MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

July 16, 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-09378

Subject: **MHI's Responses to US-APWR DCD RAI 400-3032 Rev.0, 401-3031 Rev.0, 402-3028 Rev.0 and 403-3027 Rev.0**

Reference:
1) "Request for Additional Information No.400-3032 Revision 0, SRP Section: 11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems, Application Section: 11.5" dates June 18, 2009.
2) "Request for Additional Information No.401-3031 Revision 0, SRP Section: 11.04 – Solid Waste Management System, Application Section: 11.4" dates June 18, 2009.
3) "Request for Additional Information No.402-3028 Revision 0, SRP Section: 11.03 – Gaseous Waste Management System, Application Section: 11.3" dates June 18, 2009.
4) "Request for Additional Information No.403-3027 Revision 0, SRP Section: 11.02 – Liquid Waste Management System, Application Section: 11.2" dates June 18, 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 4.

As indicated in the enclosed materials, documents (Enclosure 2 and 5) and attachment data (Enclosure 8) 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 and 6). In the non-proprietary versions, the proprietary information, bracketed in the proprietary versions, is replaced by the designation "[ ]".

This letter includes the proprietary documents (Enclosure 2 and 5), non-proprietary documents (Enclosure 3,4,6 and 7) and proprietary digital data (Enclosure 8), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2,5 and 8 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.400-3032 Rev.0 (proprietary)

3. Responses to Request for Additional Information No.400-3032 Rev.0 (non-proprietary)

4. Responses to Request for Additional Information No.401-3031 Rev.0 (non-proprietary)

5. Responses to Request for Additional Information No.402-3028 Rev.0 (proprietary)

6. Responses to Request for Additional Information No.402-3028 Rev.0 (non-proprietary)

7. Responses to Request for Additional Information No.403-3027 Rev.0 (non-proprietary)

8. CD1:"Attachment of Responses to RAI's items 11.02-20 and 11.03-12 of NRC Requests"

The files contained in this CD1 are listed in Attachment 1.

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

Contact Information
C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373 – 6466

ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-09378

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.400-3032 Rev.0", "Responses to Request for Additional Information No.402-3028 Rev.0" and "Attachment of Responses to RAI's items 11.02-20 and 11.03-12 of NRC Requests" 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 11.05-12 involves MHI's know-how, and to make these input data of RATAF code from a lot of design parameters requires knowledge and know-how about using the RATAF code and also to make the modification of the PWR-GALE code requires the knowledge about PWR-GALE code.

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 16th day of July, 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09378, Rev.0

Responses to Request for Additional Information No.400-3032
Revision 0

July 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 400-3032 REVISION 0**
SRP SECTION:	**11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems**
APPLICATION SECTION:	**11.5**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.05-12

In response to the Staff's question (RAI 249-1978, Question 11.05-5, item 1) MHI states,

> "The Containment radiation gas monitor (RMS-RE-41) will be deleted from the TS leakage detection methods since this monitor (RMS-RE-41) does not have enough leakage detection capability assuming that no failed fuel exists. The Containment radiation particulate monitor (RMS-RE-40) will remain as a diverse detection method. The DCD Chapter 16 Technical Specification will be revised to delete this monitor's description (see also RAI 165, Question 1967)."

Information requested in the Staff's question supporting the Applicant's response on the minimum required sensitivity for the containment particulate radiation monitor (RMS-RE- 40) necessary to satisfy the RCS leakage rate technical basis for detecting an increase of 1 gpm within 1 hour using a realistic primary coolant concentration was not provided.

In response to the Staff's question on the containment radiation particulate monitor MHI states,

> "This range provides the capability to detect leakage of less than 0.5 gpm within one hour of detector response time. This conforms to the requirement to detect 1 gpm as stated in RG 1.45."

Information requested in the Staff's question (RAI 249-1978, Question 11.05-5, item 2) supporting the Applicant's response on the methodology to satisfy the technical basis for RCS leakage detection instrumentation using a realistic radioactive concentration in the RCS was not provided.

The Staff requests the Applicant to:

1. Submit a detailed evaluation which demonstrates that the containment particulate radiation monitor range provides the capability to detect leakage of "less than 0.5 gpm within one hour of detector response time" using a realistic radioactive concentration in the RCS, or describe the program and procedure that will be used to satisfy the RCS leakage rate technical basis and RG 1.45 (Rev 1). Revise the DCD to include this information and provide a markup in your response.

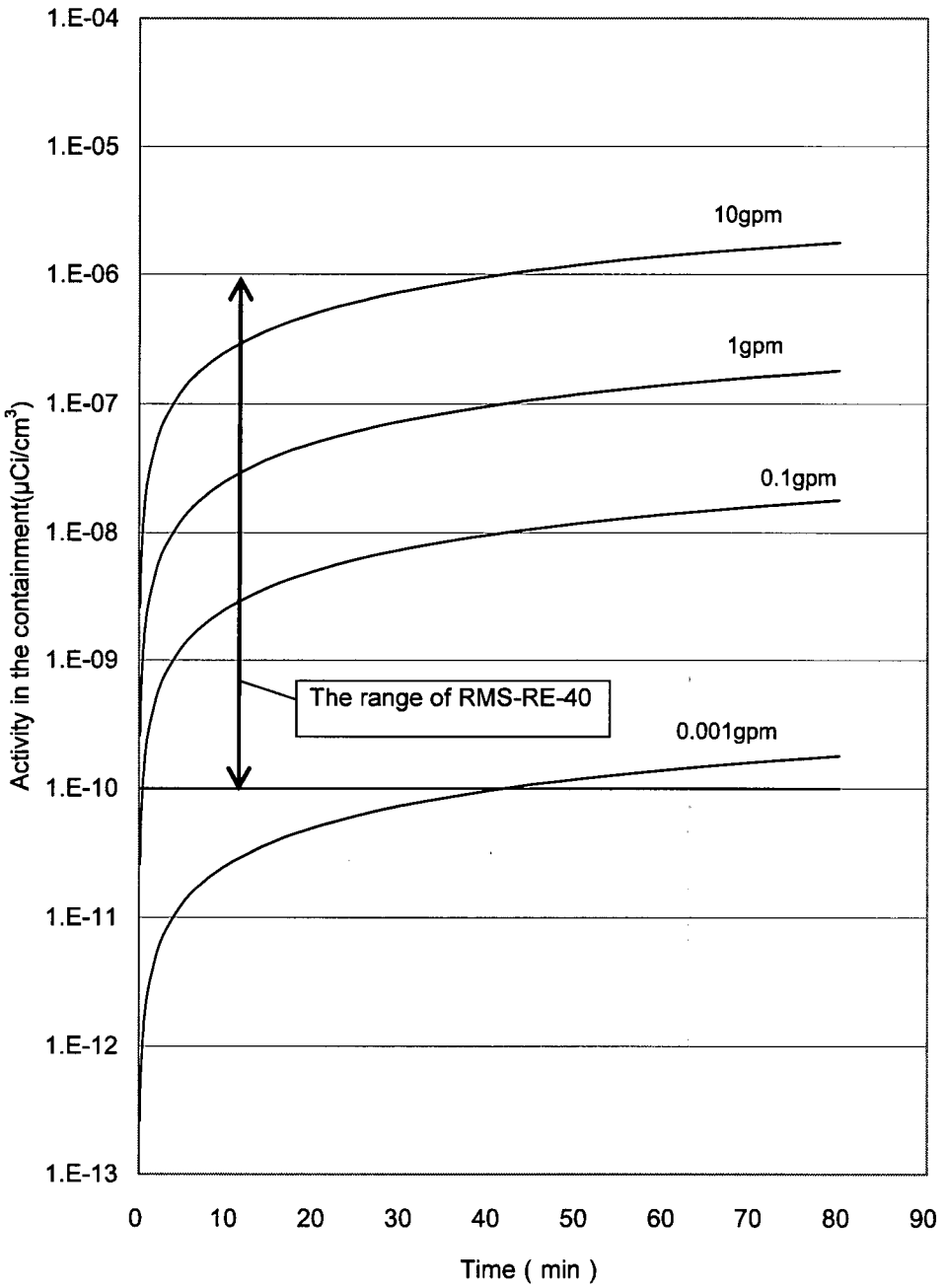
2. Update reference of RG 1.45 (Rev 0) to its current revision (Rev 1) in Table 1.9.1- 1 of DCD Section 1.9, and DCD Sections 5.2.5.4.1.2 and 5.2.7, and provide a markup in your response.

ANSWER:

1. When reactor coolant leakage into the containment has occurred, the radioactivity concentration inside the containment increases is modeled by the following differential equation (Equation 1), based on an activity balance of the RCS. The integrated form of this equation is presented in Equation 2. The radioactivity inside the containment resulting from varying leakage rates was then calculated as a function of time, based on the parameters listed in Table 1. The results of this calculation are presented in Figure 1. Activity in the containment is detected by the containment radiation particulate monitor (RMS-RE-40) which has a range of 1×10^{-10} to 1×10^{-6} $\mu Ci/cm^3$ as listed in DCD Table 11.5-1 and indicated with arrows in Figure 1. This range ensures the capability to detect leakage of less than 0.5 gpm within one hour of detector response time.

   Activity in the containment is calculated from the following equations.

Figure 1 Containment Activity



2. MHI will update the reference of RG 1.45 from Rev 0 to Rev 1 in DCD Table 1.9.1-1 and in DCD Section 5.2.7.

Impact on DCD

DCD Section 11.5.2.2.1 "Containment Radiation Monitors" will be revised to add the following information as the third paragraph:

"The containment radiation monitor used to detect leakage into the containment from the RCS has a range which is determined to have the capability of detecting less than 0.5 gpd leakage within one hour of response time. This determination is made by applying an activity balance based on the containment radioactivity concentration analysis and showing that this equation, as a function of time, represents an activity leak rate that is within the selected range for the monitor."

DCD Section 1.9.1, Table 1.9.1-1 "US-APWR Conformance with Division 1 Regulatory Guides" will be revised as follows:

1.45 Reactor Coolant Pressure Boundary Leakage Detection Systems
~~(Rev. 0, May 1973)~~ **(Rev.1, May 2008)**

DCD Section 5.2.7 "References" will be revised as follows:

5.2-15 Reactor Coolant Pressure Boundary Leakage Detection Systems, Regulatory Guide 1.45, ~~Rev.0, May 1973~~. **Rev.1, May 2008**.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 400-3032 REVISION 0**
SRP SECTION:	**11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems**
APPLICATION SECTION:	**11.5**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.05-13

In response to the Staff's question (RAI 249-1978, Question 11.05-6, item 1) it states,

> "The ranges of these three types of radiation monitors described in DCD Tables 11.5-1 through 11.5-3, respectively, are sufficient to provide the capability to detect 30 gpd primary-to-secondary leakage. This conforms to the requirement of NEI 97- 06 and EPRI Guidelines and no specific sensitivity requirement needs to be stated in DCD Tables 11.5-1 through 11.5-3."

Information requested in the Staff's question supporting the Applicant's response on the minimum required sensitivities for the steam generator blowdown water radiation monitor (RMS-RE-55), high sensitivity main steam line monitors (RMS-RE-65A, 65B, 66A, 66B, 67A, 67B, 68A, 68B), and condenser vacuum pump exhaust line radiation monitors (RMS-RE-43A, 43B) necessary to satisfy the primary-to-secondary leakage rate detection sensitivity technical basis was not provided.

In response to the Staff's question (RAI 249-1978, Question 11.05-6, item 2) MHI states,

> "Primary-to-secondary leakage is verified by these radiation monitors and compared to leakage rates calculated by using other monitors to ensure the validity of these methods."

The response does not identify and discuss the "other monitors" used to calculate primary-to-secondary leakage rates.

Information requested in the Staff's question supporting the Applicant's response on the methodology to satisfy the technical basis for primary-to-secondary leakage detection instrumentation using a realistic radioactive concentration in the RCS was not provided.

The response to the Staff's question also states,

> "The condenser vacuum pump exhaust line radiation monitors are the primary monitors used to estimate the primary-to-secondary leakage rate. The primary-to-secondary leakage rate can be estimated by comparing the fission gas activity, such as Xe-133, in the condenser exhaust gas to the fission gas activity in the reactor coolant system (RCS). When fission gas concentrations are low in the RCS, other isotopes such as Ar-41 can be used, taking into consideration the effect of their shorter half-lives."

This information on the condenser vacuum pump exhaust line radiation monitors as the "primary monitors" and how they are used to estimate primary-to-secondary leakage rate using fission gas activity in the RCS is absent in Section 11.5.2.4.2 of the DCD.

The Staff requests the Applicant to:

1. Submit a detailed evaluation which demonstrates that the ranges of the primary-to-secondary radiation monitors are sufficient to provide the capability to detect 30 gpd primary-to-secondary leakage using a realistic radioactive concentration in the RCS, or describe the program and procedure that will be used to satisfy the primary-to-secondary leakage rate technical basis, NEI 97-06 and EPRI Guidelines. Revise the DCD to include this information and provide a markup in your response.

2. Identify the "other monitors" described and discuss how they are used to calculate primary-to-secondary leakage rates to ensure the validity of these methods. Revise the DCD to include this information and provide a markup in your response.

3. Revise DCD Section 11.5.2.4.2 to include the information identifying use of the condenser vacuum pump exhaust line radiation monitors as primary monitors and how they are used to estimate primary-to-secondary leakage rate using fission gas activity in the RCS, and provide a markup in your response.

4. Identify the "other isotopes" described and discuss how these isotopes are used to estimate primary-to-secondary leakage rate in condenser exhaust gas when fission gas concentrations are low in the RCS given that Ar-41 composition in air is very small (<1%). Revise the DCD to include this information and provide a markup in your response.

ANSWER:

1. Activity in the secondary coolant rises when SG tube leakage has occurred. The condenser vacuum pump exhaust line radiation monitors (RMS-RE-43A and RMS-RE-43B) are installed to detect leakage from the RCS to the secondary coolant system. These monitors have a lower limit of their detection range of 5×10^{-8} μCi/cm^3 as listed in DCD Table 11.5-3.

The basic relationship for leak rate measurements can be described based on the condenser off-gas analysis with the following equation from Section 5.2 of the EPRI Guideline "PWR Primary-To-Secondary Leak Guidelines," Revision 3:

$$LR = \frac{\frac{A_g}{e^{-\lambda T}} \cdot F_g \cdot C}{A_{RCS}}$$

Where:

LR	: Primary-to-secondary leak rate (gpd)
A_g	: Activity concentration of noble gas radionuclide in the condenser off-gas sample (μCi/cc)
A_{RCS}	: Activity concentration of noble gas isotope in the reactor coolant (μCi/g)
F_g	: Flow rate of the condenser off-gas (SCFM)
C	:1.08 X 10^4 (gal•cc•min)/(g•ft3•day), conversion constant which includes: 60 minutes per hour 24 hours per day 28317 cc/ft3

1 gal per 3785 ml
1ml per g reactor coolant

λ :Decay constant of Ar-41 (half life is 110 min)

T :Transit time(min)

This calculation can be used for an instantaneous leak rate determination by applying the following assumptions:

1. The transport decay effects are corrected due to the long transport time considered and the short half-life of the Ar-41 used for the analysis;
2. There are no significant mother/daughter decay relationship effects;
3. RCS noble gas concentrations remain constant (i.e., no power transients, RCS degassing, etc.);
4. All of the noble gas radionuclides are instantaneously transported into the steam flow upon entering the steam generator via the leak;
5. All of the noble gas radionuclides are removed via the condenser off-gas system so that the entire noble gas isotope inventory enters the secondary system at the steam generator and exits at the condenser off-gas;
6. The condenser off-gas flow is accurately measured and accurately sampled;
7. There are no significant changes in steam or off-gas flows;
8. Plants with mechanical vacuum pumps may have high air flow rates which can reduce the sensitivity.

The parameters which describe the Reactor Coolant System leakage conditions are provided in Table 2 and the activity concentration of the radionuclide, Ar-41, in the condenser off-gas sample with a 30 gpd primary-to-secondary leakage rate is provided in Table 3. As shown in Table 3, the condenser vacuum pump exhaust line radiation monitors (RMS-RE-43A and RMS-RE-43B) have sufficient range to provide the capability to detect primary-to-secondary leakage of 30 gpd.

Table 2 Leakage Calc Parameters

Parameter	value
LR (gpd)	30
A_{RCS} (μCi/g) (1)	0.5
F_g (SCFM) (2)	72
T (min) (3)	20

Notes
(1) RCS concentration of Ar-41 as provided in DCD Subsection 11.1.1.5
(2) 3 condenser vacuum pumps, each capacity is 24 SCFM as provided in DCD Table 10.4.2-1
(3) Based on EPRI Guideline Appendix B

Table 3 Activity Results

Activity concentration of Ar-41 in the condenser off-gas sample (μCi/cc)	1.7E-5
Lower limit of the detection range for the condenser vacuum pump exhaust line radiation monitors (μCi/cc)	5.0E-8

2. The term "other monitors" refers to the steam generator blowdown water radiation monitor (RMS-RE-55) and the high sensitivity main steam line monitors (RMS-RE-65 A/B, RMS-RE-66 A/B, RMS-RE-67A/B and RMS-RE-68 A/B). When primary-to-secondary leakage occurs, radionuclides from the reactor coolant system enter the steam generator bulk water. Radiogases, as they have a very low solubility, will

quickly be transferred into the steam and can be subsequently quantified in the steam generator blowdown. Measurement of the concentration of the selected radionuclide by using the steam generator blowdown water radiation monitor can be used to estimate the primary-to-secondary leak rate by applying an activity balance around the leaking steam generator. The activity balance of the system relates the change in activity concentration in the steam generator as the difference between the activity entering (leak rate, feed water) and the activity leaving (blowdown, decay, and main steam). The method for estimating the leakage rate using the steam generator blowdown is taken from Section 5.3 of the EPRI Guideline "PWR Primary-to-Secondary Leak Guidelines," Revision 3 and can be used to verify the capability of the radiation monitors to detect primary-to-secondary leakage.

Determination of the leakage rate using the high sensitivity main steam line monitors uses a very similar method to that which was applied for the condenser off-gas method described in Item 1. The activity concentration of a radioactive noble gas (e.g. Ar-41) in the reactor coolant system, as well as the measured activity concentration of the radionuclide in the main steam line, is the primary input into the activity balance equation. The constants are adjusted for the main steam line system as described in Section 5.5 of the same EPRI Guideline. In addition to the described calculation methods, radiochemical grab sampling is used to verify the performance of the radiation monitors, alarm setpoints, and confirm leakage rate estimates.

3. Section 11.5.2.4.2 will be revised to include information on the use of the condenser vacuum pump exhaust line radiation monitors as the primary monitors used to estimate the primary-to-secondary leakage rate.

4. The other isotopes which can be used in the primary-to-secondary leak rate analysis in the application of the condenser vacuum pump exhaust line radiation monitor measurements are the noble gas isotopes due to their inert nature and are assumed to remain at a constant concentration within the RCS. Isotopes are selected for the leakage rate analysis based on having long half-lives and having a relative abundance within the RCS. The preferred isotopes to be used in this analysis, as listed by EPRI, are: Xe-133, Xe-135, Xe-135m, Kr-85m, Kr-88, Ar-41, Kr-87, and C-11. Additional isotopes which are exist in the secoundary coolant due to primary-to-secondary leakage are N-13 and F-18. However, in the provided analysis (Item 1), only Ar-41 is considered as the isotope.

Impact on DCD

DCD Section 11.5.2.4.2 "Condenser Vacuum Pump Exhaust Line Radiation Monitors" will be revised to add the following information as the third paragraph:

"The condenser vacuum pump exhaust line radiation monitors are the primary monitors used to estimate the primary-to-secondary leakage rate. The primary-to-secondary leakage rate can be estimated by comparing the fission gas activity, for an isotope such as Xe-133, in the condenser exhaust gas to the fission gas activity in the reactor coolant system (RCS). When fission gas concentrations are low in the RCS, other isotopes such as Ar-41 can be used, taking into consideration the effect of their shorter half-lives. The condenser vacuum pump exhaust line radiation monitors have a range which is determined to have the capability of detecting 30 gpd leakage. This determination is made by applying the basic relationship for leak rate measurements based on the condenser off-gas analysis as provided in Section 5.2 and Appendix B of the EPRI guideline, "PWR Primary-to-Secondary Leak Guidelines (Ref. 11.5-34)."

The following reference will be added to DCD Section 11.5.6 "References":

11.5-34 Electric Power Research Institute, "PWR Primary-to-Secondary Leak Guidelines," Rev 3, December 2004.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 400-3032 REVISION 0**
SRP SECTION:	**11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems**
APPLICATION SECTION:	**11.5**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.05-14

In response to the Staff's question (RAI 249-1978, Question 11.05-7) it states,

"The ranges of the radiation monitors described in DCD Tables 11.5-1 through 11.5-3 provide the capability to detect SG Tube leakage of an amount in conformance with NEI 97-06 and EPRI Guidelines. These three types of radiation monitors are identified in DCD Tier 1 in Table 2.7.6.6-1 and the ITAAC information is given in Table 2.7.6.6-2."

The Staff reviewed Section 2.7.6.6, and Tables 2.7.6.6-1 and 2.7.6.6-2 in Tier 1 of the DCD, but was not able to find the ITAAC to address the sensitivity, response time, and alarm limit for the SG tube leak detection instruments.

The Staff requests the Applicant to:

1. Provide the ITAAC in Tier 1 of the DCD to address the sensitivity, response time, and alarm limit of the SG tube leak detection instruments. Revise the DCD to include this information and provide a markup in your response.

ANSWER:

The Tier 1 information referenced in response to RAI No. 249-1978 Question No. 11.05-7 applies to the Process Effluent Radiation Monitoring and Sampling System and includes ITAAC to verify the as-built monitors are consistent with the functional arrangement as described in Design Description of Tier 1 Subsection 2.7.6.6 and Table 2.7.6.6-1. The Tier 1 and ITAAC level of detail is consistent with other non-Class 1E radiation monitors in the PERMS. MHI considers numeric values for sensitivity, response time and alarm limits for these instruments to be below the level of detail for DCD Tier 1. More comprehensive testing is provided in the preoperational test program (refer to the response to RAI No. 400-3032, Question No. 11.05-15). The capability to adequately measure steam generator (SG) tube leakage and maintain leakage within acceptable limits is ensured during plant operation by the requirements of Technical Specification (TS) 3.4.13, "RCS Operational Leakage" and TS 5.5.9, Steam Generator (SG) Program.

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

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 400-3032 REVISION 0**
SRP SECTION:	**11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems**
APPLICATION SECTION:	**11.5**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.05-15

In response to the Staff's question (RAI 249-1978, Question 11.05-8) it states,

"These monitors are the part of the Process and Effluent Radiological Monitoring System and their preoperational test is described in DCD Section 14.2.12.1.78 for operation.

The Staff reviewed Section 14.2.12.1.78 in Tier 2 of the DCD, but was not able to find the sensitivity, response time, and alarm limit of the SG tube leak detection instruments being included in the tests.

The Staff requests the Applicant to:

1. Provide the preoperational tests in Tier 2 of the DCD to demonstrate the sensitivity, response time, and alarm limit of the SG tube leak detection instruments. Revise the DCD to include this information and provide a markup in your response.

ANSWER:

Preoperational testing of RCS leakage detection instrumentation (including SG tube leakage detection instrumentation) is expanded and clarified in MHI's response to RAI No. 371-2617 Question No. 14.02-117 dated June 17, 2009 (MHI Ref: UAP-HF-09324). Changes to DCD Tier 2 Subsection 14.2.12.1.115, "RCPB Leak Detection Systems Test" include the addition of a cross-reference to RCS leakage detection design features to be tested as described in Subsection 14.2.12.1.78, "Process and Effluent Radiological Monitoring System, Area Radiation Monitoring System and Airborne Radioactivity Monitoring System Preoperational Test." This cross reference table includes the radiation monitors used to measure primary-to-secondary leakage. Preoperational testing of these monitors includes verification of calibration, alarm functions and alarm setpoints, whose numeric values will be specified as part of the detailed design.

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

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 400-3032 REVISION 0**
SRP SECTION:	**11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems**
APPLICATION SECTION:	**11.5**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.05-16

In response to the Staff's questions (RAI 249-1978, Question 11.05-9) to describe design features of the main steam line radiation monitors in regards to environment factors it states,

"The main steam line radiation monitor detectors and other instruments will be placed outside of the main steam lines, detecting gamma ray from radioactive nuclides which come through the wall of the main steam line pipe and the room where the monitors will be placed is ventilated. The monitors will not be affected by high temperature and humidity of main steam."

"The detectors will be shielded to minimize the effect of gamma radiation except from the steam line being monitored."

"... detectors will be shielded to minimize ambient radiation effects."

The Staff requests the Applicant to:

1. Revise the DCD to include this information and provide a markup in your response.

ANSWER:

Tier 2 DCD Section 11.5.2.2.4 will be revised to include the information provided in response to RAI 249-1978, Question 11.05-9 as requested by the NRC.

Impact on DCD

- Replace the second paragraph in Section 11.5.2.2.4 with the following:

"The monitors continuously measure the concentration of radioactive materials in the main steam line from the SG. Monitors are provided on each main steam line. **The main steam line radiation monitor detectors and other instruments will be placed outside of the main steam lines, detecting radioactive nuclides gamma ray emissions which come through the wall of the main steam line pipe. Hence, the monitors will be protected from the high temperature and humidity of main steam.** This high sensitivity main steam line (N-16ch.) monitor is used for measuring the concentration of N-16 in the main steam line during normal

operation (RMS-RE-65A, RMS-RE-65B, RMS-RE-66A, RMS-RE-66B, RMS-RE-67A, RMS-RE-67B, RMS-RE-68A and RMS-RE-68B). If the measured concentrations exceed the predetermined setpoint, an alarm is activated in the MCR. The main steam line is normally expected to be slightly radioactive. High levels of radioactive material in the line indicate a leakage of reactor coolant in a SG. Main steam line monitor measures the concentration of radioactive materials during the accident (RMS-RE-87, RMS-RE-88, RMS-RE-89, and RMS-RE-90).

- Add the following as fourth paragraph to DCD Section 11.5.2.2.4:

"The detectors will be shielded to minimize the effect of gamma radiation (due to high-energy gamma radiation on the response of main steam line monitors located near one another) and also to minimize ambient radiation effects from direct or scattered radiation during LOCA conditions regarding the response of main steam line monitors located near containment penetrations."

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 400-3032 REVISION 0**
SRP SECTION:	**11.05 – Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems**
APPLICATION SECTION:	**11.5**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.05-17

A. In response to the Staff's question (RAI 1978, Question 8274, item 2) to require a check source (or justify its exclusion) for the four main steam line accident monitors it states,

"To verify the function of the main steam line accident monitors, we will use a radiation source which indicates lower than the low limit of the measurement range. The monitoring system measures the resulting signal with output for the operators, and verifies the function of the detector by detecting loss of signal. For associated electronics, the test signal generating system and electronics will be verified by that signal."

B. In response to the Staff's question (RAI 1978, Question 8274, item 4) to provide the isotope to calibrate (or justify its exclusion) the high sensitivity main steam line (N-16 ch) monitors it states,

"detectors which will be installed at the plant are examined by another isotope which emits lower energy gamma rays and associated with the response for N-16 by analysis or type test."

The Staff requests the Applicant to:

1. Revise Table 11.5-1 to include information on how the function of the main steam line accident monitors is verified and provide a markup in your response.

2. Describe how performance monitoring checks are performed and trended in accordance with Section 4.3.3 of EPRI TR-104788-R2 (2000). Revise the DCD to include this information and provide a markup in your response.

3. Address the potential energy response dependence for detectors that will be installed when using another isotope of lower energy to calibrate the N-16 channel.

4. Provide the isotope used to calibrate the high sensitivity main steam line (N-16 ch) monitors such as the reference source or other qualified source described in Section 4.3.2.4 of EPRI TR-104788-R2 (2000). Revise DCD Table 11.5-1 to include this information and provide a markup in your response.

ANSWER:

1. MHI will revise the DCD Table 11.5-1 and provide a markup which will include information described below.
   The detector includes a radiation source which gives a detector output signal lower than the low limit of the measurement range. The output signal level of the detector is monitored in the signal processor. If the output level is lower than the set point, the signal processor will alarm to indicate failure of the monitor. The included radiation source is called a "Live zero source" and not a checked source for other monitor channels.

2. The performance monitoring checks and trending are continuously performed by the "live zero source" method as described above. MHI will revise the DCD and provide a markup which will include the information provided above.

3. NaI or CsI scintillation detector will be used (as described in EPRI TR-104788, Section 4.3.2.4) and the energy response dependence can be determined for each of these detectors. By using that dependence, the N-16 channel using lower energy isotope can be calibrated.

4. The calibration source for the detector is dependent on vendor recommendations, therefore a specific source such as Cs-137 or Co-60 etc, cannot be provided. According to the EPRI report (TR-104788, Section 4.3.2.4), "calibration can be performed with a special source or other qualified process depending on vendor recommendations." Hence, DCD Table 11.5-1 is noted "calibration source and qualification process is provided by vendor recommendation during detailed design."

Impact on DCD

Table 11.5-1 is revised as shown in the attached markup.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

Table 11.5-1 Process Gas and Particulate Monitors

Item No.	Monitor Number	Service	Type	Range µCi/cm3	Calibration Isotopes	Check Source	Safety-Related	Control Function	Quantity	Schematic Number	GA Drawing Number
1	RMS-RE-41	Containment radiation gas *The concentration of radioactive gas in the containment*	β Scint	5E-7 to 5E-3	Kr-85 Xe-133	Yes	No	Termination	1	11.5 – 1a	11.5 – 2h
2	RMS-RE-40	Containment radiation particulate *The concentration of radioactive particulate in the containment*	γ	1E-10 to 1E-6	Cs-137	Yes	No	Termination	1	11.5 – 1a	11.5 – 2h
3	RMS-RE-23	Containment low-volume purge radiation gas *The concentration of radioactive material in the duct of the Containment Depressurization Purge System*	β Scint	1E-6 to 1E-1	Kr-85 Xe-133	Yes	No	No	1	11.5 – 1b	11.5 – 2h
4	RMS-RE-22	Containment exhaust radiation gas *The concentration of radioactive material in the duct of the containment exhaust system*	β Scint	1E-6 to 1E-1	Kr-85 Xe-133	Yes	No	No	1	11.5 – 1b	11.5 – 2h
5	RMS-RE-65A RMS-RE-65B RMS-RE-66A RMS-RE-66B RMS-RE-67A RMS-RE-67B RMS-RE-68A RMS-RE-68B	High sensitivity main steam line (N-16ch.) *The concentration of nitrogen-16 in the main steam line*	γ	1E-8 to 5E-3	— (1)	Yes	No	No	8	11.5 – 1c	11.5 – 2i
6	RMS-RE-87 RMS-RE-88 RMS-RE-89 RMS-RE-90	Main steam line *The concentration of radioactive material in the main steam line (in accident)*	γ	1E-1 to 1E+3	Cs-137	No (2)	No	No	4	11.5 -1c	11.5 – 2i
7	RMS-RE-72	Gaseous radwaste discharge *Radioactive material conc. discharged from the GWMS*	γ	1E-3 to 1E+1	Cs-137	Yes	No	Termination	1	11.5 – 1d	11.5 - 2a
8	RMS-RE-84A RMS-RE-84B	Main control room outside air intake gas radiation *The concentration of radioactive gas in the MCR (in accident)*	β Scint	1E-7 to 1E-2	Kr-85 Xe-133	Yes	Yes	Termination	2	11.5 – 1e	11.5 – 2g
9	RMS-RE-85A RMS-RE-85B	Main control room outside air intake iodine radiation *The concentration of radioactive iodine in the MCR (in accident)*	γ	1E-11 to 1E-5	I-131	Yes	Yes	Termination	2	11.5 – 1e	11.5 – 2g
10	RMS-RE-83A RMS-RE-83B	Main control room outside air intake particulate radiation *The concentration of radioactive particulate in the MCR (in accident)*	γ	1E-12 to 1E-7	Cs-137	Yes	Yes	Termination	2	11.5 – 1e	11.5 – 2g
11	RMS-RE-101	TSC outside air intake gas radiation *The concentration of radioactive gas in the TSC (in accident)*	β Scint	1E-7 to 1E-2	Kr-85 Xe-133	Yes	No	Termination	1	11.5 – 1e	11.5 – 2g
12	RMS-RE-102	TSC outside air intake iodine radiation *The concentration of radioactive iodine in the TSC (in accident)*	γ	1E-11 to 1E-5	I-131	Yes	No	Termination	1	11.5 – 1e	11.5 – 2g
13	RMS-RE-100	TSC outside air intake particulate radiation *The concentration of radioactive particulate in the TSC (in accident)*	γ	1E-12 to 1E-7	Cs-137	Yes	No	Termination	1	11.5 – 1e	11.5 – 2g

Notes (1) Calibration source and qualification process is provided by vendor recommendation during detailed design

(2) The detector includes a radiation source which gives a detector output signal lower than the low limit of the measurement range. The output signal level of the detector is monitored in the signal processor. If the output level is lower than the set point, the signal processor will alarm to indicate failure of the monitor.

Enclosure 4

UAP-HF-09378, Rev.0

Responses to Request for Additional Information No.401-3031
Revision 0

July 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 401-3031 REVISION 0**
SRP SECTION:	**11.04 – Solid Waste Management System**
APPLICATION SECTION:	**11.4**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.04-18

In response to the Staff's question (RAI 185-2031, Question 11.04-1, items 1 and 2) the US-APWR design was changed since Revision 1 to use a non-porous material (i.e., epoxy coating) for lining cubicles in the SRST rooms of the SWMS instead of steel to meet the intent of 10 CFR 20.1406 and RG 4.21. Section 11.2 was also revised to use an epoxy coating to line cubicles of the LWMS instead of stainless steel (RAI 164-1925, Questions 11.02-1 and 11.02-2). MHI's response states,

"The following design features will be added to the DCD in Section 12.3.1.1.1.2.E:
- Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The cubicles are equipped with drainage system to direct any leakage and overflows to sumps with pumps to redirect flow to other tanks. The above design approach fully meets the intent of 10 CFR 20.1406 and RG 4.21. The DCD will be changed to document the additional design features."

On the use of epoxy coatings to minimize environment and groundwater contamination MHI states,

> "The cubicles are epoxy coated to ease decontamination. Further, the epoxy coating also serves to minimize the potential for contamination of groundwater in the event that a tank fails or overflows." (Revision to Section 11.4.1.2)
>
> "The cubicles are epoxy coated to minimize the potential for contamination of the groundwater system in the event that the tank fails or overflows." (Revision to Section 11.4.1.2)
>
> "Tank cubicles are epoxy coated to minimize the potential for accidental releases to the environment in accordance with BTP 11-3 (Ref. 11.4-14), 10 CFR 20.1302 (Ref. 11.4-15) and 10 CFR 20.1406 (Ref. 11.4- 16)." (Revision to Section 11.4.1.4)
>
> "The SRST rooms are epoxy coated up to the cubicle wall height equivalent to full tank volume to minimize the potential for cross contamination to the groundwater system in accordance with BTP 11-3 (Ref. 11.4-14) and 10 CFR 20.1406 (Ref. 11.4- 16). The epoxy coated approach is also beneficial for ease of decontamination and decommissioning." (Revision to Section 11.4.2.5)

and to contain leaks,

"Isolate leak and use other SRST. Repair the leak. The floor is epoxy lined to contain

the leak." (Revision to DCD Table 11.4-5)

In response to the Staff's questions (RAI 185-2031, Question 11.04-1, item 3) to provide (or justify exclusion of) ITAAC to ensure complete and acceptable construction of cubicle liners in SRST rooms of the SWMS before the design change using epoxy coatings MHI states,

> "An Initial Test Program will be utilized for these coating systems using normal construction testing practices will be utilized with qualified coating inspections per the ASTM D4537-04a "Standard Guide for Establishing Procedures to Qualify and Certify Inspection Personnel for Coating Work in Nuclear Facilities". Hence, no ITAAC is necessary."

The Staff requests the Applicant to:

1. Justify the use of epoxy coatings as an acceptable liner for SRST rooms of the SWMS to minimize contamination of the environment and groundwater (i.e., justify the capability of epoxy coatings to retain liquids given that coatings are typically applied to protect the surfaces of facilities and equipment from corrosion and contamination, and because coatings are not approved for retention of liquids per BTP 11-6).

2. Describe the maintenance and inspection program that will be implemented to ensure the integrity of epoxy coatings for sealing floor and wall surfaces to minimize contamination of the facility.

3. Clarify how guidance in BTP 11-3 is applied to epoxy coating of tank cubicles and SRST rooms in the SWMS (see response statements above).

4. Identify the described ITP on coating systems, construction practices and qualified inspections for lining cubicles in the SRST rooms of the SWMS in the DCD, and provide a markup in your response.

Revise the DCD to include this information and provide a markup in your response.

ANSWER:

1. Use of Epoxy Coatings as an Acceptable Liner for SRST Rooms

The suitability of epoxy coatings for lining tank rooms is addressed in response to RAI No. 403, Revision 0 Question 11.02-18. The potential liquid release from a spent resin storage tank (SRST) is bounded by the liquid waste management system (LWMS) tank releases evaluated as described in DCD Subsection 11.2.3.2. MHI will revise the DCD accordingly. Therefore, the response to RAI 403, question 11.02-18 also applies to the use of coatings in SRST rooms.

2. Maintenance and Inspection Program to Ensure the Integrity of Epoxy Coatings

As is the case with epoxy coatings used in the LWMS described in response to RAI No. 403, Revision 0 Question 11.02-18, the SWMS tank room coatings are Service Level II coatings per NRC Regulatory Guide (RG) 1.54 Revision 1. Therefore, MHI will revise the DCD to refer to the guidance of RG 1.54 and related ASTM standards that will be applied to the Service Level II epoxy coatings in the SRST tank rooms.

3. Applicability of BTP 11-3 to Epoxy Coatings

Branch Technical Position (BTP) 11-3 does not explicitly address the use of epoxy coatings for lining SWMS rooms. BTP 11-3 refers to NUREG-0800 Appendix 11.4-A, "Design Guidance for Temporary Storage of Low-Level Radioactive Waste," which includes design guidance generally applicable to the SWMS. This guidance referenced by BTP 11-3 includes general provisions to minimize contamination of the facility and the environment, to the extent practical, in accordance with 10 CFR 20.1406. MHI considers the use of epoxy coatings in the SRST rooms to be consistent with this guidance. MHI will revise the DCD to clarify the purpose of the coatings.

4. Initial Test Program, Construction and Qualified Inspection of Epoxy Coatings

The provisions of RG 1.54 and related ASTM standards that are described in response to RAI No. 403, Revision 0 Questions 11.02-18 and 11.02-19 are applicable to the SRST room epoxy coatings. MHI will revise the DCD to address these provisions specifically for the SWMS.

Impact on DCD

As shown in response to RAI No. 403, Revision 0 Question 11.02-18, MHI will revise the RG 1.54 position in DCD Tier 2 Table 1.9.1-1 to refer to DCD Section 11.4.

- DCD Tier 2 Subsection 11.4.1.2, "Design Criteria", fourth bullet, will be revised as follows: (The changes below replace the corresponding changes committed to in the response to RAI 185, question 11.04-1.)

  - The SRSTs are cross-connected so that the failure or maintenance of one component does not impair the system or the plant operation. Table 11.4-5 provides typical failure scenarios. The spent resin storage tanks (SRSTs) are housed in individual cubicles, each with a shield wall thickness commensurate with the projected maximum dose rate of its content. The cubicles **that contain significant quantities of radioactive material are coated with an impermeable epoxy liner (coating), up to the cubicle wall height equivalent to the full tank volume, to facilitate decontamination of the facility in the event of tank leakage and failure. This design feature, in conjunction with early leak detection, drainage and transfer capabilities, serves to minimize the release of the radioactive liquid to the groundwater and environment in accordance with the BTP 11-6 (Ref. 11.4-32) and 10 CFR 20.1406 (Ref. 11.4-16). As an additional precaution, the COL Applicant is also required to provide an environmental monitoring system (Section 11.5.5)** ~~are lined with steel to ease decontamination. Further, the steel liner also serves to minimize the potential for contamination of groundwater in the event that a tank fails or overflows.~~ Other design features addressing release requirements are described in Section 11.2

- Delete sixteenth bullet from DCD Tier 2 Subsection 11.4.1.2, "Design Criteria" as Subsection 11.4.1.2, "Design Criteria" fourth bullet contains redundant information.

  ~~- The SRSTs are housed in individual cubicles, each with a shield wall thickness commensurate with the projected maximum radioactive dose rate of its content. The cubicles are steel lined to minimize the potential for contamination of the groundwater system in the event that the tank fails or overflows. Other design features addressing the release requirements are described in Section 11.2.~~

- DCD Tier 2 Subsection 11.4.1.4, "Method of Treatment", ninth bullet, will be revised as follows: (The changes below replace the corresponding changes committed to in the response to RAI 185 Question 11.04-1.)

  - Each of the SRST cubicles is designed to contain the maximum liquid inventory in the event that the tank ruptures. **These cubicles are coated with an impermeable epoxy**

liner (coating), up to the cubicle wall height equivalent to the full tank volume, to facilitate decontamination of the facility in the event of tank leakage and failure. This design feature, in conjunction with early leak detection, drainage and transfer capabilities, serves to minimize the release of the radioactive liquid to the groundwater and environment in accordance with the BTP 11-6 (Ref. 11.4-32) and 10 CFR 20.1406 (Ref. 11.4-16). As an additional precaution, the COL Applicant is also required to provide an environmental monitoring system (Section 11.5.5) ~~Tank cubicles are steel-lined to minimize the potential for accidental releases to the environment in accordance with BTP 11-3 (Ref. 11.4-14), 10 CFR 20.1302 (Ref. 11.4-15) and 10 CFR 20.1406 (Ref. 11.4-16).~~

- DCD Tier 2 Subsection 11.4.2.5, "Operation and Personnel Doses", first paragraph, will be revised as follows: (The changes below replace the corresponding changes committed to in the response to RAI No. 185 Question 11.04-1.)

  The SRSTs are located in individually shielded cubicles in the A/B. **These cubicles that contain significant quantities of radioactive material are coated with an impermeable epoxy liner (coating), up to the cubicle wall height equivalent to the full tank volume, to facilitate decontamination of the facility in the event of tank leakage and failure. This design feature, in conjunction with early leak detection, drainage and transfer capabilities, serves to minimize the release of the radioactive liquid to the groundwater and environment in accordance with the BTP 11-6 (Ref. 11.4-32), 10 CFR 20.1302 (Ref. 11.4-15) and 10 CFR 20.1406 (Ref. 11.4-16). As an additional precaution, the COL Applicant is also required to provide an environmental monitoring system (Section 11.5.5).** ~~The SRST rooms are equipped with a steel liner up to the cubicle wall height equivalent to full tank volume to minimize the potential for cross contamination to the groundwater system in accordance with BTP 11-3 (Ref. 11.4-14) and 10 CFR 20.1406 (Ref. 11.4-16). The steel lining approach is also beneficial for ease of decontamination and decommissioning.~~ Normally, these cubicles are not occupied and the entrance is under administrative control and physical control with locked doors. Entrances are provided for inspection for ease of ingress and egress; therefore, minimizing stay time and radiation doses.

- DCD Subsection 11.4.6, "Testing and Inspection Requirements," will be revised by adding the following after the third paragraph:

  "Epoxy coatings in cubicles that contain significant quantities of radioactive material, including the SRST rooms, are Service Level II coatings as defined in RG 1.54 Revision 1, and are subject to the limited QA provisions, selection, qualification, application, testing, maintenance and inspection provisions of RG 1.54 and standards referenced therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D 4537 (Reference 11.4-31) using the inspection plan guidance of ASTM D 5163 (Reference 11.4-32)."

- DCD Subsection 11.4.9, "References," will be revised to add the following:

  11.4-31 American Society for Testing and Materials, "Standard Guide for Establishing Procedures to Qualify and Certify Inspection Personnel for Coating Work in Nuclear Facilities," ASTM D 4537-04a

11.4-32 American Society for Testing and Materials, "Standard Guide for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants," ASTM D 5163-08.

- DCD Tier 2 Table 11.4-5, "Equipment Malfunction Analysis", revise Alternate Action for Equipment Items on SWMS and Spent resin storage tank external valve leak as follows: (The changes below replace the corresponding changes committed to in the response to RAI 185, question 11.04-1.)

Equipment Item	Malfunction	Result(s)	Alternate Action
SWMS	Earthquake damage	Spent resin storage tank rupture; resin and fluid leakage.	~~Both resin and liquid are~~ **Resin is** contained in the room. **Liquid release is bounded by LWMS tank release.** Dose consequences are within the design guidance of SRP 11.4 appendix 11.4-A.
Spent resin storage tank external valve leak	Fluid leaks out	Local contamination.	Isolate leak and use other SRST. Repair the leak. The floor is **epoxy lined (coated) to facilitate decontamination in the event of valve leakage.** ~~steel lined to contain the leak.~~

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

Enclosure 6

UAP-HF-09378, Rev.0

Responses to Request for Additional Information No.402-3028
Revision 0

July 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 402-3028 REVISION 0**
SRP SECTION:	**11.03 – Gaseous Waste Management System**
APPLICATION SECTION:	**11.3**
DATE OF RAI ISSUE:	**6/18/2009**

QUESTION NO. : 11.03-12

In response to the Staff's question (RAI 164-1925, Question 11.02-7; RAI 189-2006, Question 11.03-6) to provide the basis for all values and assumptions used in the PWR-GALE code calculation of expected liquid and gaseous effluent releases MHI states, "Reactor coolant leak rate to the containment for noble gases: 0.0002/d this value is determined by the ratio of 10 gpd described in DCD Table 11.2-2 and the reactor coolant mass of 646,000 lb (along with a unit conversion)." And, "... this value is integrated in the PWR-GALE program code, the code has been modified to reflect the parameter."

The Staff requests the Applicant to:

1. Provide the "unit conversion" mentioned and its basis.

2. Justify the reactor coolant leak rate of 0.0002/d for noble gas releases from containment. Include the basis (e.g., operational data) for an expected leakage rate value of 10 gpd inside containment (to containment sump).

3. Provide information on the modified PWR-GALE code to include:
   a. An executable copy of the modified code and a printout of the source code.

   b. Identify all modifications made to the code and the specific lines of source code changed.

   c. QA/QC performed on the modified code.

   d. Other documentation to support use of the modified code.

ANSWER:

1. The unit conversion mentioned in the original response is to convert the pounds of reactor coolant into gallons (using the density of water), so that the units are consistent with the 10 gpd "Expected Input Rate" to the WHTs from leakage inside containment (as stated in DCD Table 11.2-2). The unit conversion is used as follows:

$$646{,}000lb \times \frac{1ft^3}{62.426lb}(density\ of\ water) \times \frac{7.48gal}{1ft^3} = 77405gal\ of\ reactor\ coolant$$

Therefore the reactor coolant leak to the containment for noble gases:

$$\frac{10gpd}{77405gal} = .00013 \approx .0002/d$$

2. The 0.0002/d value is calculated as shown above. It is based on 10 gpd leakage inside containment, from DCD Table 11.2-2. This value is based on ANSI 55.6-1993, Table 7, which gives a 10 gpd value for leakage inside containment.

3. a. MHI will send the executable copy separately.

b. The modifications are shown in the attached tables. MHI made modifications to PWR-GALE code as follows.

- Reflecting the source term values described in ANSI/ANS-18.1-1999.
- Changing the Reactor Coolant leak rate to the containment for noble gases to 0.0002.

c. Regarding the modification to the PWR-GALE code, MHI conducted QA/QC activities in line with MHI's QA manuals, documented the QA activities as follows (These documents are written in Japanese).

1. Computer software validation and installation plan
2. Computer software validation and installation report
3. User document
4. Configuration control
5. In-use check

d. There is the document which shows MHI's interpretation of the algorithm of the PWR-GALE code as MHI internal document (This document is written in Japanese).

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

This completes MHI's response to the NRC's question.

The modifications to PWR-GALE Code

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 402-3028 REVISION 0**
SRP SECTION:	**11.03 – Gaseous Waste Management System**
APPLICATION SECTION:	**11.3**
DATE OF RAI ISSUE:	**6/18/2009**

QUESTION NO. : 11.03-13

Several inputs to the PWR-GALE code in regards to treatment provided for removal of airborne iodine and radioactive particulates from gaseous effluents in ventilation exhaust are identified as "None" or "0".

The Staff requests the Applicant to:

1. Justify the exclusion of the following items in the design:
    a. Gas waste system (no HEPA filter installed)
    b. Auxiliary building (no HEPA and charcoal filter installed)
    c. High volume purge exhaust system (no charcoal filter installed)
    d. Containment atmosphere internal cleanup system and filtration

ANSWER:

The exclusion of the filters listed above from the PWR-GALE code inputs (in DCD Table 11.3-5) is correct, as these filters are not included in the MHI design of the listed systems. The average and maximum gaseous releases calculated by the PWR-GALE code are given in DCD Table 11.3-5 (Sheets 1 through 6). These values are used in the GASPAR II code to calculate annual doses, shown in Table 11.3-8. These doses are less than the numerical ALARA guidance given in 10 CFR 50 Appendix I (as discussed in DCD Section 11.3.3.1). Therefore, the MHI design meets the release objectives without the listed features included, and their exclusion is justified.

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

This completes MHI's response to the NRC's question.

Enclosure 7

UAP-HF-09378, Rev.0

Responses to Request for Additional Information No.403-3027
Revision 0

July 2009
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 403-3027 REVISION 0**
SRP SECTION:	**11.02 – Liquid Waste Management System**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.02-18

In response to the Staff's question (RAI 164-1925, Question 11.02-1) the US-APWR design was changed since Revision 1 to use a non-porous material (i.e., epoxy coating) for lining LWMS cubicles instead of stainless steel to meet the intent of 10 CFR 20.1406 and RG 4.21. Unlike stainless-steel liners, coatings are not approved by the NRC as a design feature for retention of liquids for compliance with 10 CFR 20.1302 and conformance with SRP 2.4.13 and 11.2.3, and BTP 11-6.

The response states that tanks in LWMS cubicles containing significant amounts of radioactive fluid are,

"lined with epoxy up to a wall height sufficient to contain the entire tank contents. This epoxy will serve as a barrier to minimize contamination of the facility, environment, and groundwater, from any leaks from the equipment."

Coatings are also to be applied to all areas inside the A/B including the floor under pumps, and to decontaminable paints and smooth-surface coatings of concrete floors and walls.

Appendix A of RG 4.21 lists design features considered for compliance with 10 CFR 20.1406 such as impermeable, durable, and readily cleanable floor liners and catch basins (A-1, item e), material selection for SSCs such as liners for storage and transport of radioactive liquids (A-1, item s), appropriate sealers and a maintenance and inspection program for seal integrity over facility life (A-1, item t), and solidly constructed catch basins that are sealed, leak proof, and sufficiently larger capacity to hold entire tank contents (A-2, item c).

The Staff requests the Applicant to:

1. Justify the use of epoxy coatings as an acceptable liner for LWMS cubicles to minimize contamination of the environment and groundwater (i.e., justify the capability of epoxy coatings to retain liquids given that coatings are typically applied to protect the surfaces of facilities and equipment from corrosion and contamination, and because coatings are not approved for retention of liquids per BTP 11-6).

2. Describe the maintenance and inspection program that will be implemented to ensure the integrity of epoxy coatings for sealing floor and wall surfaces to minimize contamination of the facility.

Revise the DCD to include this information and provide a markup in your response.

ANSWER:

1. Use of Epoxy Coatings as an Acceptable Liner for LWMS Cubicles

As described in DCD Subsection 11.2.3.2, the radioactive effluent releases due to the postulated failure of a large tank containing radioactive liquid is postulated per NRC Branch Technical Position BTP 11-6 with complete release of the tank contents with results in 10CFR20.1301-acceptable doses for this event at the potable water supply in the unrestricted area, without any credit for liquid retention by liners or coatings. Although BTP 11-6 does not allow credit for retention by coatings or leakage barriers outside the building foundation, NRC regulations and guidance do not preclude the use of epoxy coatings to serve as a barrier to minimize contamination of a facility, the environment, and groundwater from any equipment leaks, such as the liquid waste management system (LWMS) and solid waste management system (SWMS) cubicles/rooms (refer to the response to RAI 401-3031, question 11.04-18 for additional information specific to the SWMS).

Epoxy coating has long been used in the nuclear industry as a design feature to minimize contamination. It provides a seal that is impermeable, durable, and with a surface that is readily cleanable and repairable, when necessary. Throughout its development and use from prior to the 1970's to the present, epoxy coating materials have been tested by manufacturers (such as Keeler & Long, Progressive Epoxy Polymers, Inc., etc.) and independent institutes (such as Franklin Institute of Philadelphia, Oak Ridge National Laboratory, etc.) and shown to be suitable for a range of nuclear applications. During this period, NRC, EPRI, ANSI and ASTM developed standards and regulatory guidance (e.g., Regulatory Guide 1.54, current Revision 1 dated July 2000, and its original issue in July 1973) to establish the minimum requirements and acceptance criteria for protective coating use in the nuclear industry. Throughout this period, the specifications for use and procedures for inspection and maintenance have been practiced and refined, and the performance of epoxy coatings has been proven and well documented. The majority of the nuclear power plants in the US, and those overseas, use epoxy coatings for equipment areas both inside the containment and outside the containment. Most nuclear power plants also developed engineering program manuals to guide its engineering activities required to implement the protective coatings program, and of outside organizations contracted to perform engineering activities within the scope of the protective coatings program. In addition, station procedures are developed for performance of maintenance, for certification and qualification of coating/lining applicators, and for certification and qualification of personnel performing inspection of coating/lining work.

Most plants also developed environmental monitoring programs for groundwater monitoring to assure public health and safety from any unexpected leakage in the event of any loss of the continued functionality of such coatings. The US-APWR design also has leak detection features in the LWMS for early detection of leakage, and has the flexibility to transfer tank contents from one tank to another. In the case of sudden tank failure, the tank content is drained to a sump that can forward the liquid to another tank for staging and processing. Hence, the US-APWR design is justified in using epoxy coating for LWMS cubicles as part of the defense-in-depth approach to minimize contamination of the facility, the environment and the groundwater.

The US-APWR design includes criteria for determining which tank cubicles require contamination minimization. The use of epoxy coating in these tank cubicles is based on operating experience regarding its efficacy in this application. This experience combined with the large body of epoxy suitability test results in similar applications will be utilized for epoxy selection and utilization. Using a defense-in-depth approach to minimize contamination of the environment and groundwater, including the use of epoxy coatings, the first defense is the tank and associated component itself. Secondly in the event of the leak each cubicle has drainage to enable detection of any of this leakage and provide for its removal. Finally the epoxy coatings are utilized to retard such potential leakage. Following table contains vendor data regarding the vapor permeability of epoxy coatings, including vapor permeability of three thicknesses of typical epoxy coatings. The table also includes data for urethane enamel designed to allow moisture to pass through. As shown, the three typical epoxies have very low moisture vapor permeability.

Despite this, no credit is taken for retention by epoxy coatings, and no credit is taken for the expectation that any leakage that does permeate through epoxy coatings would be significantly inhibited from transport to the groundwater by the concrete walls adjacent to the epoxy.

Based on the above, MHI considers that the use of epoxy coatings to line the LWMS cubicles is appropriate for the application, as part of a defense-in-depth approach to minimize contamination of the facility, environment and groundwater.

Table.1 ASTM D1653 Moisture Vapor Permeability of Organic Coating Films

Coating	DFT	Specific Permeability	
		$g/m^2/mil/24hrs$	$g/m^2/mm/24hrs$
No. 5500 Kolor-poxy self leveling floor coating	40 mils	0.6	2.2
No. 3500 Kolor-poxy self-priming surf, enamel	16 mils	2.7	10.5
No. 3200 Kolor-poxy white primer	9 mils	2.2	8.7
N-series Neothane Enamel	7 mils	18.0	70.9

2. Maintenance and Inspection Program to Ensure the Integrity of Epoxy Coatings

The epoxy coatings in LWMS cubicles are not credited with performing a safety function. Their failure would not prevent normal operating performance of any structures, systems or components (SSCs), nor adversely affect the safety function of any SSC. As defined in NRC Regulatory Guide (RG) 1.54 revision 1, "Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants," Service Level I coatings are used inside the reactor containment and Service Level III coatings' failure could adversely affect the safety function of a safety-related SSC. Therefore, of the three service levels defined in RG 1.54, Service Level II is considered applicable to epoxy coatings in LWMS cubicles:

> "Service Level II coatings are used in areas where coatings failure could impair, but not prevent, normal operating performance. The functions of Service Level II coatings are to provide corrosion protection and decontaminability in those areas outside the reactor containment that are subject to radiation exposure and radionuclide contamination. Service Level II coatings are not safety-related."

The maintenance and inspection program that will be implemented to ensure the integrity of epoxy coatings for sealing floor and wall surfaces to minimize contamination of the facility and the environment will be developed based on similar programs typically used in the nuclear industry. MHI will apply the guidance of RG 1.54 Revision 1 to maintenance and inspection of the epoxy coatings in the LWMS cubicles, in combination with accessibility and ALARA dose considerations for areas of inspection.

ASTM D 5144, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants," is the top-level standard for coatings in nuclear plant applications. ASTM D 5144-00 and selected standards referenced therein are endorsed by the NRC via RG 1.54, subject to use of the RG 1.54 Service Level definitions, and implementation of any additional commitments in a licensee's QA program description. The ASTM standards are not particularly prescriptive for Service Level II coatings, due to their relatively low safety significance. However, RG 1.54 endorses the use of applicable provisions of the following standards for Service Level II coatings:

- ASTM D 3843-00, "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities," may be used as the basis for limited QA provisions for Service Level II protective coatings.

- ASTM D 4082-95, "Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Light-Water Nuclear Power Plants," for evaluating the effects of gamma radiation on the lifetime radiation tolerance of Service Level I and II coatings.

- ASTM D 5163, "Standard Guide for Establishing Procedures To Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant" (now entitled "Standard Guide for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants) for establishing an in-service coatings monitoring program as applicable to Service Level II coatings.

Therefore, MHI will revise the DCD to establish upper tier maintenance and inspection criteria for the Service Level II coatings in LWMS cubicles, consistent with RG 1.54 Revision 1.

Impact on DCD

MHI will revise DCD Tier 2 Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 4 of 15) as follows:

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	Conformance with exceptions. Programmatic/operational and site-specific aspects are not applicable to US-APWR design certification. **ASTM standard revision levels may differ from RG 1.54 as specifically referenced in the "Corresponding Chapter/Section/Subsection"**	6.1.2 **11.2** **11.4**

Impact on DCD (continued)

- DCD Tier 2 Subsection 11.2.1.2, "Design Criteria", sixth bullet, will be revised as follows:

  - The waste collection and monitor tanks are provided with an overflow connection at least as large as the inlet. The location of the overflow is above the high-level alarm setpoint. Each cell housing these tanks is **coated with an impermeable epoxy liner (coating), up to the cubicle wall height equivalent to the full tank volume, to facilitate decontamination of the facility in the event of tank leakage and failure. This design feature, in conjunction with early leak detection, drainage and transfer capabilities, serves to minimize the release of the radioactive liquid to the groundwater and environment in accordance with the BTP 11-6 (Ref. 11.2-17) and 10 CFR 20.1406 (Ref. 11.2-7).** ~~designed to contain the contents of the tank in the event that the tank ruptures.~~

- DCD Tier 2 Subsection 11.2.2.2.2, "Tanks," second paragraph, will be revised as follows (changes in response to RAI 164, question 11.02-1 and RAI 403, question 19 are included for clarity):

  The tanks are equipped with overflows (at least as large as the largest inlet) into the appropriate sumps. The cells/cubicles housing tanks that contain significant quantities of radioactive material are ~~lined~~ **coated** with ~~stainless steel~~ **epoxy** to a height that is sufficient to hold the tank contents in the event of tank failure. **These coatings are Service Level II as defined in RG 1.54 Revision 1, and are subject to the limited QA provisions, selection, qualification, application, testing, maintenance and inspection provisions of RG 1.54 and standards referenced therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D 4537 (Reference 11.2-22) using the inspection plan guidance of ASTM D 5163 (Reference 11.2-23).** Level-detecting instrumentation measuring the current tank inventories is provided. High- and low-level alarms are provided. These alarms are annunciated in the radwaste control room located in the A/B and also in the MCR.

- DCD Tier 2 Subsection 11.2.1.4, "Method of Treatment", fifth paragraph, will be revised as follows (changes in response to RAI 164, question 11.02-1 is included for clarity):

  Filters, the activated charcoal filter, and ion exchange columns are designed with remote handling capabilities such that contact maintenance is not required. Component connections are butt welded to minimize leakage. Tanks are equipped with high-level alarms which either shut off the feed pumps or alert operators to re-direct the flow to other storage tanks to minimize the potential for overflow. In addition, **cubicles that contain significant quantities of radioactive material are coated with an impermeable epoxy liner (coating), up to the cubicle wall height equivalent to the full tank volume, to facilitate decontamination of the facility in the event of tank leakage and failure. This design feature, in conjunction with early leak detection, drainage and transfer capabilities, serves to minimize the release of the radioactive liquid to the groundwater and environment in accordance with the BTP 11-6 (Ref. 11.2-17) and 10 CFR 20.1406 (Ref. 11.2-7). As an additional precaution, the COL Applicant is also required to provide an environmental monitoring system (Section 11.5.5).** ~~where the radioactive liquid is stored are curbed and lined up to a wall height equivalent to one full tank volume of liquid for that tank. This liner system acts as a barrier to minimize the contamination of the groundwater system, and to ease decontamination in the event of an overflow or break.~~ Overflow from tanks or standpipe is directed to a near-by sump. The sump has liquid level detection. At high liquid levels, the level switch automatically activates the sump pump to forward the liquid to the WHT for processing. This design minimizes the potential for contamination of the facility and the environment, facilitates decommissioning, and minimizes the generation of radioactive waste.

- The following references will be added to DCD Tier 2 Subsection 11.2.5:

11.2-22 American Society for Testing and Materials, "Standard Guide for Establishing Procedures to Qualify and Certify Inspection Personnel for Coating Work in Nuclear Facilities," ASTM D 4537-04a

11.2-23 American Society for Testing and Materials, "Standard Guide for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants," ASTM D 5163-08.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 403-3027 REVISION 0**
SRP SECTION:	**11.02 – Liquid Waste Management System**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.02-19

In response to the Staff's questions (RAI 164-1925, Question 11.02-2, item 2) to provide (or justify exclusion of) ITAAC to ensure complete and acceptable construction of stainless-steel liners for LWMS cubicles before the design change using epoxy coatings MHI states,

"With respect to the Initial Test Program for these coating systems, normal construction testing practices will be utilized with qualified coating inspections per the ASTM D4537-04a "Standard Guide for Establishing Procedures to Qualify and Certify Inspection Personnel for Coating Work in Nuclear Facilities". Hence, no ITAAC is necessary."

The Staff requests the Applicant to:

1. Identify the described ITP on coating systems, construction practices and qualified inspections for LWMS cubicles in the DCD, and provide a markup in your response.

ANSWER:

The Initial Test Program on coating systems, construction practices and inspections for LWMS (and SWMS) cubicles is based on inspection programs using ASTM D 5163, "Standard Guide for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants," typically used in the nuclear industry for Service Level I coatings, as presented below:

During post-construction walk-through, visually examine coated surfaces for any visible defects, such as blistering, cracking, flaking / peeling, rusting, and physical damage, as follows, with the acceptance criteria indicated.

Blistering — Compare any blistering found to the blistering pictorial standards for coatings defects (refer to Test Method ASTM D 714) and record size and frequency. If the blisters are larger than those on the comparison photographs, measure, record size and extent, and photograph. Report if blistered portions are intact. Report if there is blister fluid. If fluid is present, samples may be taken.

Cracking—Cracking can be limited to the one layer of a lining system or can extend through to the substrate. Measure the length of the crack or, if extensive cracking has occurred, measure the size of the area affected. Determine if the cracking is isolated or is part of a

pattern. Record measurements and describe crack depth and pattern on the inspection report. Photograph the area affected.

Flaking/Peeling/Delamination—Measure the approximate size of the peels and note the pattern formed. Carefully test to see if lifting can be easily achieved beyond the obvious peeled area. Note all observations on the inspection report and photograph the area affected.

Rusting—Compare with the pictorial standards such as Test Method ASTM D 610 to determine the degree of rusting. Try to determine the source of rusting (that is, is it surface stain caused by rusting elsewhere or is it a failure of the lining allowing the substrate to rust). Photograph the affected area and record observations on the inspection report.

If no defects are found, indicate that the lining is intact and has no defects on the inspection report.

If portions of the lining cannot be inspected, note the specific areas on the location map-inspection report, along with the reason why the inspection cannot be conducted.

Written or photographic documentation, or both, of lining inspection areas, failures, and defects shall be made and the process of documentation standardized by the facility owner/operator. Written documentation practice for inspection of a lining system may be used for guidance.

For lining surfaces determined to be suspect, deficient, or degraded, one or more physical tests, such as dry film thickness (Test Methods ASTM D 1186, D 1400, and SSPC-PA2) and adhesion (Test Methods ASTM D 3359, D 6677, and D 4541), may be performed when directed by the evaluator. Samples may be gathered, and the size and extent of defective patterns may be described. To the extent that such testing is intrusive, lining repair criteria should be defined.

MHI will revise the DCD to refer to the ASTM D 5163 inspection plan and ASTM D 4537 inspector qualifications as being applicable to the LWMS cubicle epoxy coatings.

Impact on DCD

Refer to the response to RAI No. 403, Revision 0 Question No. 11.02-18 for the DCD changes.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/15/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 403-3027 REVISION 0**
SRP SECTION:	**11.02 – Liquid Waste Management System**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**06/18/2009**

QUESTION NO. : 11.02-20

In response to the Staff's question (RAI 164-1925, Question 11.02-6, item 1) to include in Section 11.2.3.2, the Tc-99 and 1-129 concentrations in the tank failure analysis, or justify their exclusion in an evaluation which considers the environmental (fate and transport) characteristics of Tc-99, 1-129, and Cs-137, MHI states that the contribution of Tc-99 and I-129 can be neglected because the same hydrological travel speed and time is used for Cs-137 which "conservatively neglects the adsorption effect by the soil."

In response to the Staff's question (RAI 164-1925, Question 11.02-6, item 2) to fully describe in Section 11.2.3, the approach used to demonstrate that liquid radioactive effluents processed by the LWMS released into the surface or groundwater from an assumed tank failure comply with the radionuclide concentrations in 10 CFR 20, Appendix B, Table 2, Column 2 (under the unity rule) and TEDE of 50 mrem/yr MHI states,

"the RATAF computer code for pressurized water reactors that is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants" is used for the evaluation."

In response to the Staff's question (RAI 164-1925, Question 11.02-6, item 2(d)) on the equipment malfunction analysis in Section 11.2 Table 11.2-18 (Sheets 1 and 2) MHI states,

"It is not necessary to describe the event stated in Table 11.2-18 since the assumption in subsection 11.2.3.2 provides greater release impact than the events stated in Table 11.2-18." The Staff was not able to find a discussion of this table in Section 11.2 of the DCD.

The Staff requests the Applicant to:

1. Clarify what is meant by "conservatively neglects the adsorption effect by the soil" because a conservative approach taken in the tank failure analysis would assume no dilution by groundwater and no credit for retardation or suspension in subsurface media. Revise the DCD to include this information and provide a markup in your response.

2. Justify the use of RATAF (NUREG-0133) based on GALE (1975) for the failed tank evaluation which predates both the source term specification in PWR-GALE (NUREG-0017) used to

calculate effluent releases in Section 11.2 and ANSI/ANSN18.1-1999 used to develop source terms in Section 11.1.

3. Revise the DCD to include the information in the Applicant's response (e.g., approach and methodology used to demonstrate compliance with the regulations and conformance with SRP 2.4.13 and 11.2.3 and BTP 11-6, tank inventories, etc.) and provide a markup in your response.

4. Submit the RATAF code input/output files used to calculate the failed tank inventories and concentrations.

5. Discuss the information presented in Table 11.2-18, which is absent from Section 11.2. Revise the DCD to include this information and provide a markup in your response.

ANSWER:

1. The calculation model used in the analysis of liquid releases from postulated tank failures was based on the entire and unmitigated tank content directly released to the groundwater system, then mixing and moving with the groundwater system. In this analysis, the model assumed the tank content is diluted with only a body of water in the vicinity of ponds surrounding the site. No other water (such as other discharges and groundwater) is credited as dilution water, and no credit is taken for retardation or suspension of radionuclide in the subsurface media. Hence the conservative assumption that the radionuclides are not filtered (or reduced) by the soil is used. In addition, groundwater transport and soil properties are site-specific parameters. The COL Applicant is responsible for assessment of this model [COLA Item # 11.2(3)] using the site specific parameters to evaluate conservativeness of this analysis.

   The DCD is revised to clarify the conservativeness of this assumption.

2. BTP 11-6 says "The radionuclide inventory in failed components is calculated based on the methods given in Chapter 4 and Appendices A and B of NUREG-0133, or by using equivalently document techniques." Appendix A of NUREG-0133 describes the RATAF code for PWR plants. Accordingly, MHI used the RATAF code.

   As shown below, for dominant nuclides (Cs-137, Cs-134 and H-3), the reactor coolant activities calculated by the RATAF code are higher than (or equal to) the realistic source terms described in DCD Chap.11.1. The realistic source terms are calculated using the methods and parameters in ANSI/ANS-18.1-1999 with the PWR-GALE code. Consequently, using RATAF code is conservative.

Table-1 Comparison of the Reactor Coolant Activity

Nuclide	RATAF Output(1) (μCi/ml)	Realistic Source Term(2) (μCi/ml)
Cs-134	1.4E-02	2.1E-05
Cs-137	1.0E-02	3.0E-05
H-3	1.0E+00	1.0E+00

Note:
1. The results in the RATAF output are based on 1% fuel defects for the reference activity in the RATAF code, and the result is multiplied by a factor of 0.12. Thus, the RATAF output is made for a fuel defect of 0.12% as per BTP 11-6 (Except Tritium).
2. The values from DCD Table 11.1-9 and the units are converted under the assumption that 1g = 1ml.

3. A paragraph reflecting the above discussion is added to DCD subsection 11.2.3.2 to address this answer.

4. MHI is submitting the requested inputs for RATAF code in a separate CD with this submittal. Please note that the inputs contain business sensitive information which is expected to be handled as such.

5. DCD subsection 11.2.3.2 evaluates the impact of radioactive effluent releases due to tank failure, whereas Table 11.2-18 evaluates the failure of sumps, sump pumps, and drainage equipment. Since these equipment items contain much smaller amount of liquid waste, the release impacts due to these equipment failures are minimal and are bounded by the impact of tank failure evaluated in subsection 11.2.3.2.

   DCD subsection 11.2.3.2 is being revised with a paragraph reflecting the above discussion.

Impact on DCD

Replace the second paragraph in DCD Tier 2, Section 11.2.3.2 with the following:

"In the evaluation, the holdup tank, the waste holdup tank and boric acid tank are selected because they contain a large amount of radioactivity. **The calculation model was based on the entire tank content directly released unmitigated to the groundwater system, then mixing and moving with the groundwater system.** It is assumed that the released liquid is diluted with 4.4E+10 gallons of water until it reaches to the location of the potable water supply. This parameter is based on the conditions of actual sites. **The model assumed the tank content is diluted with only this body of water in the vicinity of ponds surrounding the site. No other water (such as other discharges and groundwater) is credited as dilution water, and no credit is taken for retardation or suspension of radionuclide in the subsurface media. Hence the conservative assumption that the radionuclides are not filtered (or reduced) by the soil is used. In addition, groundwater transport and soil properties are site-specific parameters. The COL Applicant is responsible for assessment of this model [COLA Item # 11.2(3)] using the site specific parameters to evaluate the conservativeness of this analysis.** In addition, the traveling time is assumed to be 365 days in order to cover the transfer rate of several radionuclides. Table 11.2-16 shows the evaluation conditions applied to each tank. The fuel defect level is set to 0.12% of the core thermal power, which is based on Branch Technical Position (BTP) 11-6 (Ref 11.2-17)."

Add the following as the third paragraph to DCD Tier 2 Section 11.2.3.2:

"Branch Technical Position (BTP) 11-6 (Ref 11.2-17) subsection B.2, endorses Appendix A of NUREG-0133, which describes the RATAF code for PWR plants. Accordingly, the RATAF code is utilized. For the dominant nuclides Cs-134 and Cs-137, the reactor coolant activities calculated by the RATAF code are higher than the realistic source terms as described in Table 11.1-9 (i.e., 1.4E-02 μCi/ml vs. 2.1E-05 μCi/g in Table 11.1-9 and 1.0E-02 μCi/ml vs. 3.0E-05 μCi/g in Table 11.1-9 for Cs-134 and Cs-137 respectively, with the conversion of 1g=1ml) and equal for tritium (H-3)."

Add the following as the fourth paragraph to DCD Tier 2 Section 11.2.3.2:

"This evaluation is limited to the impact of radioactive effluent releases due to tank failure. Table 11.2-18 evaluates the failure of sumps, sump pumps, and drainage equipment. Since the Table 11.2-18 equipment items contain much smaller amount of liquid waste, the release impacts due to these equipment failures are minimal and are bounded by the impact of tank failure evaluated herein."

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

This completes MHI's response to the NRC's question.

ATTACHMENT 1

FILES CONTAINED IN CD 1

CD 1: "Attachment of Responses to RAI's items 11.02-20 and 11.03-12 of NRC Requests"
–Proprietary information

Contents of CD

File Name	Size	Sensitivity level
• RATAF Input:		
- RATAF_INP_WHT_1Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_VCT_1Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_RWSAT_0Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_RB_Sump_Tank_1Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_PMT_0Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_HT_1Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_CDT_1Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_BAT_1Y_R1.DAT (txt format)	2KB	Proprietary
- RATAF_INP_BAEVAPO_1Y.DAT (txt format)	2KB	Proprietary
- RATAF_INP_AB_Sump_Tank_1Y.DAT (txt format)	2KB	Proprietary
• RATAF output:		
- RATAF_INP_WHT_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_VCT_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_RWSAT_0Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_RB_Sump_Tank_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_PMT_0Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_HT_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_CDT_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_BAT_1Y_R1.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_BAEVAPO_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
- RATAF_INP_AB_Sump_Tank_1Y.DAT.outlist (txt format)	0.10MB	Proprietary
• PWR-GALE executable copy:		
- pgalegs_1999V2.exe (binary format)	0.50MB	Proprietary
- pgalelq_1999V2.exe (binary format)	0.58MB	Proprietary
• PWR-GALE source code:		
- pgalegs_1999V2.f (txt format)	0.03MB	Proprietary
- pgalelq_1999V2.f (txt format)	0.14MB	Proprietary