heat generation rate conditions for applicability given in footnote 11 to RG 1.183 Table 3.

ANSWER:

According to footnote 11 to RG 1.183 Table 3, the gap fractions assume a peak rod burnup of 62 GWD/MTU and that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for rod average burnups exceeding 54 GWD/MTU.

The peak rod burnup limitation of the US-APWR is 62 GWD/MTU as described in DCD Section 4.2.1. Therefore the peak rod burnup of the US-APWR meets the maximum rod average burnup requirements for using Table 3 "Non-LOCA Fraction of Fission Product Inventory in Gap."

For the maximum linear heat generation rate, a typical 24 month equilibrium cycle was used to confirm the requirement, since a peak rod burnup in the first cycle described in the DCD 4.3.2.1 did not exceed 54 GWD/MTU at the end of cycle. The equilibrium cycle used for this analysis is described in MUAP-07016-P Appendix A which was submitted to the NRC (MHI Ref: UAP-HF-08040) on February 27, 2008.

Figure 15.00.03-19-1 shows the relationship between rod burnup and rod average linear heat generation rate calculated with the core design code (ANC) at the beginning, middle, and end of cycle. The rod average linear heat generation rates generally decrease monotonically with the increase in rod burnup because of fissile material depletion. At 54 GWD/MTU, the rod average linear heat generation rates have sufficient margin compared with the 6.3 kW/ft requirement assumed in Table 3 "Non-LOCA Fraction of Fission Product Inventory in Gap." Therefore, the rod average linear heat generation rates of the US-APWR meet the assumptions of RG 1.183 Table 3.

Also, in the fuel handling accident (FHA) analysis, all fuel rods in the fuel assembly analyzed for the FHA are assumed to correspond to the design limit value of $F_{\Delta H}$ with uncertainty and all rods are assumed to have failed. Therefore, a conservative gap activity release is assumed in the dose evaluations.

(PROP)

Figure 15.00.03.19-1 Fuel rod burnup vs rod average linear heat generation rates of 24 month equilibrium core.

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

8/22/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	**NO.38 REVISION 1**
SRP SECTION:	**15.00.03 – Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors**
APPLICATION SECTION:	**15**
DATE OF RAI ISSUE:	**7/24/2008**

QUESTION NO. : 15.00.03-20

DCD 15A.1.1.3 states that the ORIGEN2.2 generation and depletion code was used to calculate the core fission product inventory. Oak Ridge National Laboratory (ORNL) does not support the ORIGEN2 code any longer, but instead recommends use of the ORIGEN-ARP or ORIGEN-S code included in the SCALE code package, which is kept up-to-date. SCALE 5.1 is the latest release and includes libraries for high burnup fuel, up to 72 gigawatt days per metric ton uranium (GWD/MTU). Please justify the use of an older unsupported version of the ORIGEN code.

ANSWER:

This question will be answered later, within 90 days after RAI issue date.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/22/2008

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APPLICATION SECTION:	**15**
DATE OF RAI ISSUE:	**7/24/2008**

QUESTION NO. : 15.00.03-21

DCD 15A.1.1.3 stated that the fuel burnup is 55 GWD/MTU in two cycles. Confirm that the cross-section libraries used in the calculation of the core fission product inventory are applicable to the maximum fuel burnup assumed.

ANSWER:

This question will be answered later, within 90 days after RAI issue date.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/22/2008

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APPLICATION SECTION:	**15**
DATE OF RAI ISSUE:	**7/24/2008**

QUESTION NO. : 15.00.03-22

DCD 15A.1.2.1 stated that the decontamination factor (DF) for the removal of elemental iodine in containment is time dependent, but did not provide the time when removal of elemental iodine is no longer effective. Please provide the time when the DF equals 200 in the associated analysis for the LOCA and the rod ejection accident (REA). For clarity, this information should be added to DCD Table 15.6.5-4 (LOCA) and Table 15.4.8-3 (REA).

ANSWER:

MHI assumes that in the dose analysis for the LOCA and the REA the time when the removal of elemental iodine is no longer effective is when the DF equals 200. This time would correspond to 15 and 13.9 hours after the occurrence of the accident for the LOCA and REA, respectively. MHI will include the respective time when the DF equals 200 in DCD Table 15.6.5-4 and Table15.4.8-3.

Impact on DCD

MHI will add the description of the time when the DF equals 200 for the LOCA and REA to DCD Tables 15.6.5-4 (LOCA) and 15.4.8-3 (REA), respectively.

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

8/22/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

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APPLICATION SECTION:	**15**
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QUESTION NO. : 15.00.03-23

Particulate iodine removal in containment was modeled by using the 10th percentile Powers natural deposition correlation in RADTRAD. Considering that the Powers natural deposition correlation was developed using operating pressurized water reactor (PWR) and boiling water reactor (BWR) information on containment geometry and power, explain why the Powers natural deposition correlation is applicable to the US-APWR containment.

ANSWER:

NUREG/CR-6189 provides the following correlation between volume and power for dry PWR containments.

$$V(m^3) = 16700(+/-5500) + 16.16(+/-1.94)*P(MWt)$$

Hence, the containment volume should be between the high and low values for the derived natural deposition correlations to be valid. The high and low values of the containment volume are shown to be:

$$V_{min}(m^3) = 11200+ 14.22*P(MWt)$$

And

$$V_{max}(m^3) = 22200+ 18.1*P(MWt)$$

At the US-APWR nominal power level of 4451 MWt, the containment volume should be between 74,500 m^3 and 102,761 m^3. The design containment free volume for the US-APWR is 79,350 m^3, which is within the upper and lower bound. Therefore, the Powers natural deposition correlation is applicable to the US-APWR.

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.