MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

February 18, 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-09056

Subject: **MHI's Responses to US-APWR DCD RAI 164-1925**

Reference: 1) "Request for Additional Information No.164-1925 Revision 0, SRP Section: 11.02 – Liquid Waste Management System, Application Section: 11.2" dates January 23, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Request for Additional Information No.164-1925 Revision 0."

Enclosed is the response to the RAI contained within Reference 1.

As indicated in the enclosed materials, the attachment data of this document (Enclosure 3) 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.

This letter includes the non-proprietary document (Enclosure 2), proprietary digital data (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 3 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.

Sincerely,

Y. Ogata

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

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Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.164-1925 Revision 0 (non-proprietary)

3. CD1:"Attachment of Responses to RAI's item 11.02-4 and 11.02-7 of NRC Requests"

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

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

Contact Information

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-09056

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 "Attachment of Responses to RAI's item 11.02-4 and 11.02-7 of NRC Requests" and have determined that the attachment data contain proprietary information that should be withheld from public disclosure.

3. The information in the 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 is that to make these data from a lot of design parameters requires knowledge and know-how about using the LADTAPII and PWR-GALE codes.

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 18th day of February, 2009.

Y. Ogata

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

Enclosure 2

UAP-HF-09056

Responses to Request for Additional Information No.164-1925
Revision 0

February 2009
(Non Proprietary)

02/18/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 164-1925 REVISION 0**
SRP SECTION:	**11.02 - LIQUID WASTE MANAGEMENT SYSTEM**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**1/23/2009**

QUESTION NO.: 11.02-1

Staff review of DCD Tier 2 (Rev 1), Section 11.2 indicates insufficient information is provided in regards to the stainless-steel liner design in cells/cubicles and housed tanks, equipment, and pumps of the LWMS for compliance with 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 20.1406; 10 CFR 20, Appendix B, Table 2, Column 2; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix I; and 10 CFR 50, Appendix A, GDC 60 and GDC 61. Section 11.2.1.4 states, "cubicles where the radioactive liquid is stored are curbed and are lined up to a wall height equivalent to one full tank volume of liquid for that tank." The liner system of these cubicles is described to serve as a barrier to minimize contamination of the facility, environment, and groundwater. Further, Section 11.2.2.2.2 states, "cells/cubicles housing tanks that contain significant quantities of radioactive material are lined with stainless steel to a height sufficient to hold the tank contents in the event of tank failure." Although Tables 11.2-2 and 11.2-3 present some design information on tanks and sources having radioactive liquid waste inflow into the LWMS, the actual tanks, equipment, and pumps associated with stainless-steel liners are not identified. Please address the following items and revise the DCD to include this information.

1. Define "significant" quantities of radioactive materials for housing tanks in lined curbed cells/cubicles.
2. Identify all tanks used for storing and processing radioactive liquids that are housed in lined curbed cells/cubicles. Justify any tanks not housed in lined curbed cells cubicles.
3. Identify all equipment and pumps from Section 11.2.2.1 used for storing and processing radioactive liquids that are housed in lined curbed cells/cubicles. Justify any equipment and pumps not housed in lined curbed cells/cubicles.
4. Provide information on the design and radioactive material inventory of the boric acid tank described in Section 11.2.3.2 as a tank that contains a "large amount of radioactivity."

ANSWER:

Since the DCD Rev. 1 was issued, the US-APWR design has changed to use an epoxy coating to line cubicles, instead of stainless steel. This design is in accordance with Regulatory Guide 4.21, and the changes were described to the NRC in RAI No. 91-1496, Revision 1, Question No 12.03-12.04-2:

"Additions to DCD:
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."

The non-porous material described above is an epoxy coating with design in accordance with RG 4.21 (for example, forming a seal that is impermeable, durable, and with readily cleanable surfaces that facilitate decontamination). The DCD will be updated to reflect this design change as shown in "Impact on DCD" below.

1. The basis for "significant" quantities of radioactive material is taken from 10 CFR 20 Appendix B Table 2. The effluent concentrations due to liquid containing tank failures are shown in Table 11.2-17. These concentrations are below the limits of 10 CFR 20 Appendix B Table 2, However, to be conservative, they are considered to be significant.

2. The tank cubicles in the Liquid Waste Management System containing significant amount of radioactive fluid as listed below are curbed and coated with an 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. The tanks are:
   - Waste holdup tanks
   - Waste monitor tanks
   - A/B Sump Tank
   - A/B Equipment Drain Sump Tank
   - R/B Sump Tank
   - Spent resin storage tanks (Solid Waste Management System)
   - Holdup tanks (CVCS)
   - Boric acid tanks (CVCS)

   The Detergent Drain Tank and Detergent Drain Monitor Tank, and the Chemical Drain Tank are not in curbed cubicles because they do not contain a significant amount of radioactive material. In addition, these tanks are processed as soon as they are filled, so the likelihood of tank failure is reduced.

3. The equipment cubicles in the LWMS are coated with the same epoxy coating as the tank cubicles. Equipment cubicles are not curbed, because the equipment (e.g., pumps, filters) does not contain enough fluid during processing to represent a significant amount of radioactive material. In addition, after each processing the equipment is flushed, ensuring that no radioactive material remains inside the equipment. The Detergent Drain Tank Pump and the Detergent Drain Monitor Tank Pump are not enclosed in a cubicle but are housed in a contained portion of the A/B. This is because the Detergent Drain Subsystem does not typically contain any significant levels of radioactive contaminants (Section 11.2.2.2.8), and therefore leaks do not represent the risk of contamination to the facility, environment, or groundwater. However, all areas inside the A/B (including the floor under these pumps) will be coated as described in 12.3.1.1.2.D, "The decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of decontaminable paints and suitable smooth-surface coatings to the concrete floors and walls."

4. This information has been included in the response to Question No.11.02-6 item 2 (b).

Impact on DCD

The following changes will be made to the DCD to reflect the design change from stainless steel liners to epoxy coating:

Chapter 11 (Section 11.2.1.4, 5th paragraph)

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 where the radioactive liquid is stored are curbed and ~~lined~~ coated up to a wall height equivalent to one full tank volume of liquid for that tank. This ~~liner system~~ epoxy coating 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.

Chapter 11 (Section 11.2.2.2.2, 2nd paragraph)

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. 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.

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

02/18/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 164-1925 REVISION 0**
SRP SECTION:	**11.02 - LIQUID WASTE MANAGEMENT SYSTEM**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**1/23/2009**

QUESTION NO.: 11.02-2

Staff review of DCD Tier 2 (Rev 1), Section 11.2 indicates insufficient information is provided in regards to conformance equipment codes and the preoperational test for stainless-steel liners in cell/cubicles used for storing and processing radioactive liquids for compliance with 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 20.1406; 10 CFR 20, Appendix B, Table 2, Column 2; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix I; 10 CFR 50, Appendix A, GDC 60 and GDC 61; and 10 CFR 52.47(b)(1). Section 11.2, Table 11.2-1 presents RG 1.143 equipment codes for LWMS components such as pressure vessels, tanks, pumps, piping and valves, flexible hoses and connections, filters, and ion exchange columns. However, codes for the design and construction, materials, welding, and inspection and testing of stainless-steel liners in cells/cubicles used to minimize contamination of the facility, environment, and groundwater are not provided. Additionally, the test for stainless-steel liners associated with the ITP as part of the initial plant start-up is not provided. Please address the following items and revise the DCD to include this information.

1. Provide the conformance equipment codes for stainless-steel liners in cells/cubicles, or justify their exclusion.
2. Provide the ITAAC to ensure that construction of stainless-steel liners in cells/cubicles is complete and acceptable, or justify their exclusion.

ANSWER:

Since the DCD Rev. 1 was issued, the US-APWR design has changed to use an epoxy coating to line cubicles, instead of stainless steel. This design is in accordance with Regulatory Guide 4.21, and the changes were described to the NRC in RAI No. 91-1496, Revision 1, Question No 12.03-12.04-2:

> *"Additions to DCD:*
> *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."

The non-porous material described above is an epoxy coating with design in accordance with RG 4.21 (for example, forming a seal that is impermeable, durable, and with readily cleanable surfaces that facilitate decontamination). The DCD will be updated to reflect this design change as shown in "Impact on DCD" below.

Proven coating systems developed and qualified in accordance with accepted nuclear industry standards will be utilized and applied directly to the concrete. Testing programs under the jurisdiction of The American National Standards Institute (ANSI) have demonstrated the radiation tolerance and the decontamination properties of a variety of epoxies. This testing was conducted by various independent laboratories, such as Oak Ridge National Laboratory, Idaho Nuclear, and The Western New York Nuclear Research Center.

The ANSI standard ANSI N5.9-1967 "Protective Coatings (Paints) for the Nuclear Industry" (Rev. ANSI N512-1974) will be utilized.

Revision of the relevant ANSI documents is presently under the jurisdiction of the American Society for Testing and Materials (ASTM) as outlined in D3842-80 "Standard Guide for Selection of Test Methods for Coatings Used in Light-Water Nuclear Power Plants". In particular, the ASTM Standard D3843-00 (2008), "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities" will be utilized, as well as the ASTM D-5144-91 "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants" and its Referenced Documents.

The following lists typical coating systems that will be considered, as qualified for Coating Nuclear Service Level One "for those systems applied to structures, systems and other safety related components which are essential to the prevention of, or the mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public".

CONCRETE COATING SYSTEMS		
System KL-2 Curing Compound/Sealer Surfacer Finish	 KL4129 EPOXY CLEAR CURING COMPOUND KL6548S EPOXY SURFACER KLE1SERIES EPOXY ENAMEL	 0.5 -1.75 mils DFT Flush – 50.0 mils DFT 2.5 – 6.0 mils DFT
System KL-8 Curing Compound/Sealer Surfacer Finish	 KL4129 EPOXY CLEAR CURING COMPOUND KL6548S EPOXY SURFACER KLD1SERIES EPOXY HI-BUILD ENAMEL	 0.5 -1.75 mils DFT Flush – 50.0 mils DFT 4.0 – 8.0 mils DFT
System KL-9 Curing Compound/Sealer Surfacer Finish	 KL4129 EPOXY CLEAR CURING COMPOUND KL65487107 EPOXY WHITE PRIMER KLD1SERIES EPOXY HI-BUILD ENAMEL	 0.5 -1.75 mils DFT 5.0 – 10.0 mils DFT 3.0 – 8.0 mils DFT
System KL-10 Curing Compound/Sealer Surfacer Finish	 KL4129 EPOXY CLEAR CURING COMPOUND KL4000 EPOXY SURFACER KLD1SERIES EPOXY HI-BUILD ENAMEL	 0.5 -1.75 mils DFT Flush – 50.0 mils DFT 3.0 – 6.0 mils DFT
System KL-12 Curing Compound/Sealer Surfacer/Finish	 KL4129 EPOXY CLEAR CURING COMPOUND KL4500 EPOXY SELF-PRIMING SURFACING ENAMEL	 0.5 -1.75 mils DFT 10.0 – 50.0 mils DFT
System KL-14 (FLOORS ONLY) Primer/Sealer Finish	 KL6129 EPOXY CLEAR PRIMER/SEALER KL5000 EPOXY SELF-LEVELING FLOOR COATING	 1.5 -2.5 mils DFT 35.0 – 50.0 mils DFT

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.

Impact on DCD

The changes described in Impact on DCD in the response to Question No.11.02-1 above will be made to the DCD to reflect the design change from stainless steel liners to epoxy coating:

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

02/18/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 164-1925 REVISION 0**
SRP SECTION:	**11.02 - LIQUID WASTE MANAGEMENT SYSTEM**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**1/23/2009**

QUESTION NO.: 11.02-3

DCD Tier 2 (Rev 1), Section 11.2.1.6 discusses a provision for a mobile system or temporary equipment for liquid radioactive waste processing that may be installed in the auxiliary building at the discretion of facility operation. The mobile system or temporary equipment for liquid waste processing is not included in the LWMS which is designed with permanently installed equipment. Although Col item 11.2(1) for a mobile system or temporary equipment is presented in Section 11.2.4, there is no explicit statement in Section 11.2.1.6 to direct the COL applicant to take responsibility for this information item. Please include a statement for the COL applicant to address this information item in the discussion of Section 11.2.1.6. Also, please include a similar statement in the relevant section of the DCD for the COL applicant to address COL item 11.2(2) in Section 11.2.4.

ANSWER:

DCD Sections 11.2.1.6 on Mobile or Temporary Equipment and 11.2.3.1 on Radioactive Effluent Releases and Dose Calculation will be revised to direct the COL applicant to Section 11.2.4 on Combined License Information.

Impact on DCD

The following changes will be made to the DCD Section 11.2.1.6 on Mobile or Temporary Equipment

<u>Chapter 11 (Section 11.2.1.6)</u>

The LWMS is designed with permanently installed equipment (i.e., tanks, filters, activated charcoal filter, ion exchange columns, and pumps). The LWMS does not include the use of mobile or temporary equipment. However, a space is provided inside the A/B to accommodate future installation of mobile or temporary equipment. Process and utility piping and electrical connections are provided to forward liquid waste to future mobile system or temporary equipment, at the discretion of the facility operation. Treated liquid can be returned to the waste monitor tanks for sampling, recycling, and/or release. <u>The COL applicant is responsible for ensuring that mobile and temporary liquid radwaste processing equipment and its interconnection to plant systems conforms to regulatory requirements and guidance such as 10 CFR 50.34a (Ref. 11.2-5), 10 CFR</u>

20.1406 (Ref.11.2-7) and RG 1.143 (Ref. 11.2-3).

Also, the following paragraph will be added between 1st and 2nd paragraphs in DCD Section 11.2.3.1 on Radioactive Effluent Releases and Dose Calculation in Normal Operation in the next DCD Revision:

Chapter 11 (Section 11.2.3.1, between 1st and 2nd paragraph)

The release physical location and configuration of the treated effluent is site specific. Detailed design information such as release point, effluent temperature and flow rate, and size and shape of flow orifices, is to be presented in the site specific detail design. The COL applicant is responsible for ensuring that the site-specific information of the LWMS, e.g., radioactive release points, effluent temperature, shape of flow orifices, etc., is to be provided in the COLA (COL 11.2(2).

Impact on COLA

There is no impact to 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

02/18/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 164-1925 REVISION 0**

SRP SECTION: **11.02 - LIQUID WASTE MANAGEMENT SYSTEM**

APPLICATION SECTION: **11.2**

DATE OF RAI ISSUE: **1/23/2009**

QUESTION NO.: 11.02-4

DCD Tier 2 (Rev 1), Section 11.2, Table 11.2-14 presents some input design parameters and values used in the LADTAP II computer code and resulting individual annual population pathway doses (mrem/yr) from liquid radioactive effluents in Table 11.2-15. Staff review indicates insufficient information is provided to independently confirm the calculated individual annual population pathway doses for compliance with 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix I; and 10 CFR 50, Appendix A, GDC 60 and GDC 61. Please address the following items and revise the DCD to include this information..

1. Provide the basis for all design parameters and values used in the LADTAP II code calculation. Include value derivations and references (e.g., pointer to FSAR section or table, RG 1.109 table, etc.).
2. Provide the LADTAP II code input/output files used to calculate the liquid effluent doses in Table 11.2-15.

ANSWER:

1. The design bases for the input parameters of the LADTAP code used for the calculation of individual dose due to liquid radioactive effluents release are given below. However, many of the necessary input parameters are site-specific environmental characteristics. As a result, there is no clear reference for some of the parameters. Instead, the values of the parameters are based on reasonable assumptions that may apply to many, but not all, sites. For those parameters in which this is the case, this response indicates that the parameter should be considered in the COL, since site-specific information will be available in the COL to justify and/or modify the assumptions (as mentioned in DCD subsection 11.2.4 COL 11.2(4)).

Midpoint of Plant Life: 30 yr

This value is based on the design life of 60 years (DCD Table 1.3-1) for the US-APWR.

Reactor effluent discharge rate: 12,900 gpm

The assumption is made that circulating water in the secondary system is used for dilution and the flow rate of circulating water is described in DCD Table 10.4.5-1. For this assumption, the site specific condition of the plant will be reflected in the COL.

Water type selection: Freshwater

Fresh water is assumed as a condition for evaluation in the DCD. For this assumption, the site specific condition of the plant will be reflected in the COL.

Reconcentration model index: 0 (no model)

"No model" is selected as a condition for evaluation in the DCD. For this assumption, the site specific condition of the plant will be reflected in the COL.

Shore-width factor: 0.2

Discharge into rivers is assumed as a condition for evaluation in the DCD. According to R.G.1.109 Table A-2, the value of this factor is set as 0.2. For this assumption, the site specific condition of the plant will be reflected in the COL.

Dilution factors ("aquatic food and boating", "shoreline and swimming", "drinking water" and "irrigation water"): 10

"10" is assumed as a condition for evaluation in the DCD. For this assumption, the site specific condition of the plant will be reflected in the COL.

Irrigation rate: 31 Liter/m^2-month

An irrigation rate of 1 Liter/m^2 per day is assumed as a condition for evaluation in the DCD, on the basis of which, 31 Liter/m^2-month is obtained by multiplying by the number of days in one month which is conservatively assumed to be 31 for all months. For this assumption, the site specific condition of the plant will reflected in the COL.

Animal considered for milk pathway: Cow

The cow is considered as a condition for evaluation in the DCD. For this assumption, the site specific condition of the plant will be reflected in the COL.

Fraction of animal feed not contaminated: 0

"0" is conservatively set as a condition for evaluation in the DCD. For this assumption, the site specific condition of the plant will be reflected in the COL.

Fraction of animal water not contaminated: 0

"0" is conservatively set as a condition for evaluation in the DCD. For this assumption, the site specific condition of the plant will be reflected in the COL.

Source terms: DCD Table 11.2-10

The expected annual radioactive release rate is described in DCD Table 11.2-10.

Other parameters: R.G. 1.109

Other parameters are as per R.G.1.109 (that is, dose conversion coefficient: Table E-6 to 14, consumption rate: Table E-5, transition time: Table E-15, feed crop for domestic animals and intake of potable water: Table E-3, elapse time from production of irrigation foods to consumption: Table E-15).

2. The input and output files of the LADTAPII code are attached as Appendix to the response to RAI No.164-1925.

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

02/18/2009

US-APWR Design Certification
Mitsubishi Heavy Industries
Docket No. 52-021

RAI NO.:	**NO. 164-1925 REVISION 0**
SRP SECTION:	**11.02 - LIQUID WASTE MANAGEMENT SYSTEM**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**1/23/2009**

QUESTION NO.: 11.02-5

DCD Tier 2 (Rev 1), Section 11.2.1.2, first bulleted paragraph, please check the applicability of the pointer to Table 11.2-1 on RG 1.143 equipment codes to describe the capacity, redundancy, and flexibility attributes for LWMS design criteria.

ANSWER:

Reference to Table 11.2-1 will be deleted from DCD Section 11.2.1.2, first bulleted paragraph in the next DCD revision.

Impact on DCD

The following changes will be made to the Tier 2 DCD, Section 11.2.1.2, first bulleted paragraph:

Chapter 11 (Section 11.2.1.2)
The LWMS has sufficient capacity, redundancy, and flexibility (see ~~Table 11.2-1 and~~ Table 11.2-19) to process incoming waste streams to meet the concentration limits of Title 10, Code of Federal Regulations (CFR), Part 20 (Ref. 11.2-1) during periods of equipment downtime and during operation at design basis fission product leakage levels (i.e., leakage from fuel producing 1% of the reactor thermal power level). The processing capabilities are such that the operation of the plant will not be impaired under these conditions.

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

02/18/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 164-1925 REVISION 0**
SRP SECTION:	**11.02 - LIQUID WASTE MANAGEMENT SYSTEM**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**1/23/2009**

QUESTION NO.: 11.02-6

Staff review of DCD Tier 2 (Rev 1), Sections 11.2.3.2 and 2.4.13, and MHI's response to RAI (Rev 0), Question 11.01-2 indicate insufficient information is provided in regards to the liquid radwaste tank failure analysis for compliance with 10 CFR 20, Appendix B, Table 2, Column 2; 10 CFR 20.1406; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix A, GDC 60 and GDC 61; and 10 CFR 50, Appendix I. Pursuant to SRP Sections 11.2 and 2.4.13, and BTP 11-6, the staff requests this information to evaluate the basis and assumptions used in developing the source term, radionuclide distributions and concentrations to ensure that the highest potential radioactive material inventory associated with normal operation and anticipated operational occurrences is selected for liquid effluents processed by the LWMS, and to determine whether the failed tank and its components will result in the highest radionuclide concentrations at the nearest potable water supply located in an unrestricted area. Please address the following items and revise the DCD to include this information.

1. In response to the staff's question, MHI provided an evaluation to exclude Tc-99 from primary and secondary coolant concentrations based on: 1) core inventories of Tc-99 (2.3E+3 Ci), I-129 (5.6 Ci), and Cs-137 (1.9E+5 Ci) calculated by the ORIGEN code; 2) effluent concentration limits (ECLs) of Tc-99 (6E-5 mCi/ml) and Cs-137 (1E- 6 mCi/ml) in 10 CFR 20, Appendix B, Table 2, Column 2; and 3) a dose consequence of Tc-99 in comparison to Cs-137 from radiological half-lives, core inventories, and ECLs assuming a 365 day travel time and no retarding effect due to radionuclide deposition. The dose consequence analysis for liquid radwaste system failures must consider the most adverse contamination in groundwater (SRP 2.4.13) and the radionuclides selected for the source term and total inventory should include those that have the highest potential exposure consequences to users of water resources, including long-lived fission and activation products and environmentally mobile radionuclides (BTP 11-6). Because Tc-99 and I-129 move readily with groundwater and Cs-137 movement is highly retarded due to interactions with soil and hydrological travel times to unrestricted areas are typically much longer than 1 year, please include the Tc-99 and I-129 concentrations in the tank failure analysis in Section 11.2.3.2, or justify their exclusion in an evaluation which considers the environmental (fate and transport) characteristics of Tc-99, I-129, and Cs-137.

2. In Section 11.2.3.2, fully describe the approach used to demonstrate that liquid radioactive effluents processed by the LWMS released into the surface or ground

water 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.

(a) Provide details of the calculations in developing the radioactive source term as radionuclide distributions and concentrations for the tank inventories. Provide the basis for all design parameters and values used. Include value derivations and references (e.g., pointer to applicable FSAR section or table, etc.).

(b) Provide the tank inventories evaluated and identify the tank selected to contain the highest inventory for the tank failure analysis.

(c) Provide the basis for Note 1 in Table 11.2-17 to exclude radionuclides with concentrations less than 1E-3 in fraction of the ECL from the total inventory of the failed tank.

(d) Discuss the equipment malfunction analysis in Table 11.2-18 (Sheets 1 and 2) not described in Section 11.2.

(e) Provide the basis for the assumed dilution of 4.4E+10 gallons of water.

(f) Describe any credit applied in the use of engineered design features for mitigating radiological consequences of the tank failure.

(g) Provide the resulting radionuclide concentrations at the receptor location.

ANSWER:

1.

In the calculation of hydrological travel time in DCD, "365 days" is assumed as the time required for ground water to travel. In general, the distribution coefficients of Tc-99 and I-129 are small compared to Cs-137, resulting in shorter travel times. However, the DCD analysis assumes a travel speed identical to that of ground water for Cs-137, which conservatively neglects the adsorption effect by the soil. As a result, the same hydrological travel time is used for Cs-137, Tc-99, and I-129. As was described previously in the response to Question 11.01-02 of RAI 29, this conservative assumption of travel time made for Cs-137 results in Cs-137 being the dominant nuclide, so that the contribution of Tc-99 and I-129 can be neglected.

2.

(a)

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.

From among the inputs to the RATAF code, the following parameters are used as general conditions:

- Core thermal power
- Reactor coolant mass
- Reactor coolant letdown flow
- CVCS cation demineralizer flow rate
- Shim bleed flow rate
- Coolant drainage flow rate
- Dirty drainage flow rate
- Fraction of reactor coolant activity(Coolant drain and Dirty drain)

The values of the above parameters are identical to those given in DCD Table 11.2-9.

In the evaluation of the activity of the primary coolant, the reference activity included in the RATAF code is used. However, since a fuel defect of 1% is assumed for the reference activity in the RATAF code, the result of RATAF code calculation is multiplied by 0.12. Thus, the evaluation is made for a fuel defect of 0.12% as per BTP 11-6.

The following parameters are set as the condition for each tank:

- Volume of tank
  - ➢ Holdup tank : 1.2E+5 gal (16,000 ft^3)
    DCD Table 9.3.4-3
  - ➢ Waste holdup tank : 3.0E+4 gal
    DCD Table 11.2-3
  - ➢ Boric acid tank : 6.6E+4 gal
    DCD Table 9.3.4-3

- Tank factor (for "iodine", "Cs and Rb" and "other nuclides")

In the calculation of tank factors, the installation conditions with respect to demineralizers and an evaporator installed upstream of each tank are taken into account, according to NUREG-0133 Appendix A.

  - ➢ Holdup tank : 1.0 (All nuclides)
    The tank factor is set at 1.0, as the demineralizer and the evaporator are not installed upstream of the holdup tank. (The effects of the Mixed bed demineralizer and the cation bed demineralizer in the CVCS upstream of the holdup tank are already taken into consideration by the RATAF code)
  - ➢ Waste holdup tank : 1.0 (All nuclides)
    The tank factor is set at 1.0, as the demineralizer and the evaporator are not installed upstream of the waste holdup tank.
  - ➢ Boric acid tank : 1.0(tritium) ,0.2 (Iodine), 0.04(Cs,Rb), 0.2(Others)
    The boric acid evaporator feed demineralizer and the boric acid evaporator are installed upstream of the boric acid tank. The tank factors for the demineralizer and the evaporator are as per NUREG-0133 Appendix A and NUREG-0017 Rev.1. The tank factor for each nuclide is given in Table 1 below. As for tritium, the tank factor for both the demineralizer and the evaporator are 1.0.

Table A2(a)-1 Tank factor

	I	Cs,Rb	Others
Boric acid evaporator feed demineralizer	10	2	10
Boric acid evaporator	0.02	0.02	0.02
Total	0.2	0.04	0.2

- Hydrological dilution factor
  The calculation method for the hydrological dilution factor of each tank is as given in DCD Table 11.2-16, Note 1.

- Hydrological travel time
  The travel time of ground water is uniformly set at 365 days for all tanks, as an evaluation condition in the DCD. Transport delay due to adsorption of nuclides by the soil is conservatively neglected. The site specific condition of the plant is taken into consideration in the COL. If the dilution can not be shown to be within the DCD evaluation, a re-evaluation will be necessary.

(b)

In performing the evaluation of postulated radioactive releases due to liquid-containing tank failures, the following tanks were considered in determining which tank would have the highest concentration and the largest volume of radionuclides:

- Holdup Tank
- Waste Holdup Tank

- Boric Acid Evaporator
- Boric Acid Tank
- Volume Control Tank
- Auxiliary Building Sump Tank
- Reactor Building Sump Tank
- Primary Makeup Water Tank
- Refueling Water Storage Auxiliary Tank
- Chemical Drain Tank

The Volume Control Tank, Chemical Drain Tank, and Sump Tanks were eliminated from consideration based on having smaller volumes and on having radionuclide contents lower than the Boric Acid Tank (BAT). The boric acid evaporator contains liquid having a concentration of comparable order to that of the BAT, but the volume is much smaller, thus it was excluded from the consideration. The Primary Makeup Water Tank was eliminated from consideration based upon the fact that the Primary Makeup Water Tank stores demineralized water from the Treatment System, and low level radioactive condensate water from the Boric Acid Evaporator. Condensate water contains low levels of radionuclide concentrations, including tritium. The Refueling Water Storage Auxiliary Tank was eliminated from consideration because it stores refueling water. Prior to refueling, tank water is supplied to the refueling cavity where the reactor coolant radionuclide concentration dilutes with refueling cavity water. The radionuclide concentration of cavity water is reduced by the purification system of the Chemical Volume and Control System and Spent Fuel Pit Cooling and Purification System during refueling operations. Upon completion of refueling, part of the cavity water is returned to this tank where the radionuclide concentration is low. Accordingly, the impact of a Refueling Water Storage Auxiliary Tank or a Primary Makeup Water Tank failure would be small.

NUREG-0133 and the RATAF code for pressurized water reactors were utilized in selecting the appropriate tank for the failure analysis. The concentration of the radioactive liquid in the tanks, such as the Boric Acid Evaporator, the Holdup Tank, and the BAT, are larger than the Waste Holdup Tank since they receive reactor coolant water extracted from the Reactor Coolant System. Since an enrichment factor of 50 is assumed for the liquid phase of the Boric Acid Evaporator, the radioactive concentrations in the liquid phase of the Boric Acid Evaporator and in the BAT (which receives the enriched liquid from the Boric Acid Evaporator) are large when compared to other tanks. The BAT has been selected since its volume is larger than the liquid phase of the Boric Acid Evaporator.

The following tables show the concentration of radioactivity and the inventory of each tank, from the evaluation results using the RATAF code for the holdup tank, waste holdup tank and BAT. These values are corrected for the fuel defect of 0.12%, as described earlier. As for other nuclides not indicated in these tables, only the total amount collectively called "All others" is output in the RATAF code since their contribution to the environment at the time of failure is insignificant. Their contribution to the fraction of concentration limit is quite small (less than 1% for all the nuclides included in "All others"), and thus is neglected in the evaluation.

The holdup tank has a relatively high tritium inventory. The 10CFR20 Appendix B concentration limit of Tritium (1.0E-03 μCi/ml), however, is far larger than the one of Cesium-134 (9.0E-07 μCi/ml) and Cesium-137 (1.0E-06 μCi/ml). Accordingly, the BAT, which has the highest inventories of Cs-134 and Cs-137, is the most critical from the standpoint of liquid tank failure analysis.

Table A2(b)-1 Holdup tank

	Concentration	Inventory(1)
H-3	7.8E-01 μCi/ml	2.8E+02 Ci
Cs-134	5.6E-03 μCi/ml	2.0 Ci
Cs-137	4.1E-03 μCi/ml	1.5 Ci

Note:

1. Assuming the water quantity equal to 80% of the tank volume

Table A2(b)-2 Waste Holdup Tank

	Concentration	Inventory(1)
Cs-134	2.5E-03 μCi/ml	2.3E-01 Ci
Cs-137	1.8E-03 μCi/ml	1.6E-01 Ci

Note:

1. Assuming the water quantity equal to 80% of the tank volume

Table A2(b)-3 Boric Acid Tank

	Concentration	Inventory(1)
Cs-134	1.1E-01 μCi/ml	2.2E+01 Ci
Cs-137	1.0E-01 μCi/ml	2.0E+01 Ci

Note:

1. Assuming the water quantity equal to 80% of the tank volume。

(c)

In the RATAF code, the calculation result outputs the total for "All others", which includes nuclides whose fraction of concentration limit is less than 1E-3. This evaluation excludes these nuclides as their contribution to the fraction of concentration limit is of negligible order (less than 1% for all the nuclides included in "All others").

(d)

In the evaluation of Liquid Containing Tank Failures, the tanks that contain a large amount of radioactivity are selected as described in subsection 11.2.3.2. 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.

(e)

The evaluation scenario in the DCD assumes that the contaminated ground water leaks into a body of water in the vicinity of ponds surrounding the site, and will be taken up downstream after dilution in the ponds.

The dilution rate in the ponds is as assumed to be 4.4E+10 gal, which is a volume of an order comparable to that of the Squaw Creek Reservoir on the Comanche Peak site.

In this condition, the site specific condition of the plant is taken into consideration in COL. If the dilution can not be shown to be within the DCD evaluation, a re-evaluation will be necessary.

(f)

Credit is taken for removal and concentration effects of the demineralizers and the boric acid evaporator.

Credit is taken for radioactive decay during travel time. Credit is taken for dilution by the hydrological dilution factor.

(g)

The resulting radionuclide concentrations at the receptor location have already been given as "Critical Receptor Concentration" in Table 11.2-17 of DCD. This phrase is taken from the output of the RATAF code.

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

02/18/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 164-1925 REVISION 0**
SRP SECTION:	**11.02 - LIQUID WASTE MANAGEMENT SYSTEM**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**1/23/2009**

QUESTION NO.: 11.02-7

DCD Tier 2 (Rev 1), Section 11.2, Tables 11.2-10 and 11.2-11 present calculated expected and maximum annual liquid radionuclide releases (Ci/yr) from some input design parameters and values in Tables 11.2-7 and 11.2-9 used in the PWR-GALE computer code. The resulting calculated annual liquid radionuclide releases are compared to 10 CFR 20 Appendix B liquid effluent concentration limits in Tables 11.2-12 and 11.2-13. Staff review indicates insufficient information is provided to independently confirm the calculated annual liquid radionuclide releases for compliance with 10 CFR 20.1302; 10 CFR 20, Appendix B, Table 2, Column 2; 10 CFR 50, Appendix I; 10 CFR 50.34a; and 10 CFR 50, Appendix A, GDC 60. Please address the following items and revise the DCD to include this information.

1. Provide the basis for all values and assumptions used in the PWR-GALE code calculation of expected and maximum annual liquid radioactive releases in Table 11.2-7. Include value derivations and references (e.g., pointer to applicable FSAR section, NUREG-0017, etc.).
2. Provide the PWR-GALE code input/output files used to calculate the expected and maximum annual liquid radionuclide releases in Tables 11.2-10 and 11.2-11.

ANSWER:

1.

The design bases for the input parameters of the PWR-GALE code used for the calculation of the expected annual liquid radioactive release are given below:

- Decontamination Factors (DFs) described in DCD Table 11.2-7
  The values are taken from NUREG-0017 Rev.1, Subsections 2.2.18, 2.2.19 and 2.2.20.

- Core thermal power: 4,451 MWt
  This is the value for 100% thermal power (DCD Table 1.3-1).

- Reactor coolant mass: 6.46E+05 lb
  This design value is the integrated volume of the primary coolant circulating in the core.

- Reactor coolant letdown flow rate: 180 gpm
  This is the design value (DCD Table 9.3.4-2).

- CVCS cation demineralizer flow rate: 7 gpm
  This is an assumption for the GALE calculation. The flow through the cation demineralizer can be increased up to the design flow rate of 110 gpm, however, a lower more conservative value is assumed.

- Number of SGs: 4
  This is the design value (DCD Table 1.3-2).

- Total main steam flow rate: 2.02E+07 lb/hr
  This is the design value (DCD Table 10.1-1).

- Secondary coolant mass in SG: 1.35E+05 lb
  This is the design value based on 100% power under normal operations.

- Total SG blowdown flow rate: 1.554E+05 lb/hr
  This is the flow rate of the SG blowdown water through the SG purification system. The value is conservatively assumed to be lower than the design value for the GALE calculation.

- Blowdown treatment method: 0
  Since the blowdown is reused in the condenser after treatment by the SG blowdown demineralizer (DCD Figures 10.4.8-1 and 10.4.8-2), "0" is input according to NUREG-0017, Rev.1, Subsection 1.5.2.9.

- Regeneration time of condensate polishing system: N/A
  The condensate demineralizer is not assumed to be used for regeneration (DCD Subsection 10.4.6.2.3.1).

- Fraction of feedwater through the condensate polishing system: 0
  Since the condensate demineralizer is conservatively assumed not to be used, "0" is input according to NUREG-0017, Rev.1, Subsection 1.5.2.10.

- 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). As this value is integrated in the PWR-GALE program code, the code has been modified to reflect the parameter.

- Decontamination factor for detergent waste: 1.0
  Since it is assumed that detergent waste will be discharged without treatment, "1.0" is used as input according to NUREG-0017, Rev.1, Subsection 1.5.2.23.

- Shim bleed
  - Shim bleed rate: 2,875 gpd
    Based on WASH-1258, eight shutdowns per 24 months (four hot shutdowns, three cold shutdowns and one refueling shutdown) are assumed, and on this basis, the amount of water that will be generated due to addition and dilution of boron in the primary coolant and expansion of the coolant is obtained as input for the parameter.

  - Decontamination factors for "I", "Cs and Rb" and "others"
    Total DFs for each demineralizer and the boric acid evaporator are the values given in

DCD Table 11.2-7, while Table A1-1 below shows the itemized list of equipment and calculation results for the total DFs.

Table A1-1 Demineralizers of Shim Bleed and Coolant Drain

	I	Cs and Rb	Others
Boric acid evaporator feed demineralizer	10	2	10
Boric acid evaporator	10^2	10^3	10^3
C - Waste demineraliizer (Mix bed)(1)	5	1	10
D - Waste demineraliizer (Mix bed)	1(2)	1(2)	1(2)
Total	$5x10^3$	$2x10^3$	10^5

(1) As this demineralizer is installed downstream of the boric acid evaporator, the DF of the evaporator condensate described in DCD Table 11.2-7 is applied to the shim bleed and coolant drain.

(2) No credit is taken for this demineralizer.

- Collection time: 20 days

  The capacity of the holdup tank to receive shim bleed and coolant drain is 16,000 ft^3/unit (DCD Table 9.3.4-3) x 3 units, while the total in-flow rate of shim bleed and coolant drain is 3,775 gpd. The above values are calculated according to the formula described in NUREG-0017, Rev.1, Subsection 1.5.2.12.3.

- Process and discharge time: 2 days

  The process time has been calculated based on the values below and according to the formula described in NUREG-0017, Rev.1, Subsection 1.5.2.12.4, with the discharge time neglected for conservatism. The initial capacity of the tank in-flow is 16,000 ft^3/unit x 3 units (holdup tanks). The equipment flow processing capacity is 30 gpm (the capacity of the Boric acid evaporator) from DCD Table 9.3.4-3.

- Fraction of waste to be discharged : 1.0

  The total discharge of waste is conservatively assumed.

- Coolant Drain
  - Coolant drainage flow rate: 900 gpd
  - Fraction of reactor coolant activity: 0.1

    RCP seal leakage is assumed as coolant drain. The amount of drain to be generated and the activity are given in DCD Table 11.2-2.

  - Decontamination factor for "I", "Cs and Rb" and "others"
  - Collection time : 20 days
  - Process and discharge time: 2 days

    Since the coolant drain flows into the holdup tank like the shim bleed, the parameter values are the same as those for shim bleed.

  - Fraction waste to be discharged: 1.0

    The total discharge of waste is conservatively assumed.

Dirty waste

- Dirty waste flow rate: 2,023 gpd
- Fraction of reactor coolant activity: 0.18

  Waste liquid flowing into the waste hold tank is treated as dirty waste. The itemized amount of dirty waste and activity are given in DCD Table 11.2-2. This information and the totals are shown in Table A1-2 below. As for the equipment and area decontamination, a shutdown period of 30 days with an input rate of 3,000 gpd is assumed, while the remaining period regarded as a normal operation period with an input rate of 40 gpd. Thus, the average amount of waste liquid per day is calculated as

in the following:
(3,000 x 30 + 40 x 335) / 365 = 283 gpd

Table A1-2 Breakdown of Dirty Waste

Sources	Expected Input Rate (gpd)	Activity
Reactor containment cooling	500	0.1% of reactor coolant
Leakage inside containment (to containment sump)	10	167% of reactor coolant
Leakage outside containment	80	100% of reactor coolant
Equipment drainage	250	100% of reactor coolant
Spent fuel pool liner leakage	25	0.1% of reactor coolant
Miscellaneous drainage	675	0.1% of reactor coolant
Equipment and area decontamination	283	1% of reactor coolant
Sampling drainage	200	5% of reactor coolant
	2,023(Total)	18% of reactor coolant(Average)

- Decontamination factor for "I", "Cs and Rb" and "others"
  Total DFs for each demineralizer and the boric acid evaporator are the values given in DCD Table 11.2-7, while Table A1-3 shows the itemized list of equipment and calculation results for the total DFs.

Table A1-3 Demineralizers of Dirty Waste

	I	Cs and Rb	Others
A – Waste demineralizer (Anion bed)	10^2	1	1
B – Waste demineralizer (Cation bed)	1	10	10
C – Waste demineraliizer (Mix bed)(1)	10^2	2	10^2
D – Waste demineraliizer (Mix bed)(1)	$10^{(2)}$	$10^{(2)}$	$10^{(2)}$
Total	10^5	$2x10^2$	10^4

(1) As the processing system for dirty waste has no evaporator, the radwaste DF described in DCD Table 11.2-7 is applied to dirty waste.
(2) The DF of the second one of the series-connected demineralizers is used.

- Collection time: 5 days
  The capacity of the holdup tank to receive dirty waste is 30,000 gal/unit (DCD Table 11.2-3) x 4 units. An in-flow rate of 4,740 gpd at the time of shutdown with higher drain generation is conservatively assumed. The above values have been calculated according to the formula described in NUREG-0017, Rev.1, Subsection 1.5.2.12.3.

- Process and discharge time: 0 days
  The process time has been calculated based on the values below and according to the formula described in NUREG-0017, Rev.1, Subsection 1.5.2.12.4. The initial capacity of the tank in flow is 30,000 gal/unit x 4 units (waste holdup tanks). The equipment flow processing capacity is 90 gpm (the design flow of the waste demineralizer) from DCD Table 11.2-6. The overall result is less than one day, which is rounded down to "0" days. The discharge time is also conservatively neglected.

- Fraction of waste to be discharged: 1.0
  The total discharge of waste is conservatively assumed.

・Blowdown waste
- Fraction of the blowdown stream processed: 1.0

It was assumed that the total amount of blowdown waste would be processed by the blowdown demineralizer.

- Decontamination factor for "I", "Cs and Rb" and "others"
  Total DFs for each demineralizer and the boric acid evaporator are the values given in DCD Table 11.2-7, while Table A1-4 shows the itemized list of equipment and calculation results for the total DFs.

Table A1-4 SG Blowdown Demineralizers

	I	Cs and Rb	Others
SG blowdown cation bed demineralizer	1	10	10
SG blowdown mix bed demineralizer	10^2	10	10^2
Total	10^2	10^2	10^3

- Fraction of waste to be discharged: 0
  As the total amount of blowdown waste is to be reused, none of the waste is to be discharged and "0" is set.

- Regenerant waste
  As the demineralizer is not assumed to be used for regeneration, no regenerant waste will be generated and thus is not applicable.

- Gaseous Waste Management System and HVAC System
- Continuous gas stripping of full letdown flow: None
  Continuous gas stripping will not be used.

- Holdup time for Xe: 45 days
  This is the design value (DCD Table 11.3-1).

- Holdup time for Kr : 2.55 days
  This is the design value of the expected delay time based on the adsorption coefficient for Kr.

- Fill time of decay tanks for gas stripper
  No decay tank is installed.

- Gas waste system: High-efficiency particulate air (HEPA) filter: None
  No HEPA filter is installed (DCD Figure 9.4.3-1 and Figure 11.3-1).

- Auxiliary building: Charcoal filter: None
  No charcoal filter is installed (DCD Figure 9.4.3-1).

- Auxiliary building: HEPA filter: None
  No HEPA filter is installed (DCD Figure 9.4.3-1).

- Containment volume: 2.74E+06 ft^3
  This is the design value (DCD Table 6.2.1-5).

- Containment atmosphere internal cleanup rate : 0 ft^3/min
- Removal efficiency of charcoal filter: 0%
- Removal efficiency of HEPA filter: 0%
  There is no containment atmosphere internal cleanup system, HEPA filters, or charcoal filters.

- Number of purges per year (in addition to two shutdown purges): 0
  Only the two shutdown purges per year are assumed.

- Removal efficiency of charcoal filter: 0%
  No charcoal filter is installed in the high volume purge exhaust system (DCD Figure 9.4.6-1).

- Removal efficiency of HEPA filter: 99%
  Following NUREG-0017, Rev.1, Subsection 1.5.2.19, the removal efficiency of the HEPA filter is set at 99% (DCD Figure 9.4.6-1).

- Containment low volume purge rate: 2,000 ft^3/min
  This is the design value (DCD Table 9.4.6-1)

- Removal efficiency of charcoal filter: 70%
  Following NUREG-0017, Rev.1, Subsection 1.5.2.20, the removal efficiency of the charcoal filter is set at 70% (DCD Figure 9.4.6-1).

- Removal efficiency of HEPA filter: 99%
  Following NUREG-0017, Rev.1, Subsection 1.5.2.20, the removal efficiency of the HEPA filter is set at 99% (DCD Figure 9.4.6-1).

- Fraction of iodine released from blowdown tank vent: 0
  Since there is no direct vent from the blowdown tank to the atmosphere, "0" is used as input according to NUREG-0017, Rev.1, Subsection 1.5.2.21.

- Fraction of iodine removed from main condenser air ejector release: 0
  Since direct release is assumed as no charcoal filter is installed, "0" is used as input according to NUREG-0017, Rev.1, Subsection 1.5.2.22.

Liquid releases with maximum defined fuel defects have been calculated as indicated below, where the primary reactor coolant activity ratio was used, while corrections were made for calculated discharges (except for detergent waste) for each path of expected release were. However, as the amount formed of the nuclides in corrosion and activation products do not depend on fuel defects, no correction was made.

As in the case of expected releases, the total amount of release includes an adjustment due to AOOs of 0.16 Ci/y for the total of all nuclides, which is distributed to each nuclide in proportion to the combined release.

$$Qij\,max = (Ci\,max/Ci\,real) \times Qij\,real$$

where:

Qij max: Amount released of nuclide "i" in path "j" (shim bleed, misc. wastes, turbine building) (maximum)
Ci max: Design basis primary reactor coolant activity of nuclide "i" (DCD Table 11.1-2)
Ci real: Realistic primary reactor coolant activity of nuclide "i" (DCD Table 11.1-9)
Qij real: Amount released of nuclide "i" in path "j" (shim bleed, misc. wastes, turbine building) (realistic)

Nuclides whose realistic primary reactor coolant activities are "0" are treated as follows:

- P-32, Ni-63, Sb-124
  Expected release of these nuclides occurs only in the case of detergent waste. As for the released radioactivity contained in detergent waste, the values in NUREG-0017, Rev.1, Table 2-27 are used, which are independent of the fuel defect rate, so that no

correction is made for these nuclides.

- Rh-103m, Rh-106, Ag-110
  As these nuclides have a secular equilibrium with their parent nuclide, the amounts released from shim bleed, misc. wastes and turbine building are assumed to be identical to those of their parent nuclide as shown in Table A1-5 below.

Table A1-5 Parent and Daughter Nuclides

Parent	Daughter
Ru-103 (Half life: 3.96E+1 day)	Rh-103m (Half life: 3.96E-2 day)
Ru-106 (Half life: 3.67E+2 day)	Rh-106 (Half life: 3.47E-4 day)
Ag-110m (Half life: 2.53E+2 day)	Ag-110 (Half life: 2.82E-4 day)

- Pr-143
  Since Ce-143 is the parent nuclide for Pr-143, the amounts released from shim bleed, misc. wastes and turbine building are corrected using the concentration ratio of Ce-143.

2.

The input and output files of the PWR-GALE code are attached as Appendix to the response to RAI No.164-1925.

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 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 item 11.02-4 and 11.02-7 of NRC Requests" –Proprietary information

Contents of CD

File Name	Size	Sensitivity level
• LADTAPII Input:		
- LADINP.DAT (txt format)	3KB	Proprietary
- LADTAP_RG1_109.LIB	0.18MB	Proprietary
• LADTAPII output:		
- LADINP.DAT.oultlist (txt format)	0.18MB	Proprietary
• PWR-GALE input:		
- 99V2LQ_INPUT (txt format)	4KB	Proprietary
• PWR-GALE output:		
- 99V2LQ_INPUT.outlist (txt format)	0.12MB	Proprietary