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TOKYO, JAPAN

November 25, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09473

Subject: **MHI's Responses to US-APWR DCD RAI 427-2909 Rev.1, 428-2910 Rev.1 and 429-3178 Rev.1**

Reference:
1) "Request for Additional Information No.427-2909 Revision 1, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2" dates July 30,2009.
2) "Request for Additional Information No.428-2910 Revision 1, SRP Section: 12.03-12.04 – Radiation Protection Design Features, Application Section: 12.3" dates July 30,2009.
3) "Request for Additional Information No.429-3178 Revision 1, SRP Section: 12.03-12.04 – Radiation Protection Design Features, Application Section: 12.3-12.04" dates July 30,2009.

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

Enclosed are the responses to RAIs contained within Reference 1 through 3.

As indicated in the enclosed materials, documents (Enclosure 2) 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. Non-proprietary versions of the documents are also being submitted in this package (Enclosure 3). In the non-proprietary versions, the proprietary information, bracketed in the proprietary versions, is 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, 4 and 5) 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 this submittal. His contact information is provided below.

D081
NRO

Sincerely,

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.427-2909 Revision 1 (proprietary and SRI included version)

3. Responses to Request for Additional Information No.427-2909 Revision 1 (non-proprietary and SRI excluded version)

4. Responses to Request for Additional Information No.428-2910 Revision 1 (non-proprietary and SRI excluded version)

5. Responses to Request for Additional Information No.429-3178 Revision 1 (non-proprietary and SRI excluded version)

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

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-09473

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and 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 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 "Responses to Request for Additional Information No.427-2909 Rev.0", and have determined that the document and attachment data contain proprietary information that should be withheld from public disclosure.

3. The information in the document and data identified as proprietary by MHI 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 are that the equations described in the response to RAI item 12.02-17 involves MHI's know-how, and to make these input data from a lot of design parameters requires knowledge and know-how.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's Application for certification of its US-APWR Standard Plant Design.

6. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design of new fuel systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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

Executed on this 25th day of November, 2009.

Y. Ogata

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

[Security-Related Information — Withheld Under 10 CFR 2.390]

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 1 of 34)
Site

Enclosure 3

UAP-HF-09473, Rev.0

Responses to Request for Additional Information No.427-2909
Revision 1

September 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 427-2909 REVISION 1**
SRP SECTION:	**12.02 – Radiation Sources**
APPLICATION SECTION:	**12.2**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.02-17

The US-APWR FSAR Revision 1 Tier 2 Sections 2 "Radiation Sources" describes design features that provide the bases for the shielding design calculations and the airborne radioactivity concentrations that provides the bases for the ventilation system design.

Supplemental Question (SQ-1) derive from RAI 141-1735

RAI 141-1735 Question 12.02-6 requested additional information about the source term associated with In Core Instrument System (ICIS) and the Spent Fuel Pool and Refueling Pool water activity concentrations.

In the response the applicant indicated that:
1. MHI provided tables 12.2-70 and 12.2-71 for equipment material specifications and revised sections 12.2.1.2.3 and 12.2.1.2.5 to describe the method for determining activation of ICIS cables. This response appears to be inadequate and inconsistent with other information provided by MHI because:

- Table 12.2-70 "Parameters and Assumptions for Calculating Spent Fuel Source Strength" notes that fuel enrichment is 4.55%, while USAPWR FSAR Tier 2 Table 4.2-1 "Fuel Assembly Design Specifications" list Fuel Enrichment as 5%.

Provide a fuel enrichment source term for spent fuel that is consistent with other sections of the DCD or provide the justification for the use of a different fuel enrichment in this section.

- As noted in US-APWR FSAT Tier 2 Section 7.7.1.5.2, the ICIS detectors are miniature fission detectors. Table 12.2-71 "Parameters and assumptions for Calculating Irradiated Incore Detector, Drive Cable and Flux Thimble Source Strength" does not show the source term and dose rates associated with these fission detectors.

Describe the dose rates, and their bases, for the fission chambers of the In Core Instrument System neutron detectors, or provide the specific alternative approaches used and the associated justification.

<u>References</u>

1. "Request for Additional Information No. 141-1735 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02-6" dated January 9, 2009 CHPB Branch

(ADAMS Accession No. ML090410551)

ANSWER:

In the response to RAI 141-1735, submitted by MHI letter UAP-HF-09047 dated February 6, 2009, MHI indicated that the Chapter 12 of the DCD would be revised to add Table 12.2-70. The purpose of adding DCD Table 12.2-70 was to provide the parameters and assumptions used for calculating the spent fuel source strength. DCD Table 12.2-70 identifies the fuel enrichment used for the calculation as 4.55 wt%. As indicated in this RAI, other sections of the DCD use other values for the fuel enrichment. DCD Table 4.2-1 and other places in Chapter 4 describe a maximum fuel enrichment of 5 wt%. The value of 5 wt% is meant to be the maximum possible enrichment for the US-APWR. In general, the fuel enrichment is much lower. DCD Figure 4.3-2 shows the initial core fuel loading pattern which contains assemblies with fuel enrichments in the range of 2.05 wt% to 4.15 wt%. As described in DCD Subsection 12.2.1.2.3, the spent fuel pit is conservatively assumed to contain 257 assemblies from unloading the full core. Based on the fuel enrichments shown in DCD Figure 4.3-2, the average enrichment on the 257 assemblies would be less than 4 wt%. Therefore, a bounding value of 4.55 wt% is conservative for this calculation.

The source strengths of the ICIS detectors were calculated using the ORIGEN code. The parameters used to determine the source strengths are presented in Table 1. The results from the code provide the Table 2 source strengths for the fission chamber of the In Core Instrument System neutron detectors. The dose rates of the fission chamber, detector (Fe-Ni cover), and drive cable were calculated using the MicroShield code where the fission chamber and detector were modeled as point sources and the drive cable was modeled as a line source. The dose rates resulting from these sources are presented in Figure 1, indicating that only the drive cable represents a significant source beyond several inches distance (in air) from the ICIS detectors.

Table 1 Parameters and Assumptions for Calculating Fission Chamber Source Strength

Parameter/Assumption	Value
^{235}U weight (g)	0.00158
Total Neutron Flux (n/cm^2/s)	3.8E+14*
Irradiation period (hours)	20

*Total flux is the neutron flux values for the four ranges of neutron energy (Table 12.2-71)

Table 2 Fission Chamber Source Strength

Gamma Ray Energy (MeV)	Fission Chamber Source Strength (MeV/sec)
0.01	1.1E+09
0.025	6.4E+08
0.0375	7.2E+08
0.0575	1.3E+09
0.085	1.4E+09
0.125	1.8E+09
0.225	9.0E+09
0.375	9.6E+09
0.575	2.1E+10
0.85	3.6E+10
1.25	4.0E+10
1.75	1.5E+10
2.25	1.5E+10
2.75	7.5E+09
3.5	6.0E+09
5.0	4.7E+09
7.0	5.3E+07
9.5	1.4E+04

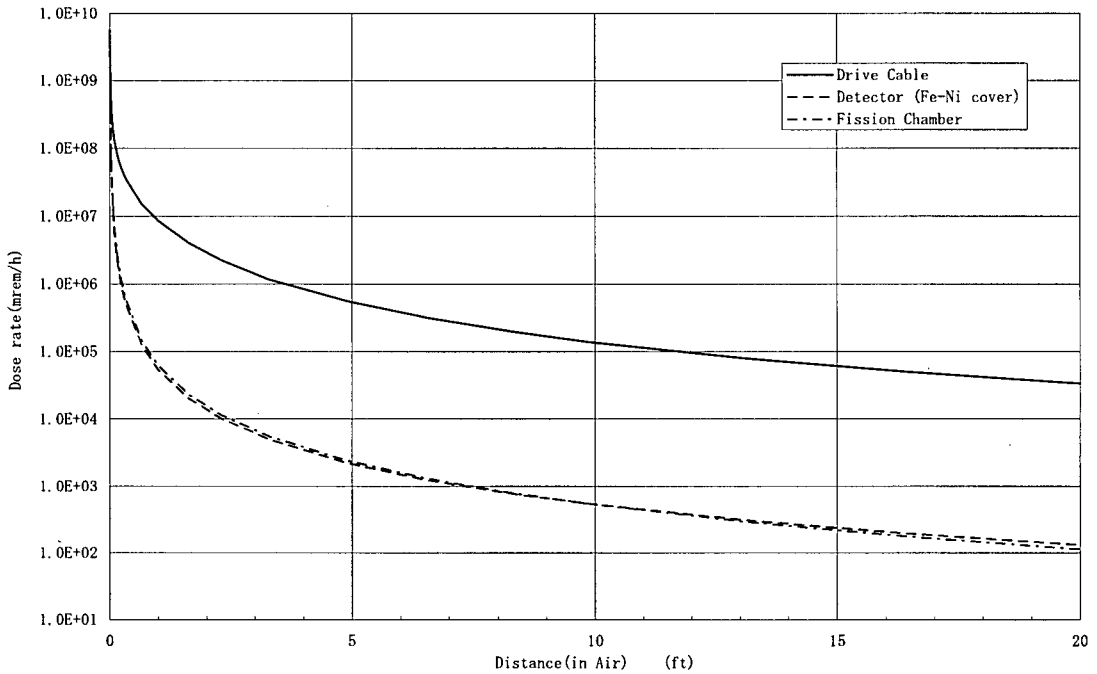

Figure 1 Dose Rate Distribution of Fission Chamber, Detector and Drive Cable

Impact on DCD

There is no impact on the COLA.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	NO. 427-2909 REVISION 1
SRP SECTION:	12.02 – Radiation Sources
APPLICATION SECTION:	12.2
DATE OF RAI ISSUE:	7/30/2009

QUESTION NO.: 12.02-18

The US-APWR FSAR Revision 1 Tier 2 Sections 2 "Radiation Sources" describes design features that provide the bases for the shielding design calculations and the airborne radioactivity concentrations that provides the bases for the ventilation system design.

Supplemental Question (SQ-2) derive from RAI 142-1733

RAI 142-1733 Questions 12.02-8 and 12.02-9 requested additional information about the bases, models and assumptions supporting the dose rates indicated around the Boric Acid Evaporator (BAE) package, the associated pumps (Concentrates and Distillate), and the airborne activity concentrations associated with sludge removal from the BAE during maintenance, under the deign basis conditions of 1% defective fuel cladding.

In summary, the Applicant Response indicated that:
1. Based on other plant experience, actual doses from this equipment will be insignificant.

This response is inconsistent with information provided by the Applicant in the US-APWR FSAR Tier 2 Section 12.2.1.1 "Sources for Full-Power Operation" which states that "The design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power". The use of
cladding defects for determining the shielding requirements is consistent with the Acceptance Criteria contained in SRP Section 12.2.

Describe the source term used as the basis for determining the dose rates, shielding requirements, airborne activity concentration and ventilation system design parameters, and the resultant ORE dose reduction design features for normal operation and abnormal operating occurence (AOO), and provide the associated revisions to the FSAR, or provide the specific alternative approaches used and the associated justification.

2. The Applicant stated that filters and demineralizers in the feed stream will remove suspended activity, so no radioactivity deposits will remain in the boron recycle system.

The assumption that no activity would be present in the process fluid streams after the demineralizers and filters is inconsistent with the applicant response to RAI 168-1739 Question 12.02-14, which provided DF values for filters and demineralizers, and Average concentration factors for fluid that has passed

through the BAE. Additionally, from NRC Staff OE with nuclear and non-nuclear evaporator systems, sludge and scale accumulate in the evaporator internals. EPRI report TR- 1011728 "Radioactive Liquid Processing Guidelines" provides
corroboration of the staff operating experience (OE). Since this material remains in the evaporator package following draining for maintenance, it is a significant source of internal and external exposure to maintenance personnel accessing the area around the BAE package. In some plants, due to these deposits, the BAE were posted as High Radiation Areas. At those plants, BAE input source terms were higher than realistically expected for current generation plants, but they were still significantly less than the US-APWR design basis source term, while filtration and purification of the BAE input stream was as good as or better than, currently proposed in the US-APWR.

Calculations performed by the staff, using the criteria described in NUREG-1400 "Air Sampling in the Workplace" Section 1 "Air Sampling Based on Potential Intakes and Concentrations" indicated the potential for maintenance related airborne activity concentrations exceeding an ALI, for distributed source activity 1/100th of that needed to cause a High Radiation Area due to activity deposited in the BAE.

Describe the source terms, their bases, and the resultant external exposure and airborne activity concentrations, associated with maintenance of internal and external components of the BAE package and the Boric Acid Evaporator Concentrates pump, under operation and maintenance associated with design bases cladding defects.

3. In the response to RAI 168-1739 Question 12.02-14, the Applicant provided Average concentration factors for fluid that has passed through the BAE. which appear to present non-conservative estimates of the amount of activity in the BAE and concentrates pump. The Applicant assumes a concentration factor of 35, based on increasing boron from 200 ppm to 7000 ppm. However from Question 168-1739 12.02-14 Table A, the volume of water processed by the BAE significantly exceeds the capacity of a Chemical and Volume Control System holdup tank (CVCS HUT), so at some point in time, the BAE will be operating with a feed stream boric acid concentration closer to 10 ppm. However, the activity concentration will not decrease like the boron concentration. In order to reach the target boric acid concentration of 7000 ppm, the concentration ratio will be closer to 700, not 35. In addition, insufficient information has been provided to allow the staff to determine if the BAE package operates with a constant concentration factor output, or a constant boron concentration output. This operational method is significant because the concentration factor in the BAE package may approach 1000 in some cases. In either case, higher fission product activity will be present resulting in higher dose rates and contamination levels, in the BAE package and concentrates pump.

Describe the BAE package external exposure and airborne activity concentrations, and the associated bases, for internal and external maintenance of the BAE following processing a boric acid solution representative if End Of Core Life (~10 ppm) with design bases cladding defects.

4. The Applicant noted that the parameters of the Boric Acid Evaporator and Vent Condenser will be added to Table 12.2-69, however, it is not clear that Table 12.2-69 refers to the BAE Vent Condenser activity

Please clarify the title of Table 12.2-69.

References

1. "Request for Additional Information No. 142-1733 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02-8" dated January 9, 2009 CHPB Branch (ADAMS Accession No. ML090410551)
2. "Request for Additional Information No. 142-1733 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02-9" dated January 9, 2009 CHPB Branch (ADAMS Accession No. ML090410551)

ANSWER:

Answer to 1
The source terms used as the basis for shielding design and dose rate calculations in the vicinity of the boric acid evaporator (BAE) are derived from the source terms for the Holdup Tank presented in DCD Chapter 12, Table 12.2-26. The Holdup Tank source term values were calculated based on the design basis conditions of 1% defective fuel cladding. The source term values for the Holdup Tank were adjusted to develop the activities in the evaporator by applying the decontamination factor associated with the BAE Feed Demineralizer. A decontamination factor of 10 was conservatively applied based on the recommendations of NUREG-0017. The source terms for the BAE are tabulated in Table 12.2-66 "Chemical and Volume Control System Radiation Sources B.A. Evaporator Activity" which will be added to the DCD as defined in the response to DCD RAI NO. 142-1733 Revision 1, Question No. 12.02-8.

The BAE performs the function within the CVCS system of separating the reactor coolant feed stream into a stream of boric acid water of approximately 7000 ppm and a stream of distilled water. The BAE is therefore operated until the concentration of 7000 ppm boric acid is reached in the concentrate which is confirmed through intermittent sampling. Based on an average operation time of 34.5 hours (for a starting concentration of 200 ppm) and a concentrate volume of 6.7 m^3 (240 ft^3), the dose rate of less than 0.25 mrem/hour is achieved in the walkway adjacent to the vault housing the BAE with the current wall thickness of 2.5 feet.

The ORE dose reduction design features include locating the BAE in a shielded vault with external walls of a thickness of 2.5 feet. The internal walls and floor are coated with epoxy to ease cleanup of potential contamination. A steel door, of sufficient thickness to provide the shielding equivalent of the 2.5 feet thick concrete walls, is provided for the vault to limit access and provide additional shielding. Airborne activity in the vicinity of the BAE is very low due to its leak-tight design. In addition, the off-gas from the unit is vented directly to the Gaseous Waste Management System (GWMS) for treatment and release.

Answer to 2
During normal operation, including AOOs, the BAE concentrate is sent to the boric acid tank and the condensate is sent to the primary makeup water tank for recycle in plant systems. If maintenance is required on the BAE, the complete evaporator package, including the Boric Acid Evaporator Concentrates pump will be washed before inspection and maintenance are performed. The wash water from this operation will be drained to the Auxiliary Building sump which will transfer it to the Floor Drain System for processing. Small equipment parts and instruments can also be brought to the hot machine shop for further decontamination prior to maintenance or part replacement.

The shielding design for the BAE vault area is based on the source terms presented in Table 12.2-66 "Chemical and Volume Control System Radiation Sources B.A. Evaporator Activity" which will be added to the DCD as defined in the response to DCD RAI NO. 142-1733 Revision 1, Question No. 12.02-8. These source terms are calculated based on the design basis failed fuel fraction (1% defective fuel cladding) and very low decontamination factors (a DF of 10 is used). During actual plant operation, the failed fuel fraction is extremely low and the current treatment technologies, including filters and Demineralizers, can achieve much higher decontamination factors. The combination of these design factors produces radiation exposure and airborne radiation concentrations, during normal operation and maintenance activities, that are insignificant, which is supported by actual operating data.

Answer to 3
Per the description of the boron recycle subsystem in DCD Subsection 9.3.4.2.5, the BAE package operates with a constant boron concentration output to the boric acid tank. While one batch of concentrated boric acid water is being processed, the boric acid evaporator continuously receives feed water and discharges distilled water to the primary makeup storage tank. The concentration of boric acid is gradually increased until it reaches 7000 ppm B. Intermittent sampling is performed to determine if the

concentrated boric acid water is ready for release to the boric acid tank or if further processing is needed. If the concentrate does not meet the specification after the concentration procedure, it is returned to the holdup tank for reprocessing. Therefore, it is not necessary to recalculate BAE package source terms and dose rates for processing a boric acid solution of approximately 10 ppm with design basis cladding defects.

Answer to 4
The Boric Acid Evaporator Vent Condenser Source Strength will be presented in Table 12.2-69 as defined in the response to DCD RAI NO. 142-1733 Revision 1, Question No. 12.02-8.
The title of this table will be revised accordingly.

Impact on DCD

The title of DCD Table 12.2-69, as presented in the response to DCD RAI NO. 142-1733 Revision 1, Question No. 12.02-8, will be changed (see attached).

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Table 12.2-69 Chemical and Volume Control System Radiation Sources B.A. Evaporator Vent Condenser Source Strength

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	4.3E+04
0.03	7.6E+05
0.04	3.7E+04
0.06	1.1E-01
0.08	1.6E+06
0.1	4.6E+01
0.15	2.7E+04
0.2	2.9E+05
0.3	8.3E+03
0.4	1.5E+04
0.5	5.9E+04
0.6	2.7E+04
0.8	3.2E+04
1.0	1.5E+04
1.5	6.4E+04
2.0	2.9E+05
3.0	2.3E+04

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 427-2909 REVISION 1**
SRP SECTION:	**12.02 – Radiation Sources**
APPLICATION SECTION:	**12.2**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.02-19

The US-APWR FSAR Revision 1 Tier 2 Sections 2 "Radiation Sources" describes design features that provide the bases for the shielding design calculations and the airborne radioactivity concentrations that provides the bases for the ventilation system design.

Supplemental Question SQ-3 derive from 143-1737

RAI 143-1737 Question 12.02-10 asked the applicant to provide the methods and assumptions for determining the airborne concentrations noted in Table 12.2-61 "Airborne Radioactivity Concentrations" and similar tables.

In summary, the Applicant Response indicated that:

1. They will revise Section 12.2 to provide a complete list of parameters and assumptions used for determining airborne activity concentrations.
2. Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactive Concentrations" will be revised,
3. Table 12.2-72 "Reactor Cavity And SFP Water Specific Activity In Refueling /Shutdown (Except Tritium)" will be added,

This response appears to be inadequate and inconsistent with other information provided by Applicant because:

1. The information provided in the revised Table 12.2-60 does not contain sufficient information to allow the staff to confirm the airborne activity concentrations provided in Table 12.2-61. For instance:
   a. The use of equation 12.2-1 to calculate airborne concentrations requires the use of unavailable room volumes and room flow rates so the λ removal factor can be determine.
   b. It is not clear that the information provided in Tables 12.2-61 (Sheets 1-6), "Airborne Radioactivity Concentrations" represents the maximum airborne concentrations in areas workers could expect to occupy. Insufficient information is available to the staff, to allow determination of the areas of maximum airborne activity concentration, due to room turn over rate, leakage rate and source strength.
2. Some of the information provided for Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactivity Concentrations is inaccurate or incomplete:
   a. Regarding Table 12.2-60 (Containment) (Sheet 1 of 3):

I) The fraction of RAM to free volume during refueling for tritium is listed as 0.1. As there is no concentration mechanism in place for tritium evaporating from the pool surface, this value should be 1.0.

II) The reference to Table 12.2-62 should actually be to Table 12.2-72.

III) Purge Flow Duration for Low Volume Purge is listed as "Continuous", but Chapter 16 Technical Specifications section 3.6.3.2 indicates that the containment isolation valves are normally shut.

b. Regarding Table 12.2-60 (Fuel Handling Area) (Sheet 2 of 3)":

I) The reference to Table 12.2-62 should actually be to Table 12.2-72.

II) The fraction of RAM to free volume during refueling for tritium is listed as 0.1. As there is no concentration mechanism in place for tritium evaporating from the pool surface, this value should be 1.0.

III) Flow Rate is listed as 24,000 cfm, however, the staff is unable to ascertain the basis for this value.

c. Regarding Table 12.2-60 (RB and AB) (Sheet 3 of 3)":

I) A number of "Radiation Zones" are listed (III to VI). Since there are other zones depicted in Figures 12.1-1 sheets 1-34 "Radiation Zones for Normal Operation/Shutdown", and the RB and AB have piping and equipment areas containing these missing zones, it is not clear how the listed zones were derived and where they are defined.

II) The assumed leakage rate provided is only for Refueling. The information provided is insufficient to allow determination of the assumed leakage rates for Operation at NOP/NOT.

III) IF the depicted leakage rates are also for NOP/NOT conditions, THEN the minimum assessed leakage rate for any area enclosing equipment containing ESF fluids, should not be less than the leakage rate assumed in Chapter 15 LOCA analysis.

IV) For those areas that enclose equipment that does not contain ESF fluids, the assumed leakage rates are not conservative for dose calculations, with respect to the Technical Specifications unidentified leakage rate of 1 gpm (12,000 lpd) or the identified leakage rate of 10 gpm (120,000 lpd)fluids. See the reply to Question 143-1737 12.02-11 MHI response.

3. The values in Table 12.2-72 do not appear to be consistent with the values in Table 12.2-52 "RHR System Activity – 4 Hours After Shutdown", even when the Table 12.2-52 values are corrected for clean up to the EPRI Primary Water Chemistry Guidelines, and then diluted with distilled water (not RWSAT) to fill the Cavity.

Requested Information

1. Identify the limiting areas of airborne activity in each radiological section of the plant. Provide all of the parameters needed by Equation 12.2-1, the bases for selection of those values, and the resultant airborne concentrations for the limiting areas of the plant or provide the specific alternative approaches used and the associated justification.
2. Revise Table 12.2-60 Sheets 1 to 3, to address the specific concerns noted, or provide the specific alternative approaches used and the associated justification.
3. Provide the methods models and assumptions used to derive the values presented in Table 12.2-72.

References

1. "Request for Additional Information No. 143-1737 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02- 10" dated January 9, 2009 CHPB Branch (ADAMS Accession No. ML090410551)

ANSWER:

Answer to 1a

For those regions characterized by a constant leak rate of the radioactive source at constant source strength and a constant exhaust rate of the region, the peak or equilibrium airborne concentration of the ith radioisotope in the region is calculated using the following equation (DCD Equation 12.2-1):

$$C_i(t) = \frac{(LR)_i A_i (PF)_i [1 - exp(-\lambda_{Ti} t)]}{V \lambda_{Ti}} \qquad \text{Eq. 1}$$

where:

$(LR)_i$	=	Leak or evaporation rate of the ith radioisotope in the applicable region (g/s)
A_i	=	Radioactivity concentration of the ith leaking or evaporating radioisotope (μCi/g)
$(PF)_i$	=	Partition factor or the fraction of the leaking radioactivity that is airborne for the ith radioisotope
λ_{Ti}	=	Total removal rate constant for the ith radioisotope from the applicable region (1/s)
	=	$\lambda_{di} + \lambda_e$, the removal rate constants in 1/s due to radioactive decay for the ith radioisotope and the exhaust from the applicable region, respectively
λ_e	=	the exhaust removal rate in 1/s defined as Q/V
λ_{di}	=	the radioactive decay rate in 1/s for the ith radioisotope
V	=	Free volume of the region in which the leak occurs (cm^3)
Q	=	Ventilation flow rate (cm^3/s)
t	=	Time elapsed from the start of the leak and the time at which the concentration is evaluated (s)
$C_i(t)$	=	Airborne concentration of the ith radioisotope at time t in the applicable region ($\mu Ci/cm^3$)

From the above equation, it is evident that the peak or equilibrium concentration, C_i, of the ith radioisotope assumed in the applicable region will be given by the following expression:

$$C_i = \frac{(LR)_i A_i (PF)_i}{V \lambda_{Ti}} \qquad \text{Eq. 2}$$

As a conservative assumption, radioactive decay of the ith radioisotope is ignored. Using this assumption, Eq. 2 can be simplified as represented below:

$$C_i = \frac{(LR)_i A_i (PF)_i}{V\left(\frac{Q}{V}\right)} = \frac{(LR)_i A_i (PF)_i}{Q} \qquad \text{Eq. 3}$$

All of the parameters needed to solve Eq. 3 are described in DCD Table 12.2-60, as revised by the previous MHI responses to RAIs 12.02-4 and 12.02-10 and summarized in MUAP-09003 Revision 1, "US-APWR DCD Tracking Report".

Answer to 1b

As described in the Answer to 1a above, DCD Subsection 12.2.2.5 describes the specific airborne radioactivity model used to calculate the peak isotopic concentrations presented in DCD Table 12.2-61. MHI will revise DCD Table 12.2-60 as indicated in MUAP-09003 Revision 1 to reflect the changes previously submitted to the NRC regarding source strength assumptions. MHI believes that these previous revisions along with Eq. 3 above are sufficient to allow the NRC to determine the areas of maximum airborne activity concentration.

Answer to 2a I)

Before refueling, the reactor cavity is filled with water resulting in more than 10 times dilution of the reactor coolant. The basis for the assumed dilution factor is discussed in Attachment-1 of the answer to item 3 in this RAI response.

This dilution effect is included in the RAM to free volume value provided in DCD Table 12.2-60, but the

description is easily misunderstood. Therefore, MHI will revise DCD Table 12.2-60.

Answer to 2a II)
The noted typographical error will be corrected in DCD Revision 2 as previously indicated in MUAP-09003 Revision 1.

Answer to 2a III)
DCD Chapter 16, Technical Specification Surveillance Requirement (SR) 3.6.3.2 states:

> Verify each 8 inch low volume purge valve is closed, except when the 8 inch containment low volume purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

This SR allows exception for the valve position for ALARA or air quality considerations for personnel entry. "Continuous" in DCD Table 12.2-60 means the airborne concentration in the containment reaches the equilibrium concentration. Therefore, this assumption is consistent with the Technical Specification SR.

Answer to 2b I)
The noted typographical correction will be corrected in DCD Revision 2 as previously indicated in MUAP-09003 Revision 1.

Answer to 2b II)
See the answer to 2a I) above.

Answer to 2b III)
The flow rate listed as 24,000 cfm is the exhaust flow rate from the fuel handling area that MHI previously described in the response to Question 09.04.02-2 of RAI 328-2436.

Answer to 2c I)
As depicted in Figure 1 below, this calculation models all cubicles assigned to the same radiation zone category as a single hypothetical cubicle. Airborne activity is calculated using Eq. 3 (see the answer to 1a above) and does not depend on cubicle volume. Therefore, this treatment considers all of the cubicles in the radiological controlled area(RCA).
Radiation Zone I is not RCA. Radiation Zone II is the RCA , but the equipment leaked radioactive liquid is not installed in this Zone. The equipment leaked radioactive liquid (e.g. pumps) is installed in zone III and higher zone. Therefore, MHI will revise DCD Table 12.2-60 and Tabl2 12.2-61.

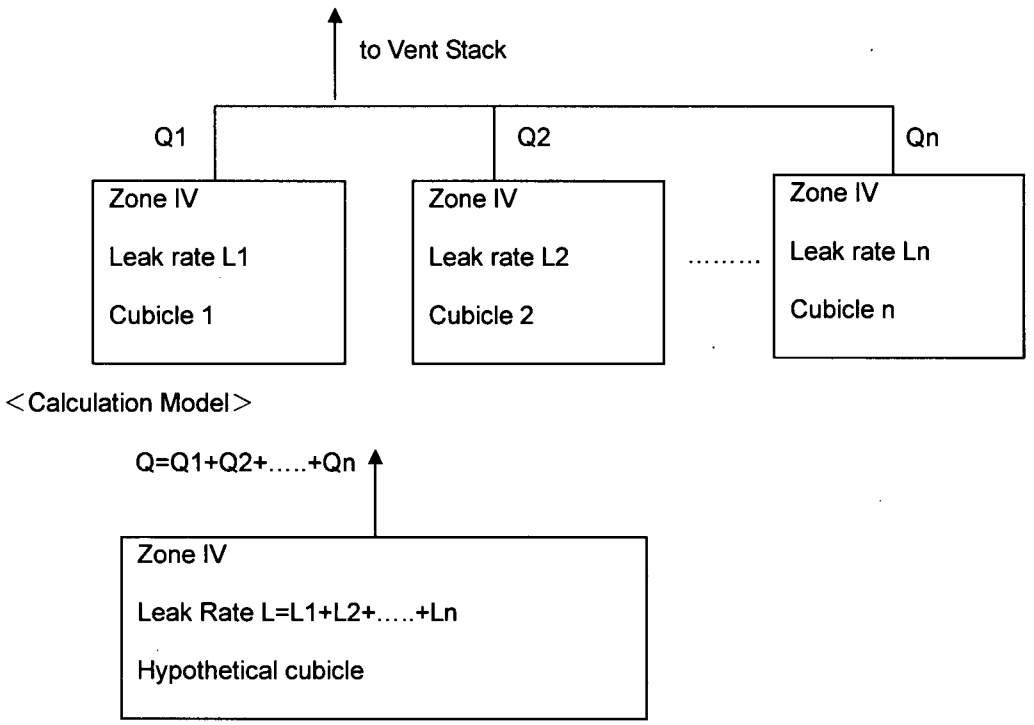

Figure 1 Treatment of the Cubicle in the Airborne Activity Calculation (example)

Answer to 2c II)
This is simply a typographical omission. The assumption described in Table 12.2-60 (Sheet 3 of 3) is for normal operation and refueling. MHI will revise DCD Table 12.2-60 to correct this omission.

Answer to 2c III)
Table 12.2-60 (Sheet 3 of 3) provides the reactor coolant leak rate. As described in the answer to 2c II) above, this leakage rate applies to NOT/NOP conditions. This leakage rate is used to calculate the airborne radioactive concentrations for the purpose of calculating Occupational Radiation Exposure of personnel. The ESF leakage rate assumed in the Chapter 15 LOCA analysis, provided in DCD Table 15.6.5-4, corresponds to 422.4 lb/d. The Chapter 15 ESF leakage rate is chosen conservatively in order to ensure that the worst case radiation exposure to the public remains below the limits. For this reason, the leakage rate assumed in Chapter 15 for accident conditions is greater than that assumed in Table 12.2-60 for NOT/NOP.

Answer to 2c IV)
DCD Table 12.2-60 (Sheet 3 of 3) applies to the airborne activity concentration calculation for the Reactor Building (RB) and Auxiliary Building (AB). Technical Specification 3.4.16 does not apply to the RB and AB calculations because this TS is for RCS operational leakage.

Answer to 3
The radioactive concentration in the reactor cavity water which is the source of airborne radioactivity is based on 24 hours of radioactive decay and purification of the CVCS. This 24 hour period corresponds to the radioactive decay time before fuel assembly handling can be initiated. Therefore, the values in DCD Table 12.2-72 are different from the values in DCD Table 12.2-52 "RHR System Activity – 4 Hours After Shutdown".

The methods, models, and assumptions used to derive the values presented in DCD Table 12.2-72 are

shown in Attachment-1 of the response to this RAI.

Impact on DCD

DCD Table 12.2-60 will be revised as previously indicated in MUAP-09003 Revision 1. In addition to those revisions, DCD Table 12.2-60 will also be revised as described in the answer to 2aI), 2bII), 2cI) and 2c II) above and shown in the attached markup. DCD Table 12.2-61 will also be revised as described in the answer to 2cI) above and shown in the attached markup (refer to Attachment-2).

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

This attachment provides the calculation model used to determine the values provided in DCD Table 12.2-72.

1. Reactor Cavity Water Activity

The calculation of the reactor cavity water activity uses the following equations:

The values of the parameters utilized in these equations are provided in Table 1 at the end of this attachment.

2. Primary Coolant Activity after Shutdown

The calculation of the primary coolant activity the after shutdown uses the following assumptions:

(PROP)

(PROP)

(PROP)

(PROP)

Table 1 Parameters Used to Calculate Reactor Cavity Water Activity

Table 2 Parameters Used to Calculate Primary Coolant Activity After Shutdown

Parameter	Unit	Value
Decontamination factor of CVCS mixed bed demineralizer	--	Br and I: 100 Cs and Rb: 2 Others: 50
Decontamination factor of CVCS cation bed demineralizer	--	Br and I: 1 Cs and Rb: 10 Others: 10
	--	
Reactor coolant mass	lb	646000
Primary coolant activity during normal operation	μCi/g	DCD Table 11.1-2

(PROP)

(PROP)

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Containment) (Sheet 1 of 3)

Parameter/ Assumption	Value
Reactor coolant leakage rate in normal operation	100 lb/d
Reactor coolant evaporation rate in refueling	1020 lb/h
Fraction of radioactive material to free volume	(in normal operation) 1.0(for noble gas) 0.45(others) (in refueling/shutdown) 1.0(for noble gas&tritium) 0.1(iodine~~&tritium~~) 0.001(others)
Fuel defect	1%
Reactor coolant specific activity in normal operation (except tritium)	Table 11.1-2
Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)	Table 12.2-72
Reactor coolant tritium specific activity	3.5μCi/g
Reactor cavity and SFP water tritium specific activity in refueling /shutdown	0.35~~3.5~~μCi/g
Low volume purge flow rate	(in normal operation) 2000 cfm
High volume purge flow rate	(in refueling) 30000 cfm
Purge flow duration	continuous

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Fuel Handling Area) (Sheet 2 of 3)

Parameter/ Assumption	Value
Reactor coolant evaporation rate in refueling	750 lb/h
Fraction of radioactive material to free volume	(in refueling/shutdown) 1.0(for noble gas&tritium) 0.1(iodine~~&tritium~~) 0.001(others)
Fuel defect	1%
Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)	Table 12.2-72
Reactor cavity and SFP water tritium specific activity in refueling /shutdown	0.35~~3.5~~μCi/g
Flow rate	24000 cfm
Flow duration	continuous

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building) (Sheet 3 of 3)

Parameter/ Assumption	Value
Reactor coolant leak rate in normal operation/refueling (Note)	100 lb/d (for Radiation Zone V ~~to VI~~ or higher) 50 lb/d (for Radiation Zone IV) 2 lb/d (for Radiation Zone III)
Fraction of radioactive material to free volume	(in normal operation) 1.0(for noble gas) 0.1(iodine&tritium) 0.001(others)
Fuel defect	1%
Reactor coolant specific activity in normal operation (except tritium)	Table 11.1-2
Reactor coolant tritium specific activity	3.5μCi/g
Flow rate	1500 cfm(for Radiation Zone V ~~to VI~~ or higher) 14000 cfm (for Radiation Zone IV) 76000 cfm (for Radiation Zone III)
Flow duration	continuous

(Note) Reactor coolant leak rates were derived from the leakage flow rates of the valves under consideration. Each Radiation Zone has a different number of valves handling radioactive fluids. Radiation Zones V and higher have many component cubicles and valve galleries. These zones have many radioactive valves. Zone IV has relatively high radiation level corridors, but has fewer radioactive valves than Zone V. Zone III has low radiation level corridors and access areas, and has fewer radioactive valves than Zone IV. As a result, the leak rate in Zone V or higher is high, while in Zones IV and III, the leak rates is low.

Table 12.2-61 Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone V ~~to VI~~ or higher) (Sheet 4 of 6)

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)
Kr-83m	3.4E-07	1E-02	Ru-106	7.9E-14	5E-09
Kr-85m	1.3E-06	2E-05	Ag-110m	7.3E-16	4E-08
Kr-85	6.9E-05	1E-04	Te-125m	3.2E-13	2E-07
Kr-87	8.7E-07	5E-06	Te-127m	1.3E-12	1E-07
Kr-88	2.5E-06	2E-06	Te-127	6.8E-12	7E-06
Xe-131m	3.1E-06	4E-04	Sb-129	2.2E-14	4E-06
Xe-133m	3.1E-06	1E-04	Te-129m	4.4E-12	1E-07
Xe-133	2.3E-04	1E-04	Te-129	5.5E-12	3E-05
Xe-135m	5.7E-07	9E-06	Sb-131	8.9E-15	1E-05
Xe-135	7.7E-06	1E-05	Te-131m	1.2E-11	2E-07
Xe-138	5.0E-07	4E-06	Te-131	6.3E-12	2E-06
I-130	4.7E-09	3E-07	Te-132	1.3E-10	9E-08
I-131	1.2E-07	2E-08	Cs-132	6.2E-13	2E-06
I-132	6.4E-08	3E-06	Te-133m	1.2E-11	2E-06
I-133	2.1E-07	1E-07	Te-133	6.0E-12	9E-06
I-134	4.4E-08	2E-05	Cs-134	5.7E-10	4E-08
I-135	1.3E-07	7E-07	Te-134	2.2E-11	1E-05
Br-82	6.4E-12	2E-06	Cs-135m	6.7E-12	8E-05
Br-83	5.8E-11	3E-05	Cs-135	1.5E-15	5E-07
Br-84	3.1E-11	2E-05	Cs-136	1.5E-10	3E-07
Rb-86	5.6E-12	3E-07	Cs-137	3.2E-10	6E-08
Rb-87	-	6E-07	Cs-138	7.4E-10	2E-05
Rb-88	3.2E-09	3E-05	Ba-140	1.7E-12	6E-07
Rb-89	7.3E-11	6E-05	La-140	4.5E-13	5E-07
Sr-89	1.4E-12	6E-08	La-141	1.2E-13	4E-06
Sr-90	9.2E-14	2E-09	Ce-141	2.6E-13	2E-07
Y-90	2.1E-14	3E-07	Ce-143	2.2E-13	7E-07
Sr-91	9.5E-13	1E-06	Pr-143	2.4E-13	3E-07
Y-91m	4.9E-13	7E-05	Ce-144	2.0E-13	6E-09
Y-91	2.2E-13	5E-08	Pr-144	2.0E-13	5E-05
Sr-92	5.3E-13	3E-06	Pm-147	2.2E-14	5E-08
Y-92	4.1E-13	3E-06	Sm-147	-	2E-11
Y-93	1.8E-13	1E-06	Eu-154	2.1E-15	8E-09
Zr-93	-	3E-09	Na-24	2.9E-11	2E-06
Zr-95	2.7E-13	5E-08	Cr-51	2.8E-12	8E-06
Nb-95m	2.0E-15	9E-07	Mn-54	1.9E-12	3E-07
Nb-95	2.7E-13	5E-07	Mn-56	9.6E-11	6E-06
Mo-99	3.3E-10	6E-07	Fe-55	1.9E-12	8E-07
Tc-99m	1.3E-10	6E-05	Fe-59	3.3E-13	1E-07
Tc-99	-	3E-07	Co-58	4.5E-12	3E-07
Mo-101	1.5E-11	6E-05	Co-60	6.6E-13	1E-08
Tc-101	1.4E-11	1E-04	Zn-65	5.4E-13	1E-07
Ru-103	2.3E-13	3E-07	H-3	2.6E-07	2E-05
Rh-103m	2.2E-13	5E-04			

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 427-2909 REVISION 1**
SRP SECTION:	**12.02 – Radiation Sources**
APPLICATION SECTION:	**12.2**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.02-20

The US-APWR FSAR Revision 1 Tier 2 Sections 2 "Radiation Sources" describes design features that provide the bases for the shielding design calculations and the airborne radioactivity concentrations that provides the bases for the ventilation system design.

Supplemental Question SQ-4 derive from RAI 143-1737

RAI 143-1737 Question 12.02-11 asked the applicant to justify the use of a 100 lpd (~0.008 gpm) RCS leakage rate for calculating airborne activity levels for occupational workers, considering that the Technical Specifications (TS) Unidentified Leakage rate (TSULR) limit is 1 gpm and the TS Identified Leakage rate (TSILR) limit is 10 gpm.

The Summary of the Applicant Response was:
1) The Safety Analysis for the TS allowable leakage rates did not consider airborne activity concentration limitations.
2) That the leakage rates in TS 3.4.13 also prevent the plant from exceeding the accident analyses radiation release assumptions.
3) The basis for the leak rate inside containment is taken from ANSI/ANS- 55.6 "Liquid Waste Processing for Light Water Reactor Plants",
4) They state that they believe the use of the lower leakage rate values is reasonable for the purpose of radiation protection.

It is true that the TS RCS leakage rate limit is based on Leak Before Break crack propagation criteria, and airborne activity concentrations are not a part of the TS leak rate safety analysis. However, SRP section 12.2 notes that review of airborne activity concentrations for dose assessment are to be based on normal operation, AOO and accident conditions. SRP Section 12.3-4 notes that the ventilation system is sized to maintain airborne activity concentrations less than 1 DAC in areas not normally occupied, where maintenance or inspections need to be performed.

However, the response to this RAI appears to be inadequate and inconsistent with other information provided by MHI because:

1) Insufficient information is presented to the staff in the US-APWR FSAR Tier 2, Chapters 11 and 15 to confirm the MHI assertion that the TS 3.4.13 also prevents the plant from exceeding the accident analysis radiation release assumptions. The leakage rates value used in Table 11.2-9 and Table

15.4.8-3 are much less that the leakage rate values listed in TS 3.4.13. Provide sufficient information to allow the staff to confirm the statement by MHI that TS 3.4.13 also prevents the plant from exceeding the accident analyses radiation release assumptions.

2) The leakage rate used by the Applicant are not representative of documented industry experience and would present non-conservative estimates of Occupational Radiation Exposure for personnel entering affected areas.

The value from ANSI/ANS-55.6 is used for determining the average liquid processing capacity required over the course of cycle. This is an appropriate value to use when the waste storage tanks act as a smoothing function for variable leak rates. However, the airborne activity concentrations within closed volumes, such as the Containment Building, are tightly linked with the instantaneous leakage rates. This tight linkage is the basis of the Regulatory Guide 1.45 position statement requirement for a Radiation Monitor based RCS leakage detection method. Based on NRC Staff OE it is not uncommon to have RCS Leakage rates inside containment well in excess of the MHI assumed leakage rate. This experience is corroborated by information presented in NUREG/CR-6861 "Barrier Integrity Research Program", Figure 17 "Distribution of Leak Rates" which documents more than 100 leaks above 0.015 gpm (180 lpd) and more than 50 leaks greater than 1 gpm (12005 lpd). Also, WCAP-16465-NP Revision 0 "Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors" notes action levels that are well in excess of 100 lpd, including:

a) The 0.15 gpm two consecutive day values noted in Section 5.2.1 "Absolute Unidentified Leak Rate Action Levels",
b) The 0.3 gpm one daily RCS leakage rate Even at a WCAP-16465-NP Tier Three Action Level leak, the corrective action may only involve identification of the source of leakage, and initiating a plan to correct the leak. This document further notes that RCS operational LEAKAGE shall be limited to 1 gpm unidentified leakage, and 10 gpm identified leakage. WCAP-16465 is not referenced in Chapters 5 or Chapter 16, so the lower values of leakage noted in this document may not be applicable. Therefore, the RCS Leakage Rate of 100 lpd (0.008 gpm) for normal operation is not conservative for dose calculations when using the Technical Specifications unidentified leakage rate of 1 gpm (12,000 lpd) or the identified leakage rate of 10 gpm (120,000 lpd).

Based on the guidance provided in SRP Section 12.2, the operating experience described by NUREG/CR-6861, and the program description described by WCAP-16465-NP, provide leakage rate assumptions consistent with Technical Specification leakage rate limits, or justify the use of 100 lpd RCS leakage rate for conservatively determining ORE airborne activity concentrations inside containment.

References

1. "Request for Additional Information No. 143-1737 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02- 11" dated January 9, 2009 CHPB Branch (ADAMS Accession No. ML090410551)

ANSWER:

The safety significance of RCS leakage varies widely depending on its source, rate and duration. Each plant is required to have a plant-specific RCS leakage monitoring program capable of monitoring and identifying changes in unidentified RCS leakage at levels that are well below the operational leakage limits established by the Technical Specifications. DCD Subsection 5.2.5 describes the US-APWR reactor coolant pressure boundary (RCPB) leak monitoring system. In accordance with position 9 of Regulatory Guide (RG) 1.45, the limiting condition for identified and unidentified reactor coolant leakages are identified in Technical Specification (TS) 3.4.13. These limits are determined with regards to RCS leakage for the purpose of protecting the reactor coolant boundary from degradation and assuring adequate core cooling.

Process leakage results in the release of radioactive material, primarily noble gases and volatile fission products, to plant areas, and subsequently to the environment. A portion of this leakage evaporates and contributes to the airborne / gaseous source term. DCD Subsection 12.2.2 documents the models, parameters, and sources used to evaluate the expected routine radioactive airborne concentrations of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected. The sources identified in DCD Subsection 12.2 are used in the design of the (1) ventilation systems, (2) personnel protective measures, and (3) for dose assessment. In accordance with SRP 12.2 Section I.2 the calculated airborne radioactivity concentrations used are based on "expected normal operation, anticipated operational occurrences, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel." The Technical Specifications in Chapter 16 of the DCD describe various conditions within which the plant must be operated; these limits are not considered expected conditions during normal operation, AOOs, or accidents. Therefore, MHI did not use the TS limits to determine the bounding radioactive airborne concentrations provided in DCD Table 12.2-61.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. Also newer, commercially available systems such as air particulate detectors can detect leakage less than 0.1 gpm. Once a leakage trend is detected, plant procedures prompt actions to identify and stop the leakage prior to reaching the TS limits, thus limiting the effect that this level of leakage would have on the overall expected routine occupational exposures. WCAP-16465-NP, which was identified by the reviewer, provides one example of administrative action levels for handling detected leakage rates significantly below (approximately $1/10^{th}$) the TS limits. Additionally, the reviewer cites Figure 17 of NUREG/CR-6861 and WCAP-16465-NP as an example of why the MHI assumed leakage rate of 100 lb/day is not conservative for dose calculations. As the reviewer pointed out, Figure 17 of NUREG/CR-6861 provides a distribution of the magnitude of identified leak events spanning from less than 0.001 kg/s (~0.016 gpm) to greater than 10 kg/s (~160 gpm) over a 20 year period. However, the report concludes that many of the leaks reported are very small (< 0.01 gpm), are detected visually rather than by instrumentation, and are reported as drips, weeping, seepage, or "very small" boric acid deposits, etc. It is these very small visually detected leaks that are likely to go undetected long enough to be considered part of the routine expected operating conditions. Therefore, it is MHI's position that if the leakage rate is such that it can be readily identified and procedurally handled it is not the appropriate leakage rate to use for the Chapter 12.2 analysis of airborne radioactive concentrations inside containment.

As described in the previous RAI response to Question 12.02-11 that was submitted to the NRC by MHI letter UAP-HF-09047 dated February 6, 2009 (ML090410551), the assumed operational RCS leakage rate inside containment used for the occupational radiation exposures determined in DCD Section 12.2 is 100 lb/day, which is approximately 0.01 gpm. This leakage rate corresponds closely to the 10 gpd assumed RCS leakage inside containment considered for plant effluent releases through the radioactive waste management systems or the plant ventilation system (see DCD Table 11.2-2). Per the previous response to RAI 12.02-11, the value of 10 gpd was taken from ANSI/ANS-55.6 "Liquid Radioactive Waste Processing for Light Water Reactors"; however, it also corresponds to the primary coolant leakage from miscellaneous sources in Table 2-26 of NUREG-0017, Rev. 1 (the PWR-GALE code). MHI assumes that this entire amount of liquid leakage contributes to the airborne radioactive source term, when realistically, only a small fraction of this leakage becomes airborne.

Based on the above discussion, MHI maintains that the assumed 100 lb/day is an acceptable airborne exposure source term for the purpose of radiation protection.

In this supplementary RAI, the NRC asked MHI to provide sufficient information to confirm the previous MHI statement that TS 3.4.13 prevents the plant from exceeding the accident analysis radiation release assumptions. The following discussion addresses this request.

Except for primary-to-secondary leakage within the steam generators, the Chapter 15 safety analyses do

not address operational RCS leakage. However, operational leakage is related to the safety analyses for LOCA since the amount of leakage can affect the probability of such an event. Additionally, primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a Steam Line Break (SLB) accident and to a lesser extent, other accidents or transients involving secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). TS 3.4.13 limits primary to secondary leakage through any one steam generator (SG) to 150 gallons per day. The DCD Subsection 15.6.3.5 analysis for SGTR assumes a total primary-to-secondary leakage rate of 600 gpd. The SLB is more limiting with respect to site radiation releases. The dose evaluation for the SLB accident in DCD Subsection 15.1.5.5 assumes a primary-to-secondary leakage rate of 150 gpd in the faulted loop and 450 gpd total for the three intact loops. In both cases, the assumed leakage is in accordance with the TS limit and dose consequences resulting from the event were determined to be well within the limits defined in 10 CFR 100.

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 427-2909 REVISION 1**
SRP SECTION:	**12.02 – Radiation Sources**
APPLICATION SECTION:	**12.2**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.02-21

The US-APWR FSAR Revision 1 Tier 2 Sections 2 "Radiation Sources" describes design features that provide the bases for the shielding design calculations and the airborne radioactivity concentrations that provides the bases for the ventilation system design.

Supplemental Question (SQ-5) derive from RAI 144-1738

RAI 144-1738 Question 12.02-12, asked the applicant to provide information regarding the thickness of the concrete walls surrounding the outside storage tanks for Primary Water Storage and Refueling Water Auxiliary Storage.

In summary, the MHI Response indicated that:

1. The design is being changed to remove the concrete shield walls, and that:
   a. Procedures will be adopted to control the activity in the tank to limit the dose rate to 0.25 mrem/h at 2 meters.
   b. Barriers will be installed at 2 meters from the tank to limit personnel access.
2. Section 12.2.1.1.10 and Tables 12.2-50, 51 and 12.3-1 will be revised to include activity concentrations for the Refueling Water Storage Auxiliary Tank, and the Primary Makeup Water Tank.

This response appears to be inadequate and inconsistent with other information provided by MHI because:

1. The response notes that following the removal of concrete shield walls procedures will be used to control the activity in the tank to limit the dose rate to 0.25 mrem/h at 2 meters. However, there is no corresponding COL action item in section 12.2.3 "Combined License Information".
2. The response indicated that barriers would be installed 2 meters from the tank to limit access to the area around the tanks. Contrary to this statement Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown Site (Sheet 1 of 34)", has not been modified to show the barriers, nor has a COL action item been added to section 12.2.3 "Combined License Information" to address the installation of these barriers.
3. Figure 12.3-1 was not changed to indicate that the areas around the tanks and inside of the barriers would be a higher radiation zone.
4. While the response indicated that the concrete walls would be removed, it did not describe the design features of the areas around the tanks that will support the requirements of 10 CFR 20.1406, with respect to minimizing contamination of the environment.

Requested Information

1. Please identify the COL action item in Section 12.2.3 that addresses the procedural guidance for limiting the amount of activity that may be added to these tanks.
2. Please modify figure 12.3-1 to show the barriers referenced in the response, or add a COL action item in Section 12.2.3 to provide this information.
3. Please modify figure 12.3-1 to show that the dose rates inside of these barriers is a higher zone.
4. Please describe the design features of the areas around the tanks will support the requirements of 10 CFR 20.1406, with respect to minimizing contamination of the environment.

References

1. "Request for Additional Information No. 144-1738 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02-12" dated January 9, 2009 CHPB Branch (ADAMS Accession No. ML090410551)

ANSWER:

In the response to RAI No.144-1738, MHI provided a description on the strength of the radiation source and shield design concept for the external tanks (the Refueling Water Storage Auxiliary Tank and the Primary Makeup Water Storage Tanks). Explanation was also given therein that no concrete wall will be installed around the outside tanks, that fences will be provided 2 meters away from the tank surface, and that the concentration of radioactivity will be controlled to ensure a dose rate of 0.25 mrem/h or less on the outside.

However, in the course of design since then, a design modification has been made such that these external tanks that were originally intended to be installed outdoors will now be placed inside a tank house with a roof and walls because leakage detecting equipment to be installed to meet the requirement of RG 4.21 may malfunction due to rainfall when placed outdoors.

As a minimum distance of 2 m is maintained between the tank house outer wall and the tank surface, the dose rate mentioned in RAI 144-1738 will be satisfied, and at the same time, inadvertent access by unauthorized persons to the tanks may physically be prevented. The dose rate in the area inside the tank surface will exceed 0.25 mrem/h but still less than 1 mrem/h, so the tank inside will be set as zone II, which will also be reflected in Fig.12.3-1 in DCD Chapter 12, with modifications incorporated accordingly.

The design features for these tanks that support the requirements of 10 CFR 20.1406, with respect to minimizing contamination of the environment are same as indicated in the response for the RAI No. 91-1496 Revision 1, Question 12.03-12.04-2, Table 12.03-1C and Table 12.03-1D.
Same as the CVCS equipments or LWMS equipments, measures to minimize leaks and spills, installation of leak detection system, contamination control and so on, are to be incorporated in the tank house.

The operational limitations regarding the radioactivity control of the contained water in these tanks, which manage the dose rate at 2 meters from the surface of the tank less than 0.25 mrem/h, are to be prescribed in some kind of program such as the Radiation Protection Program.

Impact on DCD

The US-APWR DCD Subsection 12.2.1.1.10 and Figure 12.3-1 will be revised as described in the Attachment.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

12.2.1.1.10 Miscellaneous Sources

The principal sources of activity outside the buildings but inside the tank house include the following:

- The refueling water storage auxiliary tank
- The primary makeup water tank

The content of the water tanks is processed by the SFP purification system, or the boron recycle system until the activity in the fluids is sufficiently low to result in dose rates less than 0.25 mrem/h at 2 meters from the surface of the tank.

Radionuclide inventories of the refueling water storage auxiliary tank and primary makeup water tank are presented in Tables 12.2-50 and 12.2-51. There are no other significant amounts of radioactive fluids permanently stored outside the buildings.

Spent fuels are stored in the SFP. When the fuel is to be moved away from the SFP, it is placed in a spent fuel shipping cask for transport.

Storage space is allocated in the radwaste processing facility for storage of spent filter cartridges and packaged spent resins.

Radioactive wastes stored inside the plant structures are shielded so that areas outside the structures meet Radiation Zone I criteria. Additional storage space for radwaste is to be provided in the detailed design by the COL Applicant. If it becomes necessary to temporarily store radioactive wastes/materials outside the plant structures, radiation protection measures are to be taken by the radiation protection staff to ensure compliance with 10 CFR 20 (Reference 12.2-1), 40 CFR 190 (Reference 12.2-6) and to be consistent with the recommendations of RG 8.8 (Reference 12.2-2).

The SWMS facilities process and store dry active waste. If it becomes necessary to install additional radwaste facilities for dry active waste, it is to be provided by the COL Applicant. Radiation shielding is to be provided such that the dose rates comply with the requirements of 10 CFR 20 (Reference 12.2-1) and 40 CFR 190 (Reference 12.2-6). Interior concrete shielding is provided to limit exposure to personnel during waste processing. The ALARA methodology of RGs 8.8 (Reference 12.2-2) and 8.10 (Reference 12.2-3) has been used in the design of this facility.

Any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography, are to be provided by the COL Applicant.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent fuel, the residual heat removal system (RHRS), and the incore instrumentation system (ICIS). Individual components may

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 427-2909 REVISION 1**
SRP SECTION:	**12.02 – Radiation Sources**
APPLICATION SECTION:	**12.2**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.02-22

The US-APWR FSAR Revision 1 Tier 2 Sections 2 "Radiation Sources" describes design features that provide the bases for the shielding design calculations and the airborne radioactivity concentrations that provides the bases for the ventilation system design.

Supplemental Question (SQ-6) derive from RAI 168-1739

RAI 168-1739 Question 12.02-14, asked the applicant to provide information regarding the assumptions used to calculate resin activity.

This response appears to be inadequate or incomplete for the following reason:

1. Table 12.2-1 listed effective density values to be used for shielding calculations, for a number of components (e.g. Steam Generator, Regenerative and Excess Let Down Heat Exchangers) that do not match stated w/% in the materials column.

Requested Information

1. Please correct the density values provided in Table 12.2-1 or provide the specific alternative approaches used and the associated justification.

References

1. "Request for Additional Information No. 168-1739 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02 - 14" dated February 3, 2009 CHPB Branch (ADAMS Accession No. ML090650632)

ANSWER:

Modeling restrictions imposed by the calculation code used for the shield calculations require the use of an artificial mixed density for certain geometries than cannot be directly modeled within the code. Heat exchangers are an example of a geometry that requires this modeling situation.
In the case of a heat exchanger as shown in Figure 1, a radiation source enters from the plenum side, is cooled by component cooling water while passing inside the U-shaped tube in the shell side, and leaves

from the plenum side. As a consequence, the only source water is in the plenum side, so the density of the plenum side in DCD Table 12.2-1 is equal to that of cold water: 1.0 g/cm^3=62.4 lbm/ft^3. However, in case of the steam generator and the regenerative heat exchanger, where high temperature and high pressure water is used, the density is 44.5 lbm/ft^3.
For the shell side, on the other hand, the model density is a value that combines the individual densities of the component cooling water, the U-shaped tube stainless steel, and the source water inside the U-shaped tube. The shell side density is obtained by the following equations:

$$V_{CW} = (d/2)^2 \times \pi \times L - V_{UT} - V_S$$
$$V_{UT} = (d_{PO}/2)^2 \times \pi \times L \times 2n - (d_{PI}/2)^2 \times \pi \times L \times 2n$$
$$V_S = (d_{PI}/2)^2 \times \pi \times L \times 2n$$
$$D_{MIX} = (V_{CW} \times D_{WO} + V_{UT} \times D_{UT} + V_S \times D_{WI}) / ((d/2)^2 \times \pi \times L)$$

where,

V_{CW}	: volume of water inside the shell but outside the U-shaped tubes (ft^3)
V_{UT}	: total volume of U-shaped tube material (ft^3)
V_S	: total volume of water in U-shaped tubes (ft^3)
d	: inner diameter of the shell (ft)
L	: length of the shell side (ft)
d_{PO}	: outer diameter of a U-shaped tube (ft)
d_{PI}	: inner diameter of a U-shaped tube (ft)
D_{MIX}	: mixed density of the shell side (lbm/ft^3)
D_{WI}	: density of water inside the U-shaped tubes (lbm/ft^3)
D_{WO}	: density of water outside the U-shaped tubes (lbm/ft^3)
D_{UT}	: density of the U-shaped tube material (lbm/ft^3)
n	: number of U-shaped tubes

Table 1 summarizes the mixed density calculation for the six components in DCD Table 12.2-1 that utilize a density calculated in the manner described above. Note that there are some differences between the mixed density reported in Table 1 and the values in DCD Table 12.2-1 (as given in DCD Revision 1). These differences are a result of changes in the inner diameter of the heat exchanger due to the design progress and some miscalculations in the mixed density of the shell side; therefore, the equipment density has been updated accordingly. DCD Table 12.2-1 will be revised so that the mixed densities are consistent with those shown in Table 1 (see Impact on DCD section below).

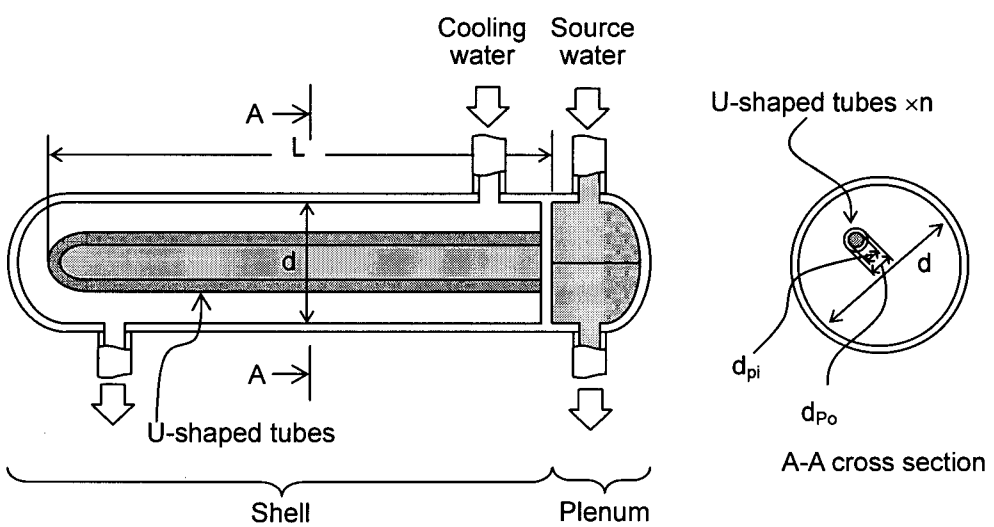

Figure 1 Calculation Model for a Heat Exchanger (Sample: Letdown Heat Exchanger)

Table 1 Parameters for the Calculation of the Mixed Density of the Shell Side of the Heat Exchangers shown in DCD Table 12.2-1

Component	Inner radius of the shell side [d/2]	Length of the shell side [L]	Outer diameter of a U-shaped tube [D_{PO}]	Inner diameter of a U-shaped tube [D_{PI}]	Number of U-shaped tubes [n]	Density of water inside the U-shaped tube [D_{WI}]	Density of water outside the U-shaped tube [D_{WO}]	Density of U-shaped tube [D_{UT}]	Mixed density of the shell side [D_{MIX}]
	(in)	(in)	(in)	(in)	(-)	(lbm/ft^3)	(lbm/ft^3)	(lbm/ft^3)	(lbm/ft^3)
Steam Generator	65.9	434.2	0.75	0.66	6747	44.5	11.3	505.7	69.2
Regenerative Heat Exchanger	8.3	132.4	0.50	0.33	159	62.4	44.5	493.2	121.5
Letdown Heat Exchanger	16.7	189.8	0.75	0.65	224	62.4	62.4	493.2	85.2
Excess Letdown Heat Exchanger	5.9	130.2	0.50	0.36	45	62.4	62.4	493.2	96.7
Containment Spray /Residual Heat Removal Heat Exchanger	31.5	264.4	0.75	0.65	1036	62.4	62.4	493.2	92.2
Seal Water Heat Exchanger	8.4	144.6	0.50	0.41	138	62.4	62.4	493.2	98.2
Values Provided in DCD Table 12.2-1	X	X							X

Impact on DCD

DCD Table 12.2-1 will be revised as indicated in the attachment.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Table 12.2-1 Radiation Sources Parameters (Sheet 1 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Inside the containment vessel							
Steam generator Plenum Side Shell Side	66.9 65.9	63.0 434.2	Homogeneous Homogeneous	Source Water Source Water 22 wt%+ Secondary Water 9wt%+ Steel 69wt%	44.5 69.2	6.1 3.4	4
Regenerative heat exchanger * Plenum Side Shell Side	8.3	23.2 132.4	Homogenous Homogenous	Water (Charging Line) Water (Letdown Line) 26 wt%+ Water (Charging Line) 6 wt%+ Steel 68 wt%	62.4 121.5	2.0	3
Letdown heat exchanger Plenum Side Shell Side	16.7	24.4 189.8	Homogenous Homogenous	Source Water Source Water 13 wt%+ Cooling water 57 wt%+ Steel 31 wt%	62.4 85.2	ignored	1
Excess letdown heat exchanger Plenum Side Shell Side	5.9	21.7 130.2	Homogenous Homogenous	Source Water Source Water 5 wt%+ Cooling water 54 wt%+ Steel 41 wt%	62.4 96.7	1.8 ignored	1

* The regenerative heat exchanger consists of three shells.

Table 12.2-1 Radiation Sources Parameters (Sheet 2 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Outside the containment vessel (Reactor Building)							
Containment spray/residual heat removal heat exchanger Plenum Side Shell Side	31.5	 56.7 264.4	 Homogenous Homogenous	 Source Water Source Water 15 wt%+ Cooling water 48 wt%+ Steel 37 wt%	 62.4 92.2	 1.8 1.2	4
Seal water heat exchanger Plenum Side Shell Side	8.4	 22.3 144.6	 Homogenous Homogenous	 Source Water Source Water 10 wt%+ Cooling water 48 wt%+ Steel 42 wt%	 62.4 98.2	ignored	1
Volume control tank Liquid Phase Vapor Phase	47.2	179.2 107.5 71.7	 Homogenous Homogenous	 Air Water	 7.6E-02 62.4	ignored	1

Enclosure 4

UAP-HF-09473, Rev.0

Responses to Request for Additional Information No.428-2910
Revision 1

September 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 428-2910 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-22

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

Supplemental Question (SQ-1) derive from RAI 171-1858, Question 12.03-12.04-9 and RAI 172-1864, Question 12.03-12.04-10

RAI 171-1858 Question 12.03-12.04-9 and RAI 172-1864 Question 12.03-12.04-10 requested additional information about the radiation protection design features associated with some large piping areas identified in the figures presented in section 12.3. Question 171-1858 12.03-12.04-9 requested that the applicant provide information regarding equipment to be located in some large piping areas and, due to the high dose rates in the areas, what design features were to be provided to keep Operational Radiation Exposure ALARA.

Question 172-1864 12.03-12.04-10 requested that the applicant provide information regarding access to the areas and the physical barriers provided to control access.

In the responses to Questions 12.03-12.04-9 and 12.03-12.04-10, the applicant indicated that:

- Although not shown on the original drawings, access openings to allow visual inspection are now anticipated, with actual locations to be determined at a later date. The applicant noted that Figure 12.3-1 will be revised to include access openings in the general locations noted in the figures attached to the response to RAI 171-1858 Question No. 12.03-12.04-9.

- There are no plans to install equipment that may require access for operation, maintenance or periodic surveillance in these areas.

- One of the areas indicated as a piping area, is actually, a valve area.

These responses appear to be inadequate, or inconsistent with information provided by MHI in other documents, for the following reasons:

1. Based upon the supplied drawings and the size and shape of the areas, changes "to allow visual inspection" to all of the areas, will require personnel access through areas that may have

dose rates up to 500 R/h. (See figure 12.3-1 Sheet 16). These are large areas, which require personnel access, to areas containing components, which are causing high dose rates. The information presented to the staff is not sufficient to allow the staff to determine the sources of radiation exposure to personnel, and the features provided to reduce exposure to personnel in these areas.

2. Contrary to the statement, "There are no plans to install equipment which may require access for operation, maintenance or periodic surveillance in this area", in the response to 172-1864 Question 12.03-12.04-10, the applicant noted that at least one of these areas is actually a valve area. Even relatively inactive components, such as valves, require access for periodic packing adjustment or replacement and actuator maintenance. If a valve in this area is in a system that could contain highly radioactive fluids, as noted in NUREG-0737 III.D.1.1, then periodic access will be required to ensure that ESF leakage remains less than the leakage rate value assumed in the FSAR Chapter 15 accident analysis.

Requested Information

1. Describe the sources of personnel radiation exposure in these areas, and the design features provided to reduce Operational Radiation Exposure (ORE) or provide the specific alternative approaches used and the associated justification.

2. Please clarify the statements provided in the responses Questions 12.03-12.04-9 and 12.03-12.04-10, regarding the type of components installed in these areas.

References

1. "Request for Additional Information No. 171-1858 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3.1, Question No.: 12.03-12.04-9" dated February 3, 2009 CHPB Branch (ADAMS Accession No. ML0906802330)
2. "Request for Additional Information No. 172-1864 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3.3, Question No.: 12.03-12.04-10" dated February 3, 2009 CHPB Branch (ADAMS Accession No. ML0906802350)

ANSWER:

The primary radiation sources in the piping room are process lines which connect various components. For example, there are lines between the B.A. evaporator and the boric acid tanks, lines between each demineralizer and the spent resin storage tanks, lines between the spent resin storage tanks and the spent resin charging room, and so on. The line which has the highest radiation level in the piping room is the spent resin transfer line. Because of its high radiation level, the entire piping room is assigned radiation zone IX. However, this line only causes high radiation when it is used for the replacement of resin in the demineralizer or the shipment of the spent resins from the storage tanks. For the remainder of the time, the radiation level in this room decreases to that of zone VIII. Consequently, personnel access to this area during resin transfer operations is strictly restricted.

There are no components or valves installed in this piping room that require periodical personnel access for maintenance or operation. Furthermore, there is no equipment that requires monitoring for ESF leakage in this piping room, because all the ESF equipment is installed in the Reactor Building.

Additionally, MHI noted in the response to RAI 172-1864 Question 12.03-12.04-10 that a portion of this piping area is actually a valve area. This area is a south-east section of the piping room on the E.L. 15'-9" floor (Figure 12.3-1 sheet 18 of 34), and it is also located at the west side of the holdup tanks. Since this area is a valve area, personnel access is required for the operation or

maintenance of the valves. However, the radiation level of this valve area is lower than that of the piping area and is assigned radiation zone VIII. Furthermore, all the source water in the piping located in this area is to be flushed out prior to the start of repairs to or maintenance of the valve and its attachment. Flushing the source water will result in reduced exposures to personnel working in this area.

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 428-2910 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-23

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

Supplemental Question (SQ-2) derive from RAI 171-1858, Questions 12.03-12.04-7 and 12.03-12.04-8

RAI 171-1858 Questions 12.03-12.04-7 and 12.03-12.04-8 requested additional information about the radiation protection design features associated with the operation and maintenance of the Boric Acid Transfer Pumps (BATP) and the Boric Acid Evaporator (BAE) and the BAE Distillate (BAEDP) and Concentrates (BAECP) Pumps.

In summary, the MHI Response indicated that:

1. Based on other plant experience, actual doses from this equipment will be insignificant.

2. That the filters and Demineralizers would remove all of the activity, so no activity will remain in the boron recycle system.

3. There is no impact on the DCD or COLA.

This response appears to be inadequate and inconsistent with other information provided by MHI:

1. The use of a source term based on operational experience is contrary to the source term stated in the US-APWR FSAR Tier 2 Section 12.2.1.1 "Sources for Full-Power Operation". In this section, the applicant makes the statement that "The design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power". The use of cladding defects for determining the shielding requirements is consistent with the Acceptance Criteria contained in SRP Section 12.2.

Describe the source term used as the basis for determining the dose rates, shielding requirements, airborne activity concentration and ventilation system design parameters, and the resultant ORE

dose reduction design features for normal operation and AOO, and provide the associated revisions to the FSAR, or provide the specific alternative approaches used and the associated justification

2. The assumption that no activity would be present in the process fluid streams after the demineralizers and filters is inconsistent with the NRC Staff Operating Experience (OE) and the applicant response to RAI 168-1739 Question 12.02-14. In that response MHI provided DF and concentration factors that are consistent with staff OE. The use of these removal and concentration factors indicate that appreciable amounts of radioactive material may be present down stream of the BAE package, under normal conditions, but particularly under design basis cladding defect operating conditions.

Describe the removal factors, and their bases, assumed for the responses to questions 12.03-12.04-7 and 12.03-12.04-8, and provide justification for the use removal factors other than those noted in the response to 168-1739 Question 12.02-14.

3. The response indicates that no changes to the DCD or COLA are required. The lack of dose reducing design features for BATP, BAEDP and BAECP is contrary to FSAR Tier 2 Section 12.1.2.3.2 "Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment" which notes that general design criteria used to reduce radiation levels near equipment requiring personnel attention included providing adequate shielding between radiation sources and access and service areas. This statement in the FSAR is consistent with 10 CFR 20.1101(b) and the Regulatory Position noted in RG 8.8 Position C2.b, regarding recommended design features to reduce maintenance and operation related exposure from high dose rate equipment. Based on US-APWR FSAR Figures provided in section 12.3-1 "Radiation Zones for Normal Operation/Shutdown" the Boric Acid Transfer Pumps (BATP) are located where the maximum dose rate could be 10 R/h in the area and the Boric Acid Evaporator Pumps are located where the maximum dose rate could be 100 R/h in the area.

Consistent with 10 CFR 20.1101(b) and RG 8.8 Position C2.b, describe the design feature provided to reduce ORE associated with the Boron Recycle System components.

References

1. "Request for Additional Information No. 171-1858 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3.1, Question No.: 12.03-12.04-7" dated February 3, 2009 CHPB Branch (ADAMS Accession No. ML0906802330)
2. "Request for Additional Information No. 171-1858 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3.1, Question No.: 12.03-12.04-8" dated February 3, 2009 CHPB Branch (ADAMS Accession No. ML0906802330)
3. "Request for Additional Information No. 168-1739 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2,", Question No.: 12.02-14" dated February 3, 2009 CHPB Branch (ADAMS Accession No. ML090650632)

ANSWER:

Answer to 1

The source terms used as the basis for shielding design and dose rate calculations in the vicinity of the boric acid evaporator (BAE) and its associated transfer pumps are derived from the source terms for the Holdup Tank presented in DCD Chapter 12, Table 12.2-26. The Holdup Tank source term values were calculated based on the design basis conditions of 1% defective fuel cladding. The source term values for the Holdup Tank were adjusted to develop the activities in the evaporator by applying the decontamination factor associated with the BAE Feed

Demineralizer. A decontamination factor of 10 was conservatively applied based on the recommendations of NUREG-0017. The source terms for the BAE are tabulated in Table 12.2-66 "Chemical and Volume Control System Radiation Sources B.A. Evaporator Activity" which will be added to the DCD as defined in the response to DCD RAI NO. 142-1733 Revision 1, Question No. 12.02-8.

Answer to 2

As indicated in the response to RAI 171-1885 Question 12.03-12.04-7, MHI agrees that the design basis source terms for the boron removal system components results in a significant radiation dose, as indicated by the DCD analysis of the design basis operating conditions assuming cladding defects. The removal factors provided in the response to RAI 168-1739 Question 12.02-14 are the removal factors used for the design basis analysis presented in the DCD. However, as indicated in the response to RAI 171-1885 Question 12.03-12.04-7, Japanese operational experience has indicated that the various filters associated with the boron removal subsystem are effective at removing suspended radioactive matter and that the "realistic" doses are significantly less than the design basis doses resulting from assumed cladding defects.

DCD Subsection 12.3.1.1.1.2 provides a description of common equipment and component designs for balance of plant equipment that are aimed at keeping maintenance doses ALARA. Part C of this subsection addresses evaporators and states that adequate space and flanged connections for easy removal are provided for the maintenance of evaporator components. Additionally, the evaporator can be operated in an automatic mode that can reduce the exposure of the operator to radiation from the equipment. Part D of this subsection addresses pumps and states that pumps are sealed with mechanical seals to reduce seal servicing time, when practical. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. All pumps in the radioactive waste systems, as well as other small pumps, are provided with flanged connections for ease of removal so that maintenance can be performed in another location. Additionally, pump casings are provided with drain connections for draining pumps for maintenance.

As a result of the common design practices to provide for adequate space and ease of removal of the components for maintenance coupled with the knowledge that "realistic" doses are significantly lower than the design basis doses reported in the DCD, MHI concluded that partitions between the components would reduce the amount of work space making it more difficult and time consuming to perform work and thus installation of shielding between the components was not consistent with ALARA principles.

Answer to 3

The ORE dose reduction design features include locating the BAE (and its associated transfer pumps, when potentially containing radioactive fluids) in shielded vaults with external walls of a thickness of 2.5 feet. The internal walls and floor are coated with epoxy to ease cleanup of potential contamination. A steel door, of sufficient thickness to provide the shielding equivalent of the 2.5 feet thick concrete walls, is provided for the vault to limit access and provide additional shielding. Airborne activity in the vicinity of the BAE is very low due to its leak-tight design. In addition, the off-gas from the unit is vented directly to the Gaseous Waste Management System (GWMS) for treatment and release.

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 428-2910 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-24

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

Supplemental Question (SQ-3) derive from RAI 147-1850, Question 12.03-12.04-4

RAI 147-1850 Question 12.03-12.04-4 asked the applicant to provide the allowable cobalt content of components in contact with the RCS.

In summary, the MHI Response indicated that:

MHI would provide table 12.3-2 for equipment cobalt specifications. This response included the following cobalt content specifications:

- Fuel assemblies – 0.05 = 500 ppm
- High Neutron flux – 0.05 = 500 ppm
- Upper/Lower core plates 0.10 = 1000 ppm
- RCS Piping, RCP Reactor Internals 0.20 = 2000 ppm

Based on this response, the allowable cobalt content for some components located in high neutron flux areas, are above recommended cobalt impurity levels provided in some industry guidance documents. Due to the significance of irradiated cobalt to Occupational Radiation Exposure (ORE), RG 8.8 C.2(e)(1) provided guidance regarding establishing specifications, to the extent practicable, for the use of low-cobalt bearing materials for primary coolant pipe, vessel internal surfaces. This guidance is consistent with the observations and recommendations noted in industry documents, including:

- EPRI TR-107991 1997 "Radiation Field Control Manual", which notes that because radiation from Co-60 accounts for >90% of ORE, the cobalt content of materials exposed to high neutron flux should be limited to 50 ppm,

- EUR 8655 "The Control of Cobalt Content In Reactor Grade Steels", which notes that residual cobalt contamination is the main source of radiation from activated structural components. This reported noted that the core barrel and shroud were the major sources of activation, due to the proximity to the neutron flux, and that the expense associated with feed material selection, may be offset by reductions in decommissioning expenses.

In light of the increased Occupational Radiation Exposure and Decommissioning costs associated with the use of higher cobalt materials located in high neutron flux areas, justify the stated allowable cobalt content in materials located in high neutron flux areas.

References

1. Response to "Request for Additional Information No. 147-1850 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.03, Question No.: 12.03-12.04-4" dated January 9, 2009 CHPB Branch (ADAMS Accession No. ML0090410552)

ANSWER:

EPRI TR-1003390 "Radiation Field Control Manual" (Final Report, December 2004) provides cobalt impurity levels in Section "3 COBALT SOURCE REDUCTION", subsection "Limiting Impurity Levels of Cobalt in Structural Alloys" as follows:

> "Cobalt impurity levels should be less than 500 ppm in stainless steels and less than 200 ppm in Inconels for all nuclear replacement components. There is little cobalt present in the zirconium-base Zircaloy alloys used for fuel rod cladding and in some fuel spacer grids, with the cobalt content of Zircaloy typically less than 50 ppm."

Consequently, the EPRI recommended cobalt impurity level for structural materials is not 50 ppm, as indicated in the NRC's question. Instead, EPRI recommends that cobalt should be limited to 500 ppm in stainless steel and 200 ppm in Inconel. Further, EPRI states that Zircaloy alloys used in fuel rod cladding typically contain less than 50 ppm cobalt impurities.

MHI's cobalt content specifications to be listed in Table 12.3-7 are for structural alloys and do not specify Zircaloy Fuel Clad. Therefore, those MHI specifications are in accordance with the recommended cobalt impurity levels provided in industry guidance documents for materials used in high neutron flux areas.

Reference: 1) "MHI's response to US-APWR DCD RAI No. 147-1850 Revision 1" by the letter UAP-HF-09048 dated February 6, 2009.

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.

Enclosure 5

UAP-HF-09473, Rev.0

Responses to Request for Additional Information No.429-3178
Revision 1

September 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-25

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" describes design features for radiation monitoring equipment provided for maintaining personnel exposure ALARA during operation including AAO and accident conditions, maintenance and refueling activities.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Question 12.03-12.04-13

Supplemental Question 8866-SQ-1

- RAI 262-1972, Question 12.03-12.04-13 (Question 8866-Q1) requested additional radiation monitors at areas potentially subject to large transient dose rates, including the SI and RHR Pump and Heat Exchanger areas and other operational areas that could be impacted by transient high dose rates.

In the responses to this question, the applicant indicated that portable radiation monitoring equipment will be used prior to entering high dose rate areas, so no additional ARMs are considered necessary.

This response is inadequate because:
In FSAR Tier 2, Table 1.9.2-12 the Applicant, does not state any exceptions to SRP Section 12.3-12.4 "Facility Design Features" Subsection 4 "Area Radiation and Airborne Radioactivity Monitoring Instrumentation". SRP Section 12.3-12.4 I.4.G and RG 1.206 Regulatory Position C.I.12.3.4 state that criteria for selection and placement of the instrumentation are in accordance with ANSI/ANS-HPSSC-6.8.1. The US-APWR FSAR Tier 2, Section 12.3.4 states that the ARMS are in conformance with ANSI/ANS HPSSC-6.8.1. ANSI/ANS HPSSC-6.8.1 Section 4.2.1 and 4.2.2 note that detectors shall be located in areas subject to significant changes in dose rates, due to operational transients or maintenance activities. ANSI/ANS HPSSC-6.8.1 Table 2 specifically mentions the RHR pumps, heat exchangers cask handling, fuel storage and waste processing areas.

Contrary to the guidance contained in SRP Section 12.3-12.4, RG 1.206 C.I.12.4 and ANSI/ANS HPSSC-6.8.1, the applicant has stated that Area
Radiation Monitors (ARMs) are not required in some areas potentially subject to significant

changes in dose rates, such as the RHR pumps, SI Pumps, RHR/SI Heat Exchanger, Cask Loading and Fuel Inspection Pits.

Based on the above regulatory guidance, the applicant should either: (1) provide ARMs in those areas where changes in plant conditions or maintenance can cause significant increases in personnel exposure rates above that expected for the area, or (2) revise and update the FSAR to describe and justify how the current placement of ARMs meets the intent of RG 1.206 and ANSI/ANS HPSSC-6.8.1.

Supplemental Question 8866-SQ-2

- RAI 262-1972, Question 12.03-12.04-13 (Question 8866-Q2) requested that the Applicant modify FSAR Tier 2 Table 12.3-5 to provide the classification of install Airborne Activity Monitoring Equipment.

In the responses to this question, the applicant indicated that the plant effluent monitors are Type E, and the other monitors are not PAM instruments.

This response is inadequate because:
In FSAR Tier 2, Table 1.9.2-12 the Applicant, does not state any exceptions to SRP Section 12.3-12.4 "Facility Design Features" Subsection I.4 "Area Radiation and Airborne Radioactivity Monitoring Instrumentation" or RG-1.97 classification criteria, for radiation monitoring equipment. The US-APWR FSAR Tier 2, Section 7.5.2.1 "Post Accident Monitoring" states that RG 1.97 endorses IEEE Std 497-2002 and that for the US-APWR, specific PAM variables are selected based on the criteria described in IEEE Std 497-2002. IEEE Std 497-2002 Subsection 4.5, Selection Criteria for Type E variables, states that Type E variables shall include criteria c) and d) of that section, which explicitly specifies monitors for radiation levels and radioactivity in the plant environs and selected plant areas where access may be required for plant recovery, in addition to plant effluent radiation monitors.

Contrary to the guidance contained in IEEE Std 497-2002 Subsection 4.5, Selection Criteria for Type E variables, in plant airborne radioactivity monitors are not classified as Type E equipment.

Based on the above regulatory guidance, the applicant should either: (1) provide Type Classifications for in plant airborne monitoring equipment consistent with RG-1.97 and IEEE Std 497-2002 or (2) revise and update the FSAR to describe and justify how the classification of these monitors meets the intent of RG 1.97 and IEEE Std 497-2002.

Supplemental Question 8866-SQ-3

- RAI 262-1972, Question 12.03-12.04-13 (Question 8866-Q4) requested that the Applicant provide information regarding the calibration frequency and methods for installed plant ARM and PRM equipment.

In the responses to this question, the applicant indicated that the calibration methods and frequencies will be deferred to the detailed design phase. The Operational Radiation Protection Program will be developed by the COL applicant and will include provisions for the definition of monitoring equipment.

This response is inadequate because:
RG 1.206 Section C.I.12.4 specifically states that the applicant is to provide the calibration methods and frequencies for the installed ARM and PRM equipment. 10CFR50 General Design Criteria notes that components important to safety are those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public as noted in GDC 19, 60, 63 and 64. Some of the monitors addressed in

this question have automatic functions that initiate protective measures for the MCR, while others allow the MCR to direct manual actions that limit the dose impact to the public. Other monitors serve to alert plant operators of abnormal dose rates during the performance of their activities supporting AOO or accident mitigating actions. Branch Technical Position 7-12 "Guidance on Establishing and Maintaining Instrument Setpoints" provides guidance for methods to establish and maintain calibration intervals, by referencing IEEE 603, ISA-S67.04. Additional regulatory guidance is been provided in RG 1.105 Rev 3 "Setpoints for Safety-Related Instrumentation". MHI document MUAP-07004-P(R2) "Safety I&C System Description and Design Process" describes the processes to be used for establishing similar criteria for other plant equipment. This methodology is used in conjunction with vendor-supplied data to determine the actual setpoints and calibration intervals. The Applicant response that the Operational Radiation Protection Program will control the calibration of installed plant equipment is inappropriate because the RP program controls portable instrumentation and equipment, but it is not applicable to installed ARMs and PRMs.

No guidance has been provided to the COL applicant regarding the methodology to be used to determine ARM and PRM setpoints and calibrations. The US-APWR FSAR Tier 2 Chapter 12 does not contain a COL action Item for establishing the ARM and PRM calibration procedures. The US-APWR FSAR Tier 2 Section 14.2.12.1.78 does not contain a criteria for verifying the adequacy of the instrument uncertainty and setpoint determinations.

Based on the above regulatory guidance, the applicant should either:
(1) Revise the US-APWR FSAR Tier 2 to:
- Describe the methodology to be employed for determining the calibration interval and set points for installed ARM and PRM monitoring equipment.
- Add a COL Action Item for the development of calibration procedures for the ARM and PRM system based on the results of the analysis of vendor data using the established methodology.
- Add an item to the US-APWR FSAR Tier 2 Section 14.2.12.1.78 to verify the instrument uncertainty and setpoint determinations. Or (2) revise and update the FSAR to describe and justify how the current method of establishing the calibration frequency and set point determination, meets the intent of the regulatory guidance noted above.

References
1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-13 " dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

ANSWER to Supplemental Question 8866-SQ-1
Although ANSI/ANS HPSSC-6.8.1-1981 was withdrawn in May 1992, both SRP 12.3-12.4 I.4.G and Regulatory Guide (RG) 1.206 Position C.I.12.3.4 require that the selection and placement of area radiation monitors (ARMs) be in accordance with the standard. ANSI/ANS HPSSC-6.8.1-1981 Section 5.4 states that the locations of the ARMs shall be chosen to satisfy the requirements of Section 4.2, which states that detectors shall be located in areas subject to significant changes in dose rates, due to operational transients or maintenance activities. Table 2 of the standard provides *example* locations that may meet the criteria described in Section 4.2 and therefore *shall be considered* when specifying plant area monitoring requirements. However, the locations in Table 2 are not requirements. This specific requirement from the standard is reiterated in the second bulleted item in DCD Subsection 12.3.4.1.2, which describes the US-APWR criteria for the location of ARMs.

DCD Subsection 12.3.4.1.2 goes on to provide a list of the specific locations where fixed ARMs are to be installed, as well as a list of areas which will utilize portable ARMs during work activities.

These portable ARMs are utilized to alert workers of increasing radiation levels in these and other areas where radiation levels could increase significantly due to transients or maintenance when personnel are present. The reviewer specifically requested that fixed ARMs be provided in the RHR and SI pump and heat exchanger areas since these areas meet the "significant change in dose rate" criteria of ANSI/ANS-6.8.1 Section 4.2. However, as indicated on DCD Figure 12.3.1 Sheet 4 of 34 and DCD Table 12.3-2, these rooms are restricted access, limited occupancy areas and are therefore normally inaccessible with locked doors and positive control of access. Under these access control conditions, it is not necessary to have fixed ARMs installed to monitor for personnel safety and the use of a portable instrument upon personnel access is considered sufficient.

ANSWER to Supplemental Question 8866-SQ-2
MHI previously described the selection basis for Type E PAM variables in Table 07.05-8.5 of the response to question 07.05-8 of RAI 238-2030. This table will be incorporated in DCD Revision 2 as Table 7.5-10 per the mark-up provided in MUAP-09003 Revision 3, "US-APWR DCD RAI Tracking Report". In this table, the following Type E required functions, which are consistent with IEEE Std. 497-2002, are indicated:

1) Monitor radiation and radioactivity levels in the control room and selected plant areas where access may be required for plant recovery.
2) Monitor the magnitude of releases of radioactive materials through identified pathways.
3) Monitor radiation levels and radioactivity in the plant environs.
4) Monitor the environmental conditions used to determine the impact of releases of radioactive materials through identified pathways.

Additionally, RAI 238-2030 Table 07.05-8.5 indicates that function 3) to measure plant environmental radiation and radioactivity will be handled by portable Type E instrumentation. DCD Table 12.3-5 describes the airborne radioactivity monitors located in the HVAC exhaust ducts used to measure normal radioactivity and thus provide indication of an abnormal release of radioactive material from the specific cubicles being monitored. As such, these monitors are not depended upon to monitor Type E radiation levels and radioactivity in the plant and thus these monitors are not considered Type E PAM instruments.

ANSWER to Supplemental Question 8866-SQ-3
·The methodology to determine the calibration interval and setpoints are described in the Section 7.2.2.7 because instruments, ARM and PRM are the part of instrumentation system. We will add a description to refer to the section 7.2.2.7 for the calibration interval and setpoints.
· The calibration procedures for the ARM and PRM are included in the Section 13.5.2.2. We will add a description to refer to the section 13.5.2.2 for the calibration procedures.
· We will add an item.

Impact on DCD
According to the ANSWER to Supplemental QUESTION 8866-SQ-1, DCD Subsection 12.3.4.1.2 will be revised as indicated below to clarify the reasoning behind the use of portable rather than fixed ARMs, in accordance with the guidance of ANSI/ANS-6.8.1.

12.3.4.1.2 Criteria for Location of Area Radiation Monitors

The locations of the area radiation monitors are shown in Figure 12.3-1.
Considerations for area radiation monitor locations include:

• Areas which are normally accessible, and where changes in plant conditions can cause significant increases in personnel exposure rate above that expected for the area

• Areas which are normally accessible and occasionally accessible where a significant increase in exposure rate resulting from operational transients or maintenance activities may occur
• The containment area where the level of radioactivity needs to be monitored to detect the presence of fission products during a DBA
• Area monitor detectors are located such that inadvertent shielding by structural materials is minimized
• In the selection of area monitors, consideration is given to the range of temperature, pressure and humidity of the areas where the detectors or electronics are located

The ARMS provides a continuous, direct indication or recording of radiation levels in the control room and raises alarms locally and in the control room when radiation levels exceed the set values.

The fixed area monitors are installed in the following locations~~:~~ to warn occupants of the area of a deteriorated radiological condition:

(a) MCR
(b) Inside of the containment
(c) Radio Chemical Lab
(d) SFP area
(e) Nuclear sampling room
(f) Inside of the containment (near the air lock)
(g) Inside of the containment (near the ICIS)
(h) Waste Management System (WMS) area
(i) TSC

For areas with positive access control features, such as normally locked doors, or areas where a radiological hazard only exists during specific work activities, a fixed ARM is not required. Instead, ~~Furthermore, during work activities,~~ a portable ARM~~S~~ is installed to warn occupants of a deteriorated radiological condition. Portable ARMs are utilized in the following locations:

(j) Refueling platform
(k) Residual heat removal pump and heat exchanger areas
(l) Hot machine shop
(m) HVAC filter area
(n) Cask handling area
(o) Equipment decontamination area
(p) Safe shutdown panel area

According to the ANSWER to Supplemental QUESTION 8866-SQ-3, the following portions of the DCD are to be revised as indicated below.

· we will add the description to the DCD Section 12.3.4.1.9 as below.
"Alarm setpoints are controlled by plant procedures and the offsite dose calculation manual, where appropriate. And the methodology to determine the calibration interval and setpoints for the Area Radiation Monitors and Process and effluent Radiation Monitors are described in the Section 7.2.2.7. The calibration procedures are described in the Section 13.5.2.2. "

· We will add the description to the DCD Section 12.3.4.1.9 as above.

· We will add an item to the Section C. of the DCD Section 14.2.12.1.78 as described below.
" C. Test Method
3. The uncertainty and determination of setpoint of each monitor is verified. "

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-26

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Questions 12.03-12.04-15

Supplemental Question 8688-SQ-1

RAI 262-1972, Question 12.03-12.04-15 (Question 8688-Q3) requested additional information regarding COL applicant use of the Mobile Liquid Waste Processing provisions.

In the responses to this question, the applicant indicated that the COL applicant would address this as part of COL 12.2 (1), and they assume that this would lead to changes in FSAR Section 12.3.

This response is inadequate because:
SRP Section 12.03-12.04 notes that the shielding design is to provide protection for operating personnel inside and outside the plant, and for the general public, and that for those areas where other than permanent
shielding is used, that alarming radiation monitors will be installed to alert personnel if shielding is removed. RG 1.206 Section C.I.12.3.2 notes that the applicant is to provide the models, codes parameters and assumptions used to demonstrate compliance with RG 1.69. The mobile liquid waste processing system is located on the grade level elevation, but no information regarding provisions for prevent of contamination of the facility or environment, in accordance with 10CFR20.1406 and RG 4.21, have been provided.

Contrary to the guidance contained in RG 1.206 and SRP Section 12.3-12.4 insufficient guidance is provided to the COL Applicant regarding the actions needed to support the use of the Mobile Liquid Waste Processing System.

Based on the above regulatory guidance, the applicant should either (1) revise and updated US-APWR FSAR Tier 2 Section 12.3.6 to add COL Item(s) to provide direction to those COL

Applicants utilizing the Mobile Liquid Waste Processing System to:

- Establish the system inlet activity limits, if different from those assumed for the Chapter 12 Shielding Design Bases, and provide the shielding design analysis consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69, which supports the regulatory limits of 10 CFR 20 and 40 CFR 190.
- Update the 10 CFR 20.1406 analysis for prevention and detection of contamination of the environment and minimization of decommissioning costs.
- Update the Radiation Zone figures provided in Figure 12.3-1. Or, (2) describe and justify how the current design meets the intent of the regulatory guidance.

References

1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-15 (Question 8688-Q3)" dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

The Mobile Liquid Waste Processing System is an optional system available for the US-APWR. Therefore, the specifications for this system are not available until a utility who adopts the US-APWR decides to utilize the system. A utility that chooses to utilize the system, must also determines the manufacturer from which to procure the system. Consequently, MHI does not describe the specifications, design features, etc. of the Mobile Liquid Waste Processing System in the US-APWR DCD. Instead, MHI will revise the DCD to add three COL Action Items in Subsection 12.3.6 as shown below:

COL 12.3(6) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69.

COL 12.3(7) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21.

COL 12.3(8) If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.

Impact on DCD

The three COL Action Items shown above will be added to Subsection 12.3.6.

Impact on COLA

The COLA will be revised to incorporate the addition of the three new COL items to the DCD.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-27

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Question 12.03-12.04-16

Supplemental Question 8815-SQ-1

RAI 262-1972, Question 12.03-12.04-16 (Question 8815-Q1) requested additional information regarding post accident mission doses to areas like the RHR pumps and heat exchangers, Safety Injection Pumps, and waste treatment areas.

In the responses to this question, the applicant indicated that no entries would be required for the first 30 days following the accident.

This response is inadequate because:
CHPB Staff discussed this response with the staff in the Branches responsible for plant operations and the review of the SRP sections of the equipment identified in 8815-Q1. The staff responsible for these areas have indicated that it is not reasonable to assume that operator action to realign, maintain monitor equipment that was running prior to the event, or that would be running after the event would not be required. It is also not clear to the staff if any manual operator actions inside radiologically affected areas of the plant as a result of compensatory Emergency Operating Procedure steps.

Please either: (1) Provide mission paths and doses associated with post LOCA activities, such as EOP directed manual operator actions, post trip equipment realignment, operation and monitoring of equipment like that noted in the question 8815-Q1, or (2) describe and justify how safe operation of plant equipment will assured during and following an accident.

Supplemental Question 8815-SQ-2

RAI 262-1972, Question 12.03-12.04-16 (Question 8815-Q2) requested additional information

regarding mission doses for some EQ equipment with short service requirement intervals. In the responses to this question, the applicant indicated that only the Main Steam Line pressure transmitters need to be addressed post accident, and they are located in a low dose rate area.

This response is inadequate because:
US-APWR FSAR Tier 2 Table 3D-2 has numerous other components that have 36 hours or two week service durations. Many of these components appear to be located in high dose rate areas of the plant. Examples include, but are not limited to:

- Sheet 2 of 57 – CVS-FT-218 and 219, SIS-FT-962-975
- Sheet 3 of 57 – SIS-PT-960-964 and SIS-PT-954-957
- Sheet 3 of 57 – RHS-PT-600, 601, 610 and 611
- Sheet 3 of 57 – RHS-FT-601, 604, 611, 614, 621, 624, 631 & 634
- Sheet 4 of 57 – RHS-PT-620, 621, 630 and 631
- Sheet 16 of 57 – VRS-TS-2730, 2733, 2734, 2735 and 2738
- Sheet 28 of 57 – RSC (remote shutdown console)
- Sheet 28 of 57 - CVS-LCV-121F and CVS-LCV-121G

Contrary to the response provided by the applicant, there are numerous examples of equipment with limited service life, installed in equipment expected to be in operation during a post LOCA environment, and located in areas subject to potential large post accident dose rates.

Based on the regulatory guidance provided in 10 CFR 50 GDC-19, NUREG-0800 and RG-8.8 C.1.12.3.5, the applicant should either: (1) revise and update USAPWR FSAR Tier 2 Section 12.3.3.1.2.2, and the associated figures and tables, to show mission paths and describe those design exposure values associated maintaining qualified equipment, or (2) describe and justify how the equipment qualification, for equipment like that noted above, will be maintained without personnel entries into radiologically controlled areas.

Supplemental Question 8815-SQ-3

RAI 262-1972, Question 12.03-12.04-16 (Question 8815-Q3) requested additional information regarding the airborne activity contribution to estimated mission doses provided in Table 12.3-3.

In the responses to this question, the applicant indicated that portable instruments would be used to determine airborne activity contributions in plant areas using the criteria of NUREG 0737 III.D.3.3 and that this was the responsibility of the COL applicant, as identified in COL Item 12.3(1).

This response is inadequate because:
The requirements of 10 CFR 50.34(f)(2)(xxvii)/(vii) and the criteria in Item III.D.3.3 of NUREG-0737 only address the requirement to have portable iodine sampling and analysis equipment capable of determining airborne activity concentrations from field air samples. SRP Section 12.3-12.4 A.1 requires compliance with 10 CFR 20.1202, which requires summing of internal and external exposure. GDC 19 requires that the dose to personnel not exceed 5 rem whole body. NUREG-0696 "Functional Criteria for Emergency Response Facilities" explicitly notes that sources of dose for transit between facilities include airborne radioactivity. RG 8.15 notes that exposure estimates should consider the impact of respiratory protection devices on worker efficiency.
Contrary to the requirements of 10 CFR 20.1202, the applicant has not considered exposure from airborne radioactive material as part of the exposure estimate.

Based on the above regulatory guidance, the applicant should either revise and update US-APWR FSAR Tier 2 to: (1) change Section 12.3.1.2.2 and Table 12.3-3 to include the estimated exposure from airborne activity as part of the mission dose, or (2) describe and justify how personnel total exposure will be maintained within regulatory limits.

References
1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-16" dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

ANSWER to Supplemental Question 8815-SQ-1
In the response to RAI 262-1972, Question 12.03-12.04-16 (Question 8815-Q1), MHI replied that, for the first 30 days after an accident, no human access would be required for RHR pumps and heat exchangers, safety injection pumps, and waste treatment areas. At this time, MHI has reconfirmed with the respective design sections whether it would be necessary to make access to the equipment rooms (RHR Hx room, RHR pump room, SI pump room, etc.) at the time of an accident. As a result, it was verified that all the instruments and switches installed in these equipment rooms are automatic or remote-operable from the MCR, thus access by operating staff is not required. Additionally, as indicated in DCD Table 3D-2, equipment required to operate to mitigate the consequences of an accident, such as the RHR pumps and heat exchangers, are environmentally qualified in order to assure their operation in a post-accident environment for a period of time in excess of 30 days. In short, MHI reconfirms that access for the first 30 days after an accident would be required only for the MCR, the TSC, the PASS, and the radiation chemistry room as shown in DCD Table 12.3-3.

ANSWER to Supplemental Question 8815-SQ-2
US-APWR FSAR Tier 2 Table 3D-2 was significantly revised as part of the response to RAI 358-2462 Rev. 1 Question No. 03.11-2. This revision was primarily focused on providing the NRC additional detail regarding the location of equipment requiring qualification and additional details regarding the radiation dose to which the equipment must be qualified. The revised US-APWR FSAR Tier 2 Table 3D-2 lists several components installed in radiation harsh areas that have 36 hour or two week operational duration times, including the specific components cited in the question above. Per the requirements of the US-APWR Equipment Environmental Qualification Program these components are designed to maintain their specified functions for the operational duration under the environmental and radiation (total integrated dose) requirements associated with their installed location, without requiring access for repair post-accident. However, some components in Table 3D-2 remain that are installed outside the CV where access can be made even in a radiation environment, with an operational duration of two weeks which is set to allow for repair, replacement or calibration, as described in Table 3D-1.

ANSWER to Supplemental Question 8815-SQ-3
DCD Table 12.3-3 (sheets 2 and 3) provides radiation exposure as a cumulative mission dose for each task, for example access to MCR from AC/B for operation, return to MCR from radiochemical laboratory, and so on. These tasks are performed inside the building, and the workers use respiratory protection devices for the purpose of keeping the dose ALARA. Therefore, in the mission dose evaluation in sheets 2 and 3, only direct dose is considered.

The post accident sampling and analysis activity includes moving from the MCR to the PASS, sampling at the PASS, moving from the PASS to the radiochemical laboratory, analysis at the radiochemical laboratory, and returning to the MCR from the radiochemical laboratory. Mission doses for each of the individual tasks are shown in DCD Table 12.3-3 (sheets 2 and 3). Total mission doses for the post accident sampling and analysis activity, obtained from the sum of the individual tasks, are shown in the following table.

Table: Mission Dose for the post accident sampling and analysis activity

Activity Description	Mission Dose [rem]	Note
Post accident sampling and analysis activity (1 hour after)	1.6E-01	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 2 of 4)
Post accident sampling and analysis activity (1 day to 1 month after)	4.6E-03	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 3 of 4)

Alternatively, the dose calculation for the MCR personnel is described in DCD Subsection 15.6.5.5 and DCD Table 15.6.5-16. In this calculation, the use of respiratory protection devices is not assumed, thus exposure by airborne activity is also considered in the dose calculation. The post-LOCA radiological consequence for MCR personnel after staying 30 days is 4.5 rem. This result is less than the requirement of GDC 19 ("not exceed 5 rem").

To clarify the above considerations and assumptions, DCD Subsection 12.3.1.2.2 will be revised as described in the "Impact on DCD" section of this response.

Impact on DCD

According to the ANSWER to Supplemental Question 8815-SQ-3, the third paragraph of DCD Subsection 12.3.1.2.2 "Accident Conditions" will be revised as follows:

> Projected dose rates and **cumulative** mission dose**s** for **tasks performed within** the vital areas at various times after an accident are given in Table 12.3-3. **In this mission dose evaluation it is assumed that workers use respiratory protection devices, thus only direct dose is considered. Alternatively, the dose calculation for the MCR personnel under accident conditions does not assume the use of respiratory protection devices, and exposure due to airborne activity is considered. The total mission doses for the post accident sampling activity and MCR personnel are given in Table 12.3-3 (sheet 4 of 4).**
>
> The US-APWR is designed to ensure the capability to achieve cold shutdown without subjecting personnel to excessive radiation exposure. This capability is further described in Chapter 7, Section 7.4. Radiation protection design features and access controls are described in Sections 12.3 and 12.5. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates defined on Figures 12.3-3 through 12.3-6 and Table 12.3-3, and appropriate time limits for presence in the area are imposed.

The title of DCD Table 12.3-3 will be revised as follows:

> Table 12.3-3 Projected Dose Rates for the Vital Areas at Various times after an Accident (sheet 1 of ~~3~~ **4**)
>
> Table 12.3-3 Mission Dose for the Vital Areas access route after an Accident (1 hour after) (sheet 2 of ~~3~~ **4**)
>
> Table 12.3-3 Mission Dose for the Vital Areas access route after an Accident (1 day to 1 month after) (sheet 3 of ~~3~~ **4**)

The new table, Table 12.3-3 (sheet 4 of 4), will be added as follows:

Table 12.3-3 Activity Mission Dose (sheet 4 of 4)

Activity Description	Mission Dose [rem]	Note
Post accident sampling and analysis activity (1 hour after)	1.6E-01	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 2 of 4)
Post accident sampling and analysis activity (1 day to 1 month after)	4.6E-03	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 3 of 4)
Operation activity in the MCR (30 days after the LOCA)	4.5	Described in Table 15.6.5-16

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-28

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Question 12.03-12.04-17

Supplemental Question 8816-SQ-1

RAI 262-1972, Question 12.03-12.04-17 (Question 8816-Q1) requested additional information regarding access to the fuel transfer tube inspection area.

In the responses to this question, the applicant indicated that Figures 12.3-1 Sheet 9 & 10 would be changed to show the access points.

This response is inadequate because:
RG 8.38 states that physical barriers should, to the extent practicable, completely enclose very high radiation areas sufficient to thwart undetected circumvention of the barrier. The area enclosed by the barriers depicted on Figure 12.3-1 Sheets 9 and 10 is large and includes one quadrant of the electrical penetration area. Figure 12.3-1 Sheets 9 and 10 contain insufficient detail to allow the staff to determine if there are any portions of the area accessible to a major portion of a person's body. Also from the information provided, it is not clear to the staff how routine outage activities, such as fire watches, and security rounds will be provided for this area during fuel movement.

Based on the regulatory guidance noted above, the applicant should revise and update the US-APWR Tier 2 FSAR to either: (1) describe those penetrations into the barricaded areas indicated on Sheets 9 and 10, that are large enough to permit access to a major portion of a person's body, and what provisions are provided to thwart entry and those entries into the barricaded areas that will be required during the period of fuel movement, or (2) describe and justify how the current description provides sufficient information to the staff to allow make the determination of the existence of a reasonable assurance of safety.

Supplemental Question 8816-SQ-2

RAI 262-1972, Question 12.03-12.04-17 (Question 8816-Q3) requested additional information regarding preventing movement of irradiated fuel or objects near the Fuel Inspection Pit or Cask Load Pit.

In the responses to this question, the applicant indicated that the pits are normally filled during refueling, do not have drains so if movement of irradiated fuel or components to areas near the weir gates is required the access will be administratively controlled.

This response is inadequate because:
RG 8.38 C.4.4 notes that areas that areas that could become Very High Radiation Areas during certain operational occurrences, such as dropped fuel, should be controlled to provide for ready evacuation of the area. ANSI/ANS HPSSC-6.8.1 Section 4.2.1 and 4.2.2 note that detectors shall be located in areas subject to significant changes in dose rates, due to operational transients or maintenance activities and Table 2 specifically mentions the cask handling areas in the Fuel Building. As noted in INPO TR6-53, there have been a number of incidents where irradiated fuel or components caused unexpected changes in dose rate, due to a failure to meet fuel handling requirements.

Based on the regulatory guidance noted above, the applicant should revise and update the US-APWR Tier 2 Section 12.3, to either: (1) provide installed ARMs in those areas susceptible to significant dose rate changed due to irradiated components or (2) describe and justify how the current placement of ARMs meets the intent of RG 1.206 and ANSI/ANS HPSSC-6.8.1.

References
1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-17" dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

ANSWER to Supplemental Question 8816-SQ-1
Access to the barricaded areas on DCD Figure 12.3-1 Sheets 9 and 10 is controlled by gates and entry to these areas is allowed only through the issuance of a Radiation Work Permit. In addition, entry to these areas is prohibited during the period of fuel movement.

ANSWER to Supplemental Question 8816-SQ-2
The US-APWR Area Radiation Monitor System (ARMS) already includes a fixed ARM in the spent fuel pool (SFP) area. DCD Subsections 12.3.4.1.2 and 12.3.4.1.8 describe the existence and give the functional description of the fixed spent fuel pool (SFP) ARM, respectively. In addition, DCD Figure 12.3-1 Sheet 11 of 34 shows the placement of the SFP ARM with respect to the SFP, Fuel Inspection Pit (FIP), and Cask Loading Pit (CLP). The existence of this fixed ARM is also indicated in DCD Subsection 9.1.2.2.2 which states that "SFP water level and temperature gauges, and an area radiation monitor in the fuel handling area are provided with alarms to the main control room (MCR)."

Additionally, DCD Subsection 12.3.4.1.2 includes provisions for an additional portable ARM in the cask handling area during work activities in that area. Since the radiological hazard due to a dropped fuel incident can only occur when fuel is being handled, it is acceptable to utilize a portable in addition to the fixed SFP ARM during those activities.

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-29

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Question 12.03-12.04-18

Supplemental Question 8817-SQ-1

RAI 262-1972, Question 12.03-12.04-18 (Question 8817-Q1) requested additional information regarding exposure reduction design features for that portion of the ISIS between the reactor head and the ISIS drive area.

In the responses to this question, the applicant indicated that temporary shielding and portable ARMs would be used to control exposure.

This response is inadequate because:
RG 8.38 states that physical barriers (e.g. 6 foot fence) should be used to prevent unauthorized entry into high radiation areas (locked high radiation areas). RG 8.8 notes that design features supporting the installation of portable shielding facilitates exposure reduction. Based on FSAR Figures 7.7-27 and 12.3-1 (sheets 10 and 11), an irradiated component located in the area between the Reactor Head and the ISIS area would be a significant source of exposure to any personnel in the area, such as would be present at the start of a refueling outage.

Contrary to the guidance provided in RG 8.38 and RG 8.8, the area between the reactor head and the ISIS drive area, which is located a considerable distance above the floor elevation, does not appear to have any provisions for:

- Personnel access to allow the installation of temporary shielding and temporary barriers for the ISIS equipment located in the elevated area between the Reactor head and the shield wall,
- Physical support for temporary shielding running between the Reactor Head and the ISIS area.
- Sufficient anchor points for the erection of effective barriers.

Based on the regulatory guidance noted above, the applicant should revise and update the US-APWR Tier 2 FSAR Section 12.3, to either: (1) describe the design features that facilitate the use of the temporary shielding and temporary barriers noted in your response, or (2) describe and justify how the current configuration provides sufficient information to allow the staff to make the determination of a reasonable assurance of safety for plant equipment and personnel working around ISIS equipment located between the Reactor Head and the shield wall.

References
1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-18 (Question 8817-Q1)" dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

ANSWER to Question 8817-SQ-1
RAI 262-1972, Questions 12.03-12.04-18 (Question 8817-Q-1) and 12.03-12.04-19 (Question 8818-Q-3) are RAIs that focused on describing the measures in place to protect personnel when an MD is stuck inside the core. In the responses to these previous questions, MHI stated that to limit excessive radiation exposure, radiation workers are prohibited from being in containment prior to moving the MDs. This means that there would be no personnel working around ICIS equipment located between the Reactor Head and the shield wall during initial MD movement.

Area Radiation Monitor (ARM) RMS-RE-7 is installed near the ICIS drive unit in order to assure that the instrument reading response will indicate the MDs approach to the MD drive unit. If an MD of the ICIS becomes stuck while it is inside the core or when it is being pulled out of the core, only the specific workers who have received a Radiation Work Permit for the ICIS repair work are allowed to enter the CV and participate in solving the problems related to the MD drive unit. These workers are to be engaged in manual operation of the MD drive unit at the location of the MD drive unit (refer to the drawing in Figure 1), while being protected by a temporary mobile shield. It has a sufficient thickness to decrease the exposure and is prepared near the MD drive unit in advance. This will allow the workers to solve the problem of the stuck MD and to perform the extraction work.

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.

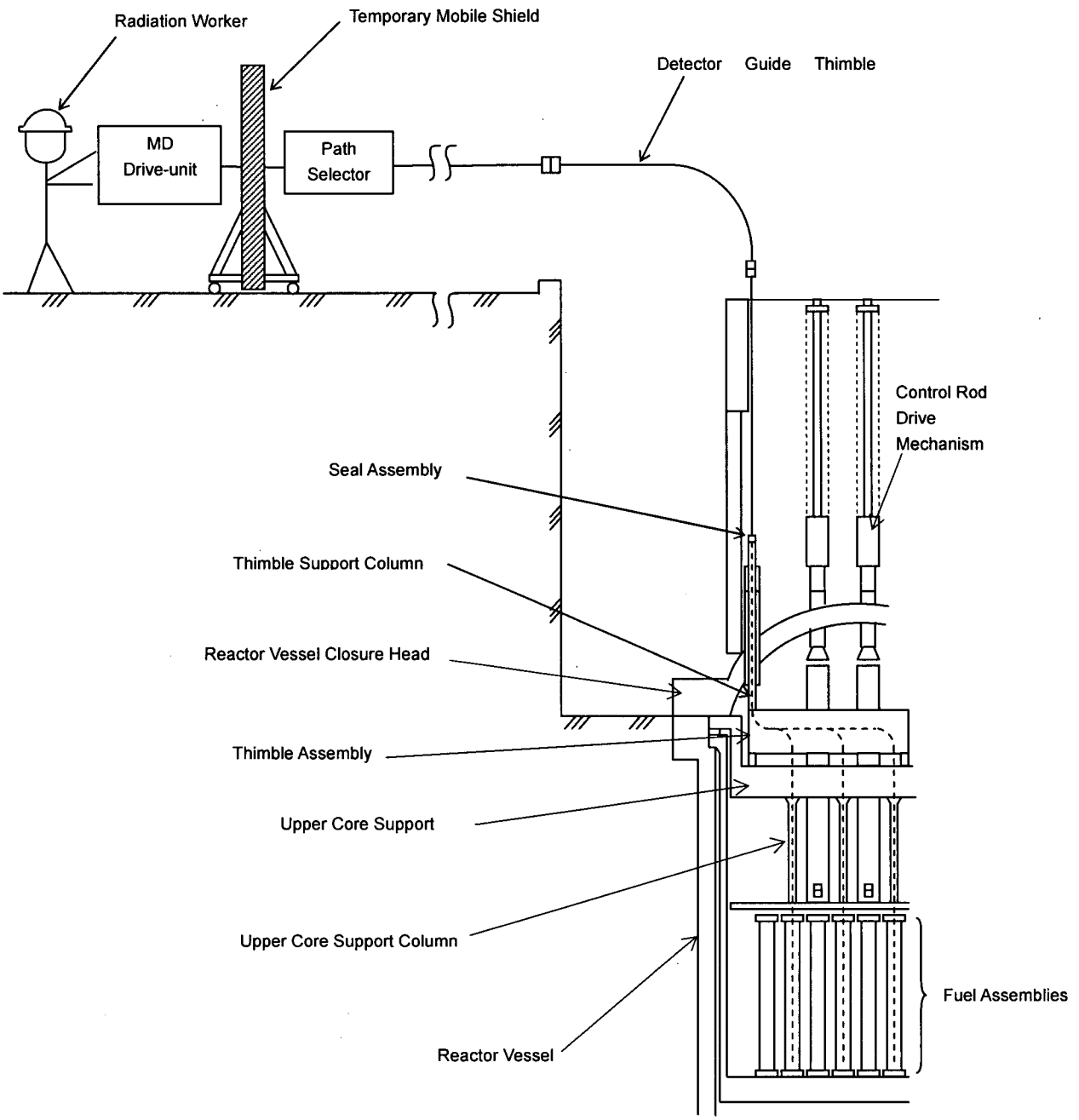

Figure 1 Worker at the MD drive unit Protected by a Temporary Mobile Shield

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-30

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel exposure ALARA and the exposure estimates associated with operation, maintenance and refueling activities.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Question 12.03-12.04-19

Supplemental Question 8818-SQ-1

RAI 262-1972, Question 12.03-12.04-19 (Question 8818-Q3) requested additional information regarding area radiation monitor design features for that portion of the ISIS between the reactor head and the ISIS drive area.

In the responses to this question, the applicant indicated that portable ARMs would be used to control exposure.

This response is inadequate because:
In FSAR Tier 2, Table 1.9.2-12 the Applicant, does not state any exceptions to SRP Section 12.3-12.4 "Facility Design Features" Subsection 4 "Area Radiation and Airborne Radioactivity Monitoring Instrumentation". SRP Section 12.3-12.4 I.4.G and RG 1.206 Regulatory Position C.I.12.3.4 state that criteria for selection and placement of the instrumentation are in accordance with ANSI/ANS-HPSSC-6.8.1. The US-APWR FSAR Tier 2, Section 12.3.4 states that the ARMS are in conformance with ANSI/ANS HPSSC-6.8.1. ANSI/ANS HPSSC-6.8.1 Section 4.2.1 and 4.2.2 note that detectors shall be located in areas subject to significant changes in dose rates, due to operational transients or maintenance activities. ANSI/ANS HPSSC-6.8.1 Table 2 specifically mentions the in core instrument area.

Contrary to the guidance provided in SRP Section 12.3-12.4 and ANSI/ANS-HPSSC-6.8.1, the area between the reactor head and the ISIS drive area, which is unshielded, subject to significant dose rate transients, and accessible to personnel is not provided with a radiation monitor to alert personnel to dose rates greater than those expected for the area during normal operation.

Based on the regulatory guidance noted above, the applicant should either: (1) provide a radiation

monitor for the ISIS equipment located in the area between the Reactor Head and the concrete shield wall, or (2) revise and update the FSAR to describe and justify how the current placement of ARMs meets the intent of RG 1.206 and ANSI/ANS-HPSSC-6.8.1.

References

1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-19 (Question 8818-Q3)" dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

We will change the place of the ICIS area monitor equipment to make it located in the area between the Reactor Head and the concrete shield wall.

Impact on DCD

We will revise the Fig12.3-1(Sheet 11 of 34) to make the ICIS area monitor located in the area between the Reactor Head and the concrete shield wall.

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

9/28/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 429-3178 REVISION 1**
SRP SECTION:	**12.03 – 12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3 – 12.4**
DATE OF RAI ISSUE:	**7/30/2009**

QUESTION NO.: 12.03-12.04-31

The US-APWR FSAR Revision 1 Tier 2 Sections 3 "Radiation Protection Design Features" and Section 12.4 "Dose Assessment", describes design features for maintaining personnel internal exposure ALARA.

These supplemental questions derive from the following original questions:

- RAI 262-1972, Question 12.03-12.04-20

Supplemental Question 8819-SQ-1

RAI 262-1972, Question 12.03-12.04-20 (Question 8819-Q1) requested additional information regarding design features provided in the containment ventilation system for reducing personnel exposure from airborne activity resulting from evaporation and drying of wetted surfaces exposed to air. In the responses to this question, the applicant indicated that the ventilation system, as described in FSAR Section 9.4.6 addressed this question.

This response is inadequate because:
Operating Experience available to the staff, indicates that some current plants have experienced airborne contamination events in containment from evaporation from the Refueling Cavity or drying of wetted RCS internal components exposed to air, that were poorly mitigated by, or exacerbated by, the containment ventilation system. These events resulted in contamination of large areas of the refueling elevation and personnel in the area. FSAR section 9.4.6 only states that the capacity of the containment high volume purge system is sized to maintain acceptably low levels of radioactivity, including noble gases, during refueling operations, that it has filtration equipment on the exhaust and that the exhaust is equipped with a radiation monitor. The section does not describe the design features provided to address airborne contamination as noted in OE, that is caused by evaporation from the Refueling Cavity or drying of RCS internal components exposed to air.

Based on the regulatory requirements of 10CFR20.1101(b), 1201 and 1202, the applicant should revise and update the US-APWR Tier 2, Section 12.3 to either: (1) describe the design features that will be provided around the Refueling Cavity to prevent or mitigate airborne contamination caused by evaporation from the Refueling Cavity or drying of RCS internal components exposed to air, or (2) describe and justify why the current ventilation equipment is sufficient to ensure a reasonable assurance of safety for personnel working in the area.

References
1. "Request for Additional Information No. 262-7972 Revision 1, SRP Section: 12.03-12.04 - Radiation Protection Design Features, Application Section: 12.3,", Question No.: 12.03-12.04-20 (Question 8819-Q1)" dated March 3, 2009 CHPB Branch (ADAMS Accession No. ML091320442)

ANSWER:

1) The Containment High Volume Purge System shown in Figure 9.4.6-1 consists of Containment High Volume Purge air handling unit and an exhaust filtration unit and is designed to maintain acceptably low levels of radioactivity, including noble gases, during refueling operations.
This is not safety-related system. Therefore, this system is not designed as post-LOCA atmosphere cleanup system. The guidance provided by Regulatory Guide 1.140 is applicable to the exhaust filtration unit of the containment high volume purge system.
The exhaust filtration unit consists of, in the direction of airflow, a high efficiency filter, a high-efficiency particulate air (HEPA) filter, and an exhaust fan. HEPA filters are constructed, qualified, and tested in accordance with UL-586 and ASME N509 (Reference 9.4.8-1), Section 9.4.8. Each HEPA filter cell is individually shop tested to verify efficiency of at least 99.97% using a mono-disperse of 0.3 μm aerosol. To reduce the particulate load on the HEPA filters and extend their service life, a high efficiency filters upstream from the HEPA filter section is installed.

2) The airflow capacity of the containment high volume purge system is based on the dose analysis and design consideration (Refer to RAI No.06.05.01-1, 6.5.1-10). The sizing of the Containment High Volume Purge System Filtration Unit is based on the RG 1.140.C.3.6. "Therefore, MHI believes that the Containment High Volume Purge system is adequately sized for safety of personnel working in the area and also satisfies the dose evaluation for occupational radiation exposure during a refueling operation as described in DCD Revision 1, Subsection 12.4.1.

3)Airborne activities due to evaporate from the Refueling Cavity during refueling have been discussed in DCD Subsection 12.2.2.1. Airborne activities estimation due to dry RCS internal components is not needed because wearing a respiratory mask and installation of a temporary area exhaust equipment protect workers from airborne contamination caused by drying RCS internal components exposed to air.

Impact on DCD
In association with answer 3), DCD Subsection 12.4.1 is to be revised.

Exposure data obtained from operating plants have been reviewed to obtain a breakdown of the doses incurred within each category. For several routinely performed operations, this information has been used to develop detailed dose predictive models. These models identify the various steps that are included in the operation, radiation zones, required number of workers, and the time to perform each step. This information has been used to develop dose estimates for each of the preceding categories. There is no separate determination of doses due to airborne activity. Experience demonstrates that the dose from airborne activity is not a significant contributor to the total doses <u>and wearing a respiratory mask and installation of a temporary area exhaust equipment protect workers from airborne contamination caused by drying RCS internal components exposed to air.</u>

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.