

  
**MITSUBISHI HEAVY INDUSTRIES, LTD.**  
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TOKYO, JAPAN

February 6, 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-09047

**Subject: MHI's Response to US-APWR DCD RAI No. 140-1732 Rev. 1 through  
145-1736 Rev. 0**

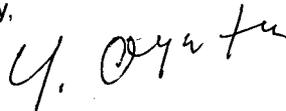
- References:** 1) "Request for Additional Information No. 140-1732 Revision 1, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2," dated January 9, 2009  
2) "Request for Additional Information No. 141-1735 Revision 1, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2," dated January 9, 2009  
3) "Request for Additional Information No. 142-1733 Revision 1, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2," dated January 9, 2009  
4) "Request for Additional Information No. 143-1737 Revision 1, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2," dated January 9, 2009  
5) "Request for Additional Information No. 144-1738 Revision 1, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2," dated January 9, 2009  
6) "Request for Additional Information No. 145-1736 Revision 0, SRP Section: 12.02 – Radiation Sources, Application Section: 12.2," dated January 9, 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 ten RAIs contained within References 1 through 6.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittal. His contact information is below.

Sincerely,



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

DO81  
NRC

Enclosures:

1. Response to Request for Additional Information No.140-1732 Revision 1
2. Response to Request for Additional Information No.141-1735 Revision 1
3. Response to Request for Additional Information No.142-1733 Revision 1
4. Response to Request for Additional Information No.143-1737 Revision 1
5. Response to Request for Additional Information No.144-1738 Revision 1
6. Response to Request for Additional Information No.145-1736 Revision 0

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

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Enclosure 1

UAP-HF-09047  
Docket No. 52-021

Response to Request for Additional Information  
No. 140-1732 Revision 1

February, 2009

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

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

**RAI NO.:** NO. 140-1732 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-4**

10 CFR 20.1101(b) requires licensees to ensure that engineering controls are used to keep occupational doses ALARA. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 "Contained Sources" notes that the applicant is to provide the models, parameters and bases for all values used to calculate source magnitudes, for normal and accident conditions. The guidance contained in NUREG-0800 "Standard Review Plan 12.2" notes under the "Acceptance Criteria" that for PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations should be based on a primary coolant concentration of 3.5  $\mu\text{Ci/gm}$ . Operational Experience from PWR plants that are using 2 year fuel cycles without recycling of RCS as Primary Make Up Water, indicate that RCS tritium activity is above 1  $\text{uCi/gm}$  during portions of the operating cycle.

**Question 1:**

10 CFR 20.1101(b) and 10 CFR 20.1701 require licensees to ensure that engineering controls are used to keep occupational doses ALARA. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 and C.I.12.2 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes. The guidance contained in NUREG-0800 "Standard Review Plan 12.2" notes under the "Acceptance Criteria" that for PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations should be based on a primary coolant concentration of 3.5  $\mu\text{Ci/gm}$ . Operational Experience from PWR plants that are using 2 year fuel cycles without recycling of RCS as Primary Make Up Water, indicate that RCS tritium activity is above 1  $\text{uCi/gm}$  during portions of the operating cycle.

The APWR DCD Section 9.3.4.2.5 notes that distillate from boric acid evaporator is either transferred to the Primary Makeup Water Tank or released to the liquid waste management system (LWMS). In DCD Table 11.1-9 "Realistic Source Terms" the value used for RCS tritium activity is listed as 1  $\mu\text{Ci/g}$ . For the purposes of occupational radiation exposure control, the value provided in Table 11.1-9 is not conservative with respect to the guidance provided in Nureg-0800 Section 12.2, and observed Operational Experience.

**ANSWER:**

The primary coolant tritium concentration used in the airborne activity calculation is 3.5  $\mu\text{Ci/g}$ . MHI will revise DCD Subsection 12.2, Airborne Radioactive Material Sources, to provide a list of

parameters (see the response to question No.12.02-10 of RAI 143-1732 revision 1).

**Impact on DCD**

See the impact on DCD in the response to question No.12.02-10 of RAI 143-1732 revision 1.

**Impact on COLA**

There is no impact on the COLA.

**Impact on PRA**

There is no impact on the PRA.

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Enclosure 2

UAP-HF-09047  
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Response to Request for Additional Information  
No. 141-1735 Revision 1

February, 2009

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** NO. 141-1735 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.:** 12.02-5

10 CFR 20.1101(b), 1201 and 1202 require licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment. The guidance contained in Regulatory Guide 8.8 notes that to provide a basis for design, the quantity and isotopic composition of the radioactive material contained, in the station equipment should be estimated and that fission product source terms should be estimated using 0.25% fuel cladding defects. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes, including isotopic composition and the bases for all values. The data provided for the SFP, Irradiated Incore Detectors, and Irradiated Incore Detector cables in Section 12.2 appears to be insufficient or consistent enough to accurately describe the source term associated with these components.

**Question 1:**

The use of a Co-60 only source term for the SFP noted in the APWR DCD Section 12.2.1.1.5 is not consistent with the APWR DCD Section 12.2.1.1 notes that the source term for design bases is operation with 1% cladding defects. DCD Section 12.2.1.1.5, regarding SFP activity states "Fission products in the reactor coolant are negligible today due to technological improvements in nuclear fuel integrity resulting in a reduced fuel defect fraction". However, DCD Tables 12.2-5 "Pressurizer Liquid Phase Source Strength" and Table 12.2-14 "Chemical and Volume Control System Radiation Sources Letdown Heat Exchanger Source Strength", assume design bases cladding defects. The gamma ray source strengths for 1, 1.5 and 2 MeV reported for fluid streams addressed in DCD Tables 12.2-5 and 12.2-14, are not conservative with respect to the use of a Co-60 only source term.

In accordance with RG 1.206, if Co-60 is the only isotope used to calculate the source SFP source term during refueling, then please revised chapter 12 to included justification for the discrepancy between the stated fuel cladding defect design bases and the assumption that fission products in the reactor coolant system would be negligible. Otherwise, please revise chapter 12 to provide information that describes the bases, methods, parameters, assumptions, and resultant dose rates from the SFP fluid.

**ANSWER:**

The activity concentrations used in the US-APWR shielding design source terms are set in the following ways, in general.

One method of determining the activity concentration assumes the leakage of fission products (FPs) due to fuel defects into the reactor coolant system (RCS), which allow for the calculation of the activity concentration contained in the equipment of each system. The concentration of corrosion products (CPs) is assumed to be constant. In this calculation, the processes of accumulation, removal, and decay of the ionic radioactive nuclides contained in the primary coolant are considered in the downstream system equipment. This is the method that is commonly applied in setting the activity concentration for the US-APWR.

Long term operating experience shows that some equipment has a high dose rate even with no fuel defects and that the increase in dose rate in that equipment is due to the accumulation of CPs. For this reason, another method of determining the activity concentration is on the basis of the upper limit of that surface dose rate on equipment, etc. This method applies to the case where the source nuclides contributing to the equipment dose rate are mainly CPs. These source nuclides move from the primary coolant to the downstream equipment. Since the water quality in the equipment changes depending on the operating conditions, the uncertainty increases in the analysis of the behavior of the accumulation of CPs. For this reason, a reliable and reasonable design basis is adopted where the dose rate is set on the basis of data from operation. As a typical nuclide, Co-60 is conservatively selected from among CP nuclides which has a relatively long half life and emits high energy gamma rays. The upper limit of the dose rate for each piece of equipment is determined on the basis of operating experience, and actual operation is controlled so as not to exceed the dose rate. This equipment includes water filters, low level water containing SFP water, and several devices such as drums with miscellaneous solid materials, etc.

As described above, the activity concentration for contained sources for the US-APWR is mainly calculated based on the assumption of 1% fuel defects. It is only for several pieces of equipment where CPs are the major source of radiation where the activity concentration is obtained from the upper limit of the dose rate of the equipment (based on operating data).

DCD Section 12.2.1.1.5 discusses activity in the SFP and states, "Fission products in the reactor coolant are negligible today due to technological improvements in nuclear fuel integrity resulting in a reduced fuel defect fraction". This is a statement concerning the improvement in control of the channel head for SFP water. The dose strength values shown in DCD Tables 12.2-5 (pressurizer liquid phase source strength) and 12.2-14 (letdown heat exchanger source strength) are those of the RCS and the CVCS, which are adequately conservative based on the assumption of FPs due to fuel defects.

**Impact on DCD**

There is no impact on the DCD.

**Impact on COLA**

There is no impact on the COLA.

**Impact on PRA**

There is no impact on the PRA.

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** NO. 141-1735 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-6**

10 CFR 20.1101(b), 1201 and 1202 require licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment. The guidance contained in Regulatory Guide 8.8 notes that to provide a basis for design, the quantity and isotopic composition of the radioactive material contained, in the station equipment should be estimated and that fission product source terms should be estimated using 0.25% fuel cladding defects. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes, including isotopic composition and the bases for all values. The data provided for the SFP, Irradiated Incore Detectors, and Irradiated Incore Detector cables in Section 12.2 appears to be insufficient or consistent enough to accurately describe the source term associated with these components

**Question 2:**

The APWR DCD Section 12.2.1.2.5 notes that maximum gamma ray source strengths for the SFP, Irradiated Incore Detectors, and Irradiated Incore Detector cables are tabulated in DCD Tables 12.2-55, 12.2-54, 12.2-55 and 12.2-56. These tables only provide source strength data in (MeV/cm<sup>3</sup>/sec) or (MeV/watt-sec). DCD Section 12.2.1.1 notes that the design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power. There does not appear to be sufficient information regarding the models, assumptions and parameters provided in chapter 12 to justify the values presented in DCD Tables 12.2-55, 12.2-54, 12.2-55 and 12.2-56. Some of the information that may be needed to calculate the source strengths and isotopic concentrations includes, but is not limited to:

- isotopic composition models and parameters, such as assumed
- exposure duration for components inserted into the core,
- alloy composition for incore detectors,
- isotopic concentrations for the spent fuel pool water,
- the assumed leakage fraction from spent fuel bundles with defective cladding,
- the initial activity of the water used to flood up the Refueling Cavity,
- the assumed dilution ratio of the RCS activity as a result of the cool down and depressurization process and

the fractional mixing between the Spent Fuel Pool and the Refueling Cavity during refueling.

In accordance with RG-8.8 and 1.206, please revise DCD chapter 12.2 to provide the information regarding the models, assumptions and parameters needed to generate the data presented in Tables 12.2-55, 12.2-54, 12.2-55 and 12.2-56.

**ANSWER:**

[SFP]

The source strength of spent fuel (SF) is significantly higher than that of spent fuel pit (SFP) water. However, the adequate depth of the SFP water from its surface down to the SF in storage acts as a shield to reduce the dose from the SF. Therefore, the dose rate due to SFP water can not be neglected compared to that of SF. This is the reason that the SFP shielding calculation for the water surface dose considers both SF and SFP water as radiation sources.

In the shielding calculation for the wall and floor directions of the SFP, on the other hand, only SF is taken into account as a radiation source. This is because of the fact that, in the wall and floor directions of the SFP, the SF has such a high local source strength that the dose rate from the SFP water can be neglected.

Source strengths of the SF are calculated using the ORIGEN code. The calculation parameters for the ORIGEN code will be added to the DCD as is described in the "Impact on DCD" section of this response. The source strengths of the SFP water are described in greater detail in the response to Question No. 12.02-5.

During operation, if leakage occurs from damaged fuel assemblies during storage in the SFP, the SFP cooling and purification system is operated so that the dose at the surface of the SFP water is always controlled to meet the dose rate criterion. In the shielding design, on the other hand, the source strength is normalized to the maximum dose on the SFP water. Therefore, leakage from damaged fuel assemblies is bounded by this conservative assumption.

[Irradiated incore detector and drive cable]

The irradiated incore detector and drive cable, as well as the flux thimble tube are activated during operation in the core by neutron irradiation. All these activities are calculated using the following equation:

$$A = \frac{1}{3.7 \times 10^4} N \cdot \sigma \cdot \phi [1 - \exp(-\lambda t_1)] \cdot \exp(-\lambda t_2)$$

where:

- A = activity ( $\mu\text{Ci}/\text{cm}^3$ )
- N = isotope number density ( $1/\text{cm}^3$ )
- $\sigma$  = activation cross section ( $\text{cm}^2$ )
- $\phi$  = neutron flux ( $\text{n}/\text{cm}^2/\text{s}$ )
- $\lambda$  = decay constant ( $1/\text{s}$ )
- $t_1$  = irradiation period (s)
- $t_2$  = time after shutdown (s)

In setting the irradiation period in the activation calculation, the irradiation is assumed to be continuous and conservatively neglects the cooling effect due to non-irradiation during the irradiation process. The calculation parameters for the activation calculation will be added to the DCD as described in the "Impact on DCD" section of this response.

The activation calculation of the irradiated incore detector and drive cable, as well as the flux thimble tube is based on the evaluation of data from a conventional Japanese PWR. Since the power density of the US-APWR is smaller than that of the conventional Japanese PWR, the neutron flux in the US-APWR will also be smaller than what is assumed in the calculation. Thus, the neutron flux used in the activation calculation is conservative.

#### **Impact on DCD**

The DCD Subsection 12.2.1.2.3 and 12.2.1.2.5 will be revised, and Table 12.2-70 and 12.2-71 will be added to the DCD, as described in the Attachment 2.

#### **Impact on COLA**

There is no impact on the COLA.

#### **Impact on PRA**

There is no impact on the PRA.

### 12.2.1.2.3 Spent Fuel

The predominant radioactivity sources in the spent fuel storage and transfer areas in the Reactor Building (R/B) are the spent fuel assemblies. The source strengths employed to determine the minimum water depth above spent fuel and shielding walls around the SFP, as well as shielding of the spent fuel transfer tube, are tabulated in Table 12.2-54. For the shielding design, the SFP is assumed to contain the design maximum number of fuel assemblies. To be conservative, 257 spent fuel assemblies, assumed to be from unloading the full core with only a 24-hour decay period, are assumed to be located in the outer rows of the spent fuel racks. The remaining assemblies, from previous refueling operations, do not significantly affect the shield wall design due to the shielding of the intervening, recently discharged assemblies.

The source strengths in Table 12.2-54 are also used in the evaluation of radiation levels for spent fuel handling, storage, and shipping. These sources are calculated using the ORIGEN code, based on specific power of 32.1 MW/MTU and burnup of 62 GWD/MTU, which is a limitation for maximum burnup for fuel rod as described in ~~chapter~~ Chapter 4, Subsection 4.2.1. Other calculation parameters are tabulated in Table 12.2-70.

### 12.2.1.2.4 Control Rods, Primary and Secondary Source Rods

As source material, byproduct material or special nuclear material, there are primary and secondary source rods. As described in Chapter 4, Subsection 4.2.2.3 and 4.2.2.3.3, a primary source rod contains californium-252 source, a secondary source rod contains antimony-beryllium source. These rods are stored in the SFP after use. Irradiated control rods are also stored in the SFP. Source strengths of these rods are less than that of spent fuel. Therefore, in radiation shielding design, source strengths of spent fuel are used as these rods' source strengths.

### 12.2.1.2.5 Incore Flux Thimbles

Irradiated incore detector and drive cable maximum gamma ray source strengths are tabulated in Table 12.2-55. These source strengths are used in determining shielding requirements and evaluating occupational radiation exposure when detectors are being moved during or following a flux mapping of the reactor core. These source strengths are given an irradiation period of 20 hours, respectively, and are given in terms of per cubic centimeters ( $\text{cm}^3$ ) of detector and per centimeters of drive cable. Irradiated incore detector drive cable average gamma ray source strengths are tabulated in Table 12.2-56. These source strengths are used in determining shielding requirements when the detectors are not in use and for shipment when the detectors have failed. The values are given in terms of per centimeters of drive cable after an irradiation period of 20 hours. Irradiated incore flux thimble gamma ray source strengths are tabulated in Table 12.2-57. These source strengths are used in determining shielding requirements during refueling operations when the flux thimbles are withdrawn from the reactor core. The values are given in terms of per  $\text{cm}^3$  stainless steel for an irradiation period of 60 years. The flux thimbles are made of type 316 stainless steel with a maximum cobalt impurity content of 0.10 weight percent.

All these activities are calculated using the following equation:

$$A = \frac{1}{3.7 \times 10^4} N \cdot \sigma \cdot \varphi [1 - \exp(-\lambda t_1)] \cdot \exp(-\lambda t_2) \quad \text{Eq. 12.2-3}$$

where:

<u>A</u>	=	<u>activity (<math>\mu\text{Ci}/\text{cm}^3</math>)</u>
<u>N</u>	=	<u>isotope number density (<math>1/\text{cm}^3</math>)</u>
<u><math>\sigma</math></u>	=	<u>activation cross section (<math>\text{cm}^2</math>)</u>
<u><math>\varphi</math></u>	=	<u>neutron flux (<math>\text{n}/\text{cm}^2/\text{s}</math>)</u>
<u><math>\lambda</math></u>	=	<u>decay constant (<math>1/\text{s}</math>)</u>
<u><math>t_1</math></u>	=	<u>irradiation period (s)</u>
<u><math>t_2</math></u>	=	<u>time after shutdown (s)</u>

Other calculation parameters are tabulated in Table 12.2-71.

### 12.2.1.3 Sources for the Design-Basis Accident

The radiation sources of importance for the DBA are the containment source and the RHRS and Containment Spray System sources.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in RG 1.183 (Reference 12.2-4). The integrated gamma ray and beta particle source strengths for various time-periods following the postulated accident are tabulated in Table 12.2-58.

The RHRS and shielding are designed to allow limited access to the RHR pumps following a DBA. The sources are based on the assumptions in RG 1.183 (Reference 12.2-4). Noble gases formed by the decay of halogens in the sump water are assumed to be retained in the water. Credit has been taken for dilution by the RCS volume plus the contents of the refueling water storage. Gamma ray source strengths for radiation sources circulating in the RHR loop and associated equipment are tabulated in Table 12.2-59.

### 12.2.2 Airborne Radioactive Material Sources

This section deals with the models, parameters, and sources required to evaluate the airborne concentration of radionuclides during the plant operations in the various plant radiation areas where personnel occupancy is expected.

**Table 12.2-70 Parameters and Assumptions for Calculating Spent Fuel Source Strength**

Parameter / Assumption	Value
Burn up	62 GWd/MTU
Core thermal output	4451 MWt
Number of fuel assemblies	257
Specific power	32.1 MW/MTU
Fuel enrichment	4.55 wt%
Effective fuel length and width	420cm x 21.4cm x 21.4cm

**Table 12.2-71 Parameters and Assumptions for Calculating Irradiated Incore Detector, Drive Cable and Flux Thimble Source Strength**

Parameter / Assumption	Value			
	Detector	Drive Cable		Flux Thimble
Chemical composition <sup>*1</sup> (%)	Fe: 49 Cr: - Ni: 50 Mn: 0.7 Co: 0.03 P: - S: - Si: 0.19 Mo: -	Fe: 65 Cr: 19 Ni: 13 Mn: 2.0 Co: 0.2 P: 0.04 S: 0.03 Si: 1.0 Mo: -	Fe: 98 Cr: 0.15 Ni: 0.7 Mn: 0.6 Co: 0.02 P: 0.04 S: 0.04 Si: 0.35 Mo: 0.6	Fe: 66 Cr: 18 Ni: 14 Mn: 2.0 Co: 0.1 P: - S: - Si: - Mo: -
Density (g/cm <sup>3</sup> )	7.85	7.9	7.85	7.98
Mixing ratio <sup>*2</sup>	1.0	0.1	0.9	1.0
Neutron flux <sup>*3</sup> (n/cm <sup>2</sup> /s)	φ <sub>1</sub> : 8.5E+13 φ <sub>2</sub> : 1.4E+14 φ <sub>3</sub> : 9.5E+13 φ <sub>4</sub> : 6.0E+13	φ <sub>1</sub> : 8.5E+13 φ <sub>2</sub> : 1.4E+14 φ <sub>3</sub> : 9.5E+13 φ <sub>4</sub> : 6.0E+13	φ <sub>1</sub> : 8.5E+13 φ <sub>2</sub> : 1.4E+14 φ <sub>3</sub> : 9.5E+13 φ <sub>4</sub> : 6.0E+13	φ <sub>1</sub> : 8.2E+13 φ <sub>2</sub> : 1.4E+14 φ <sub>3</sub> : 9.3E+13 φ <sub>4</sub> : 5.8E+13
Irradiation period	20 (h)	20 (h)	20 (h)	60 (y)

\*1) For chemical composition, a "-" after an element indicates "essentially zero".

\*2) Drive cable consists of two materials. Source strength for drive cable was calculated by summing source strength for each material with this mixing ratio.

\*3) Subscripts for neutron flux indicate range of neutron energy. (1: E > 1MeV, 2: 5.53 keV < E < 1 MeV, 3: 0.414 eV < E < 5.53 keV, 4: E < 0.414 eV)

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Enclosure 3

UAP-HF-09047  
Docket No. 52-021

Response to Request for Additional Information  
No. 142-1733 Revision 1

February, 2009

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** NO. 142-1733 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-7**

10 CFR 20.1101(b), 1201 and 1202 requires licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment. Regulatory Guide 8.8 notes that the applicant should estimate the quantity and isotopic composition of the radioactive material to be contained, deposited, or accumulated in station equipment. The guidance contained in Regulatory Guide 1.206 section C.1.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes, including isotopic composition and the bases for all values.

**Question 1:**

The APWR DCD Section 9.3.4.2.5 "Boron Recycle Subsystem" notes that the concentration of the boric acid is gradually increased until it reaches 7,000 ppmB then the concentrate is transferred to the boric acid tank. Any process that increases the concentration of boric acid, will also increase the concentration of the activity remaining after the water passes through the Boric Acid Feed Demineralizer. DCD Section 12.2.1.1 notes that the design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power. DCD Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown (Sheet 15 of 34) Auxiliary Building at Elevation -26'-4"" identifies the Boric Acid Storage Tanks as a Zone VIII (Maximum Dose Rate 100 R/h) area and the Boric Acid Transfer Pumps as a Zone VII (Maximum Dose Rate 10 R/h) area. DCD Section 12.2.1.1.3 "Sources for Full Power Operation" does not discuss the Boric Acid Storage Tanks as a source of radiation exposure.

In accordance with RG 1.206, please update chapter 12 to provide models, parameters and bases used to calculate the source magnitudes in the Boric Acid Storage Tanks and the Boric Acid Transfer Pumps. The information provided should reflect the expected affect on the source term contained in these components resulting from the operation of the Boric Acid Evaporator for an operating cycle with design basis cladding defects.

**ANSWER:**

The boron recycle system is installed downstream of the holdup tanks to condense the primary coolant processed by the B.A. evaporator feed demineralizer to have a concentration of 7,000 ppmB. In the calculation of activity concentration for shield design, the boron concentration is

condensed from 200\* ppmB to 7,000 ppmB (by assuming condensation of coolant having a lower boron concentration, the activity concentration of concentrate in the boron recycle system becomes conservative). In other words, the activity concentration in the boron recycle system is condensed to a concentration of approximately 35 times that of effluent from the B.A evaporator feed demineralizer when compared to nuclides having long half life (but, the concentration rate of each nuclide varies depending on its own half life since decay during the condensing operation is taken into account).

\*: 200 ppmB is a shielding design value, which is determined in consideration of the amount of extracted water to the holdup tank (averaged over the operation period) and the dilution rate of the boron concentration. A reverse calculation is used to obtain the concentration rate of boron in the coolant at the time when the amount of extracted water becomes equal to the average over the operation period. This value is conservatively reduced to obtain the boron concentration of the liquid processed by the boron recycle system.

The boron recycle system continuously condenses the coolant processed by the B.A evaporator feed demineralizer, while extraction of the concentrate condensed to 7,000 ppmB is performed by a batch system, so that the concentrate intermittently flows into the boric acid tanks installed downstream of the boron recycle system. The activity concentration used for the shielding design of the boric acid tanks is based on the assumption that the concentrate condensed to 7,000 ppmB by the boron recycle system intermittently flows into the boric acid tanks and mixes, while the concentrate to be stored in the boric acid tanks decays during the time decay (equal to the condensing operation period of the boron recycle system) to receive the concentrate from the boron recycle system. In this calculation, the analysis time (allowing for decay of short lived nuclides) of the concentrate is neglected during preparation of batch operation, which means that the calculation results are conservative.

The boric acid transfer pump is used for circulating the concentrate in the boric acid tanks and also for transferring it to the charging system. The activity concentration of water inside the boric acid transfer pump is therefore equal to that of the boric acid tanks. However, because the amount of water contained in the boric acid pump is significantly less than that of the large boric acid tanks, the dose rate of the boric acid pump is less than that of boric acid tank (The boric acid transfer pump is assumed as a piping with nominal diameter of 2 in. and 3 in. in the shielding calculation).

#### **Impact on DCD**

The DCD will be revised to add the Subsection 12.2.1.1.3.F, and the parameters of the boric acid tanks (diameter, height) will be added to Table 12.2-1 (Sheet 3 of 4), as described in the Attachment 3.

#### **Impact on COLA**

There is no impact on the COLA.

#### **Impact on PRA**

There is no impact on the PRA.

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

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

**RAI NO.:** NO. 142-1733 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-8**

10 CFR 20.1101(b), 1201 and 1202 requires licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment. Regulatory Guide 8.8 notes that the applicant should estimate the quantity and isotopic composition of the radioactive material to be contained, deposited, or accumulated in station equipment. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes, including isotopic composition and the bases for all values.

**Question 2:**

The APWR DCD Section 12.2.1.1 notes that the design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power while activation product activity is calculated independently of the fuel defect level. Some of the activity contained in the fluid processed by the Boric Acid Evaporator, will deposit on interior components, resulting in the Boric Acid Evaporator becoming a significant source of exposure during maintenance activities. DCD Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown (Sheet 17 of 34) Auxiliary Building at Elevation 3'-7" ", identifies the Boric Acid Evaporator as a Zone VIII (Maximum Dose Rate 100 R/h) area. Based on the information depicted in this figure, two pumps are located next to the evaporator. Section 12.2 does not contain any information regarding the estimated activity contained within the Boric Acid Evaporator package and pumps.

In accordance with 10 CFR 20.1101(b) and RG 1.206, please provide in chapter 12, information that describes the bases, methods and assumptions for the source term, and dose rates, associated with the Boric Acid Evaporator, involving operation under "realistic" conditions, and for operation with design basis cladding defects. The information provided is to be consistent with the radiation zone information also provided in chapter 12.

**ANSWER:**

The boron recycle system is installed downstream of the coolant holdup tanks to condense the primary coolant processed by the B. A. evaporator feed demineralizer. After non-condensable

gases are removed from the primary coolant by means of the degasification device, condensate and distilled water are generated through continuous processing by the B. A. evaporator. The non-condensable gases (including noble gases) that were separated are fed to the B.A. evaporator vent condenser to be disposed of by the Gaseous Waste Management System.

Since the B.A. evaporator feed pumps are used for receiving the primary coolant from the holdup tanks, the activity concentration of these pumps is equal to that of the effluent from the holdup tanks that is processed by the B.A. evaporator feed demineralizer. Two B.A. evaporator feed pumps are installed, while only one of them is normally used.

Figure 12.3-1 (Sheet 17 of 34) in DCD Chapter 12 shows the radiation zone of the B.A. evaporator room to be VIII. Leakage of fission products (FPs) due to fuel defects of 1% from the fuel into the reactor coolant system (RCS) is assumed (the concentration of corrosion products assumed to be constant). A calculation is made to determine the source strength in downstream equipment of ionic radioactive nuclides contained in the primary coolant, considering the process of accumulation, removal and decay. The above radiation zone of VIII is based on the dose rate based on this analysis and is conservative.

In the US-APWR, the primary coolant extracted from the RCS passes through the mixed bed demineralizer inlet filter, the reactor coolant filter and the B.A. evaporator feed demineralizer filter, and then flows into the boron recycle system. As these filters remove suspended matter adequately, no radioactivity deposits will remain inside the US-APWR boron recycle system, so resulting dose rates will be insignificant for necessary maintenance activity.

#### **Impact on DCD**

The DCD Subsection 12.2.1.1.3.E will be revised, the parameters of the B.A. evaporator and the vent condenser (diameter, height) will be added to Table 12.2-1 (Sheet 3 of 4), and the Tables 12.2-66 through 12.2-69 will be added to the DCD, as described in the Attachment 3.

#### **Impact on COLA**

There is no impact on the COLA.

#### **Impact on PRA**

There is no impact on the PRA.

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

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

**RAI NO.:** NO. 142-1733 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-9**

10 CFR 20.1101(b), 1201 and 1202 requires licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment. Regulatory Guide 8.8 notes that the applicant should estimate the quantity and isotopic composition of the radioactive material to be contained, deposited, or accumulated in station equipment. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes, including isotopic composition and the bases for all values.

**Question 3:**

The APWR DCD Section 12.2.1.1 notes that the design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power. The purpose of the Boric Acid Evaporator is to concentrate boric acid contained in this fluid. Due to deposition of some of this activity, the evaporator may become a significant source of exposure. Figure 12.3-1 depicts it as a Zone VIII (Maximum Dose Rate 100 R/h) area. Beta, gamma and airborne exposure may be significant for maintenance, waste handling and dewatering activities that involve removal of the sludge that collects in the bottom of the evaporator. Section 12.2 does not provide any information that describes the source terms applicable for the removal and processing of this material.

In accordance with 10 CFR 20 and RG 1.206, please provide in chapter 12, information that describes the bases, methods and assumptions for the source terms, expected dose rates and airborne concentrations and the accumulated TEDE exposure, associated with the removal handling and dewatering of the Boric Acid Evaporator bottoms. This data provided should include separate presentations for operation under "realistic" conditions, and for operation with design bases cladding defects.

**ANSWER:**

As shown in the ANSWER to Question 2, the dose rate calculation uses the conservative source strength based on 1% fuel defect, and the radiation zone inside the B.A. evaporator room is VIII. However, since the actual fuel defect rate is much smaller than 1%, and there are three filters

installed between the RCS and the boron recycle system to sufficiently remove particulate radioactive materials, this dose rate is conservative. Operating experience shows no significant doses during maintenance activities of the boron recycle system.

**Impact on DCD**

There is no impact on the DCD.

**Impact on COLA**

There is no impact on the COLA.

**Impact on PRA**

There is no impact on the PRA.

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**D. Tanks**

- Volume control tank

The radiation sources in the volume control tank are based on a nominal operating level in the tank of 400 ft<sup>3</sup> in the liquid phase, 270 ft<sup>3</sup> in the vapor phase, and on the stripping fractions tabulated in Table 12.2-30, assuming no purge of the volume control tank. The values of the stripping fraction with assumption of purging are given in Table 12.2-31.

- Holdup tank

The radiation sources in the holdup tank are based on the maximum activity for both the liquid phase and vapor phase considering continuous inflow of the coolant and the decay during storage.

**E. B. A. evaporator**

The B. A. evaporator is used to remove nitrogen, hydrogen, and gaseous fission products from the reactor coolant and to concentrate the remaining borated water for reuse in the RCS. Effluent from the holdup tanks is processed by the B.A evaporator feed demineralizer, and the primary coolant that has been processed is received by the boron recycle system using the B.A. evaporator feed pumps. The primary coolant is condensed by the B.A. evaporator, and the coolant, after condensation, is sent to the boric acid transfer pumps; separated non-condensable and noble gases pass through the B.A. evaporator vent condenser to be disposed of by the GWMS. The source term in the B. A. evaporator is based on the intermittent processing of the coolant. The source terms for the B. A. evaporator and B. A. evaporator vent condenser are tabulated in Tables 12.2-66 through 12.2-69.

**F. Boric acid tank**

Boric acid tanks receive the concentrate processed by the B.A. evaporator intermittently. Boric acid in the boric acid tanks is reused as primary coolant after adjustment of the concentration by the boric acid blender.

**12.2.1.1.4 Essential Service Water System and Component Cooling Water System**

The essential service water system and the component cooling water system are normally non-radioactive or, because of inleakage, have very low activity. Radiation monitoring for these systems is described in Chapter 11, Subsection 11.5.2. For shielding and dose assessment purposes, the essential service water system and component cooling water system do not yield substantive doses.

**Table 12.2-1 Radiation Sources Parameters  
(Sheet 3 of 4)**

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft <sup>3</sup> )	Equipment Self-Shielding (in.)	
<b>Auxiliary Building</b>							
Mix bed demineralizer	23.7	68.9	Homogeneous	Water	62.4	ignored	2
Cation-bed demineralizer	15.9	65.6	Homogeneous	Water	62.4	ignored	1
Holdup tank Liquid Phase Vapor Phase	147.6	410.0 229.7 180.3	Homogenous Homogenous	Water Air	62.4 7.6E-02	ignored	3
Spent fuel pit demineralizer	23.7	68.9	Homogeneous	Water	62.4	ignored	2
Steam generator blowdown demineralizer	44.3	63.4	Homogeneous	Water	62.4	ignored	4
Waste holdup tank	128.0	138.6	Homogeneous	Water	62.4	ignored	4
Waste demineralizer	23.7	68.9	Homogeneous	Water	62.4	ignored	4
Charcoal bed Charcoal Phase Vapor Phase	23.7	126.0 68.8 57.2	Homogenous Homogenous	Charcoal Air	34.4 7.6E-02	ignored	4
Waste gas surge tank	74.8	167.0	Homogeneous	Air	7.6E-02	1.0	4
Spent resin storage tank	59.1	131.2	Homogeneous	Water	62.4	ignored	2
<u>B.A. evaporator</u>	<u>26.9</u>	<u>188.5</u>	<u>Homogeneous</u>	<u>Water</u>	<u>62.4</u>	<u>ignored</u>	<u>1</u>
<u>B.A. evaporator vent condenser</u>	<u>5.0</u>	<u>78.1</u>	<u>Homogeneous</u>	<u>Air</u>	<u>7.6E-02</u>	<u>ignored</u>	<u>1</u>
<u>Boric acid tank</u>	<u>118.1</u>	<u>361.5</u>	<u>Homogeneous</u>	<u>Water</u>	<u>62.4</u>	<u>ignored</u>	<u>2</u>
<b>Plant Yard Area (Outside the Power Block)</b>							
Refueling water storage auxiliary tank	236.2	536.6	Homogeneous	Water	62.4	ignored	1
Primary makeup water tank	183.1	316.9	Homogeneous	Water	62.4	ignored	2

**Table 12.2-66 Chemical and Volume Control System Radiation Sources**  
**B.A. Evaporator Activity**

<u>Nuclide</u>	<u>Activity</u> <u>(<math>\mu\text{Ci}/\text{cm}^3</math>)</u>	<u>Nuclide</u>	<u>Activity</u> <u>(<math>\mu\text{Ci}/\text{cm}^3</math>)</u>
<u>Br-82</u>	<u>2.0E-04</u>	<u>Te-129m</u>	<u>4.0E-04</u>
<u>Br-83</u>	<u>9.8E-05</u>	<u>Te-129</u>	<u>5.1E-06</u>
<u>Br-84</u>	<u>2.8E-06</u>	<u>Te-131m</u>	<u>6.7E-04</u>
<u>Rb-86</u>	<u>6.2E-02</u>	<u>Te-131</u>	<u>7.0E-07</u>
<u>Rb-88</u>	<u>3.1E-01</u>	<u>Te-132</u>	<u>9.7E-03</u>
<u>Rb-89</u>	<u>3.7E-04</u>	<u>Te-133m</u>	<u>6.5E-06</u>
<u>Sr-89</u>	<u>1.7E-04</u>	<u>Te-134</u>	<u>6.7E-06</u>
<u>Sr-90</u>	<u>8.4E-06</u>	<u>I-130</u>	<u>1.8E-03</u>
<u>Sr-91</u>	<u>2.4E-05</u>	<u>I-131</u>	<u>4.4E-02</u>
<u>Sr-92</u>	<u>2.2E-06</u>	<u>I-132</u>	<u>4.9E-02</u>
<u>Y-90</u>	<u>1.3E-03</u>	<u>I-133</u>	<u>4.3E-02</u>
<u>Y-91m</u>	<u>3.1E-05</u>	<u>I-134</u>	<u>1.3E-04</u>
<u>Y-91</u>	<u>2.2E-05</u>	<u>I-135</u>	<u>1.0E-02</u>
<u>Y-92</u>	<u>1.1E-05</u>	<u>Cs-132</u>	<u>7.1E-03</u>
<u>Y-93</u>	<u>4.8E-06</u>	<u>Cs-134</u>	<u>6.5E+00</u>
<u>Zr-95</u>	<u>2.5E-05</u>	<u>Cs-135m</u>	<u>4.1E-04</u>
<u>Nb-95</u>	<u>9.0E-05</u>	<u>Cs-136</u>	<u>1.6E+00</u>
<u>Mo-99</u>	<u>2.4E-02</u>	<u>Cs-137</u>	<u>3.7E+00</u>
<u>Mo-101</u>	<u>5.5E-07</u>	<u>Cs-138</u>	<u>2.7E-02</u>
<u>Tc-99m</u>	<u>1.3E-01</u>	<u>Ba-137m</u>	<u>2.0E+01</u>
<u>Ru-103</u>	<u>2.0E-05</u>	<u>Ba-140</u>	<u>1.5E-04</u>
<u>Ru-106</u>	<u>7.3E-06</u>	<u>La-140</u>	<u>1.7E-03</u>
<u>Ag-110m</u>	<u>6.7E-08</u>	<u>Ce-141</u>	<u>2.4E-05</u>
<u>Te-125m</u>	<u>2.9E-05</u>	<u>Ce-143</u>	<u>1.3E-05</u>
<u>Te-127m</u>	<u>1.2E-04</u>	<u>Ce-144</u>	<u>1.8E-05</u>
		<u>Pr-144</u>	<u>9.3E-04</u>
		<u>Pm-147</u>	<u>2.0E-06</u>
		<u>Eu-154</u>	<u>1.9E-07</u>
		<u>Na-24</u>	<u>1.1E-03</u>
		<u>Cr-51</u>	<u>2.5E-04</u>
		<u>Mn-54</u>	<u>1.8E-04</u>
		<u>Mn-56</u>	<u>3.7E-04</u>
		<u>Fe-55</u>	<u>1.7E-04</u>
		<u>Fe-59</u>	<u>3.0E-05</u>
		<u>Co-58</u>	<u>4.1E-04</u>
		<u>Co-60</u>	<u>6.1E-05</u>
		<u>Zn-65</u>	<u>5.0E-05</u>

**Table 12.2-67 Chemical and Volume Control System Radiation Sources**  
**B.A. Evaporator Source Strength**

<u>Gamma Ray Energy (MeV)</u>	<u>Source Strength (MeV/cm<sup>3</sup>/sec)</u>
<u>0.015</u>	<u>1.3E+02</u>
<u>0.02</u>	<u>7.8E+00</u>
<u>0.03</u>	<u>1.7E+03</u>
<u>0.04</u>	<u>5.2E+02</u>
<u>0.05</u>	<u>2.6E+00</u>
<u>0.06</u>	<u>4.6E+02</u>
<u>0.08</u>	<u>3.1E+02</u>
<u>0.1</u>	<u>2.7E+01</u>
<u>0.15</u>	<u>1.8E+03</u>
<u>0.2</u>	<u>1.8E+03</u>
<u>0.3</u>	<u>1.1E+04</u>
<u>0.4</u>	<u>5.9E+02</u>
<u>0.5</u>	<u>3.3E+03</u>
<u>0.6</u>	<u>5.7E+05</u>
<u>0.8</u>	<u>2.3E+05</u>
<u>1.0</u>	<u>6.9E+04</u>
<u>1.5</u>	<u>1.3E+04</u>
<u>2.0</u>	<u>5.6E+03</u>
<u>3.0</u>	<u>1.4E+03</u>
<u>4.0</u>	<u>1.1E+01</u>
<u>5.0</u>	<u>8.8E+01</u>

**Table 12.2-68 Chemical and Volume Control System Radiation Sources  
B.A. Evaporator Vent Condenser Activity**

<u>Nuclide</u>	<u>Activity (<math>\mu\text{Ci}/\text{cm}^3</math>)</u>
Kr-83m	7.3E-01
Kr-85m	4.5E+00
Kr-85	4.3E+02
Kr-87	1.1E+00
Kr-88	6.3E+00
Xe-131m	1.9E+01
Xe-133m	1.9E+01
Xe-133	1.4E+03
Xe-135m	1.6E+00
Xe-135	3.9E+01
Xe-138	1.1E-01

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

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

Docket No. 52-021  
MHI Ref. UAP-HF-09047

Enclosure 4

UAP-HF-09047  
Docket No. 52-021

Response to Request for Additional Information  
No. 143-1737 Revision 1

February, 2009

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

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

**RAI NO.:** NO. 143-1737 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-10**

10 CFR 20.1101(b), 1201 and 1202 require licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. Regulatory Guide 8.8 notes that the applicant should estimate the quantity and isotopic composition of the radioactive material contained, deposited or accumulated in station equipment. The guidance contained in Regulatory Guide 1.206 sections C.I.12.2.1 "Contained Sources" C.I.12.2.2 "Airborne Radioactive Material Sources" notes that the applicant is to provide the models, assumptions and parameters used to calculate airborne radioactive material source magnitudes, including isotopic composition and the bases for all values. The applicant should describe, and identify sources by location and magnitude and then tabulate the calculated concentrations of airborne radioactive material by nuclides. This information should be provided for normal operation, AOO, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. Airborne activity concentration values reported in Section 12.2, and the bases for the values presented, appear to be incomplete or inconsistent with other sections.

**Question 1:**

The APWR DCD Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactive Concentrations" only provides a partial list of the parameters and assumptions needed to calculate the airborne concentrations depicted in Table 12.2-61 "Airborne Radioactive Concentrations", and similar tables. These parameters should include assumptions such as the frequency of use of low volume purge systems, removal rates due to internal recirculation/cooling systems, any plate out of material on internal plant surfaces, the time after leak initiation for the stated concentration values and initial density of the leaking fluid.

In accordance with RG 1.206, please revise DCD section 12.2 to provide a complete list of parameters and assumptions that are consistent with, and support the airborne activity concentrations reported in section 12.2.

**ANSWER:**

MHI will revise DCD Subsection 12.2.2, Airborne Radioactive Material Sources, to provide a list of parameters and assumptions.

**Impact on DCD**

The DCD Subsection 12.2.2 and Table 12.2-60 will be revised, and Table 12.2-72 will be added to the DCD, as described in the Attachment 4.

**Impact on COLA**

There is no impact on the COLA.

**Impact on PRA**

There is no impact on the PRA.

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

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

**RAI NO.:** NO. 143-1737 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-11**

10 CFR 20.1101(b), 1201 and 1202 require licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. Regulatory Guide 8.8 notes that the applicant should estimate the quantity and isotopic composition of the radioactive material contained, deposited or accumulated in station equipment. The guidance contained in Regulatory Guide 1.206 sections C.I.12.2.1 "Contained Sources" C.I.12.2.2 "Airborne Radioactive Material Sources" notes that the applicant is to provide the models, assumptions and parameters used to calculate airborne radioactive material source magnitudes, including isotopic composition and the bases for all values. The applicant should describe, and identify sources by location and magnitude and then tabulate the calculated concentrations of airborne radioactive material by nuclides. This information should be provided for normal operation, AOO, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. Airborne activity concentration values reported in Section 12.2, and the bases for the values presented, appear to be incomplete or inconsistent with other sections.

**Question 2:**

The APWR DCD Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactive Concentrations" provides an assumed leakage rate of 100 lb/d. This corresponds to a leakage rate of about 0.01 gpm. DCD Chapter 16 section 3.4.13 RCS Operational Leakage states that unidentified leakage shall be limited to less than 1 gpm and identified leakage to less than 10 gpm. The use of 100 lb/d of leakage for calculating occupational airborne exposure concentrations is not conservative with respect to the stated allowable leakage rates.

In accordance with RG 1.206, please provide in chapter 12.2 the models, parameters and assumptions that justify the use of 100 lb/d leakage rate for calculating airborne concentration values for the purpose of radiation protection, when the Technical Specification allowable leakage rates are 1 gpm for unidentified and 10 gpm for identified.

**ANSWER:**

The calculation of airborne concentrations for the purpose of radiation protection is not part of the safety evaluation and is not associated with Technical Specification (TS) 3.4.13. The TS 3.4.13

LCO deals with the protection of the reactor coolant pressure boundary (RCPB) from degradation and the reactor core from inadequate cooling. TS 3.4.13 also prevents the plant from exceeding the accident analyses radiation release assumptions. The leakage values in the TS of 1 gpm unidentified leakage and 10 gpm identified leakage are therefore overly conservative to use for the radiation protection calculation. Instead, the operational leakage rate in the containment used in the calculation of occupational exposures in DCD Section 12.2 is 100 lb/d (about 0.01 gpm). The basis for this value is the leakage rate inside containment of 10 gpd (less than 0.01 gpm) taken from ANSI/ANS-55.6 "Liquid Radioactive Waste Processing for Light Water Reactor Plants," which is described in DCD Section 11.2 and shown in Table 11.2-2. Therefore, the assumed leakage rate during normal operation of 100 lb/h (slightly greater than 10 gpd) is a reasonable value for use in calculating the airborne concentrations for the purpose of radiation protection.

**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.

All these activities are calculated using the following equation:

$$A = \frac{1}{3.7 \times 10^4} N \cdot \sigma \cdot \varphi [1 - \exp(-\lambda t_1)] \cdot \exp(-\lambda t_2) \quad \text{Eq. 12.2-3}$$

where:

<u>A</u>	=	<u>activity (<math>\mu\text{Ci}/\text{cm}^3</math>)</u>
<u>N</u>	=	<u>isotope number density (<math>1/\text{cm}^3</math>)</u>
<u><math>\sigma</math></u>	=	<u>activation cross section (<math>\text{cm}^2</math>)</u>
<u><math>\varphi</math></u>	=	<u>neutron flux (<math>\text{n}/\text{cm}^2/\text{s}</math>)</u>
<u><math>\lambda</math></u>	=	<u>decay constant (<math>1/\text{s}</math>)</u>
<u><math>t_1</math></u>	=	<u>irradiation period (s)</u>
<u><math>t_2</math></u>	=	<u>time after shutdown (s)</u>

Other calculation parameters are tabulated in Table 12.2-71.

### 12.2.1.3 Sources for the Design-Basis Accident

The radiation sources of importance for the DBA are the containment source and the RHRS and Containment Spray System sources.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in RG 1.183 (Reference 12.2-4). The integrated gamma ray and beta particle source strengths for various time-periods following the postulated accident are tabulated in Table 12.2-58.

The RHRS and shielding are designed to allow limited access to the RHR pumps following a DBA. The sources are based on the assumptions in RG 1.183 (Reference 12.2-4). Noble gases formed by the decay of halogens in the sump water are assumed to be retained in the water. Credit has been taken for dilution by the RCS volume plus the contents of the refueling water storage. Gamma ray source strengths for radiation sources circulating in the RHR loop and associated equipment are tabulated in Table 12.2-59.

### 12.2.2 Airborne Radioactive Material Sources

This section deals with the models, parameters, and sources required to evaluate the airborne concentration of radionuclides during the plant operations in the various plant radiation areas where personnel occupancy is expected.

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Radioactive material that becomes airborne may come from the RCS, spent fuel pit, and refueling water storage pit. The calculation of potential airborne radioactivity in equipment cubicles, corridors, or operating areas normally occupied by operating personnel is based on reactor coolant activities given in Chapter 11, Section 11.1.

The assumptions and parameters required to evaluate the isotopic airborne concentrations in the various applicable regions are tabulated in Table 12.2-60 and table 12.2-72.

The CVCS and the RHRS are designed to provide the capability to purify the reactor coolant through the purification demineralizer after the reactor shutdown and cooldown. This mode of operation will ensure that the effect of activity spikes does not significantly contribute to the containment airborne activity during refueling operations.

Sources resulting from the removal of the reactor vessel head and the movement of spent fuel are dependent on a number of operating characteristics (e.g., coolant chemistry, fuel performance) and operating procedures followed during and after shutdown. The permissible coolant activity levels following de-pressurization are based on the noble gases evolved from the RCS water upon the removal of the reactor vessel head. The endpoint limit for coolant cleanup and degasification is established based on the maximum permissible concentration considerations and containment ventilation system capabilities of the plant.

The exposure rates at the surface of the refueling cavity and spent fuel pit water are dependent on the purification capabilities of the refueling cavity and spent fuel pit cleanup systems. A water total activity level of less than 0.005  $\mu\text{Ci/g}$  for the dominant gamma-emitting isotopes at the time of refueling leads to a dose rate at the water surface less than 2.5 mrem/h.

The detailed listing of the expected airborne isotopic concentrations in all the various plant regions is presented in Table 12.2-61. The final design of the plant ensures that all the expected airborne isotopic concentrations in all normally occupied areas are well below the derived air concentration (10 CFR 20 Appendix B [Reference 12.2-5]). If entry is needed in areas where airborne concentrations exceed the limit (such as containment during normal operation), occupancy time will be adjusted such that personal doses are in compliance with 10 CFR 20 (Reference 12.2-1).

#### **12.2.2.1 Containment Vessel Atmosphere**

The detailed listing of the expected airborne isotopic concentrations in the containment vessel atmosphere is presented in Table 12.2-61.

#### **12.2.2.2 Reactor Building Atmosphere**

The detailed listing of the expected airborne isotopic concentrations in the R/B atmosphere is presented in Table 12.2-61.

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

Parameter/ Assumption	Value
Reactor coolant leakage rate in normal operation	100 lb/d
Reactor coolant evaporation rate in refueling	1020 lb/h
Fraction of radioactive material to free volume	(in normal operation) 1.0(for noble gas) 0.45(others) (in refueling/shutdown) 1.0(for noble gas) 0.01(halogeniodine&tritium) 0.001(others)
Fuel defect	1%
<u>Reactor coolant specific activity in normal operation (except tritium)</u>	<u>Table 11.1-2</u>
<u>Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)</u>	<u>Table 12.2-62</u>
<u>Reactor coolant tritium specific activity</u>	<u>3.5<math>\mu</math>Ci/g</u>
<u>Reactor cavity and SFP water tritium specific activity in refueling /shutdown</u>	<u>3.5<math>\mu</math>Ci/g</u>
Low volume purge flow rate	(in normal operation) 2000 cfm
High volume purge flow rate	(in refueling) 30000 cfm
<u>Purge flow duration</u>	<u>continuous</u>

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

Parameter/ Assumption	Value
Reactor coolant evaporation rate in refueling	750 lb/h
Fraction of radioactive material to free volume	(in refueling/shutdown) 1.0(for noble gas) 0.01(halogeniodine&tritium) 0.001(others)
Fuel defect	1%
<u>Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)</u>	<u>Table 12.2-62</u>
<u>Reactor cavity and SFP water tritium specific activity in refueling /shutdown</u>	<u>3.5<math>\mu</math>Ci/g</u>
Flow rate	24000 cfm
<u>Flow duration</u>	<u>continuous</u>

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

Parameter/ Assumption	Value
Reactor coolant leak rate in refueling (Note)	100 lb/d (for Radiation Zone V to VI) 50 lb/d (for Radiation Zone IV) 2 lb/d (for Radiation Zone III)
Fraction of radioactive material to free volume	(in normal operation) 1.0(for noble gas) 0.01(halogeniodine&tritium) 0.001(others)
Fuel defect	1%
<u>Reactor coolant specific activity in normal operation (except tritium)</u>	<u>Table 11.1-2</u>
<u>Reactor coolant tritium specific activity</u>	<u>3.5<math>\mu</math>Ci/g</u>
Flow rate	1500 cfm(for Radiation Zone V to VI) 14000 cfm (for Radiation Zone IV) 76000 cfm (for Radiation Zone III)
<u>Flow duration</u>	<u>continuous</u>

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

**Table 12.2-72 Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)**

<u>Nuclide</u>	<u>Specific Activity (μCi/g)</u>	<u>Nuclide</u>	<u>Specific Activity (μCi/g)</u>
Kr-83m	1.6E-06	Ru-106	2.9E-10
Kr-85m	1.2E-05	Ag-110m	2.7E-12
Kr-85	2.9E-01	Te-125m	1.2E-09
Kr-87	7.8E-11	Te-127m	4.7E-09
Kr-88	1.3E-06	Te-127	8.6E-05
Xe-131m	2.4E-02	Sb-129	1.7E-12
Xe-133m	1.6E-02	Te-129m	1.6E-08
Xe-133	1.7E+00	Te-129	5.4E-10
Xe-135m	2.2E-04	Sb-131	-
Xe-135	1.0E-02	Te-131m	2.4E-08
Xe-138	-	Te-131	-
I-130	1.6E-07	Te-132	3.8E-07
I-131	3.8E-06	Cs-132	1.0E-07
I-132	4.6E-04	Te-133m	6.7E-16
I-133	3.2E-06	Te-133	2.7E-14
I-134	1.3E-14	Cs-134	9.3E-05
I-135	3.8E-07	Te-134	3.4E-18
Br-82	1.4E-08	Cs-135m	7.2E-15
Br-83	1.9E-10	Cs-135	2.5E-10
Br-84	-	Cs-136	2.3E-05
Rb-86	8.8E-07	Cs-137	5.3E-05
Rb-87	1.1E-14	Cs-138	6.2E-18
Rb-88	1.5E-06	Ba-140	6.0E-09
Rb-89	-	La-140	3.6E-06
Sr-89	5.2E-09	La-141	6.2E-12
Sr-90	3.4E-10	Ce-141	9.6E-10
Y-90	4.9E-06	Ce-143	5.0E-10
Sr-91	6.0E-10	Pr-143	1.0E-07
Y-91m	1.7E-07	Ce-144	7.3E-10
Y-91	1.4E-09	Pr-144	1.6E-04
Sr-92	4.2E-12	Pm-147	8.2E-11
Y-92	9.4E-10	Sm-147	6.8E-20
Y-93	1.3E-10	Eu-154	7.6E-12
Zr-93	1.3E-15	Na-24	3.5E-08
Zr-95	9.9E-10	Cr-51	1.0E-08
Nb-95m	9.7E-09	Mn-54	7.1E-09
Nb-95	1.5E-07	Mn-56	5.6E-10
Mo-99	9.5E-07	Fe-55	6.8E-09
Tc-99m	6.2E-04	Fe-59	1.2E-09
Tc-99	3.0E-12	Co-58	1.6E-08
Mo-101	-	Co-60	2.4E-09
Tc-101	-	Zn-65	2.0E-09
Ru-103	8.2E-10		
Rh-103m	2.3E-05		

Docket No. 52-021  
MHI Ref: UAP-HF-09047

Enclosure 5

UAP-HF-09047  
Docket No. 52-021

Response to Request for Additional Information  
No. 144-1738 Revision 1

February, 2009

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No. 52-021**

**RAI NO.:** NO. 144-1738 REVISION 1  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-12**

10 CFR 20.1101(b), 1301 and 10 CFR 20.2203, requires licensees to control occupational and public radiation exposure. 10 CFR 20.1301 requires compliance with the provisions of 40 CFR 190. NUREG-0800 Chapter 11.4 "Solid Waste Management System", notes that 10 CFR 20.1301(e) requires that facilities licensed by the U.S. Nuclear Regulatory Commission comply with the provisions of 40 CFR Part 190. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes

**Question 1:**

The APWR DCD Section 12.2.1.1.10 notes that the Refueling Water Auxiliary Storage tank (RWAST) and the Primary Water Makeup Water Storage Tanks (PWMUST) concrete tank walls will provide shielding that results in surface dose rates of less than 0.25 mrem/h. Table 12.3-1 "Thicknesses of Concrete walls that enclose the major components" which list the thickness of the concrete shielding surrounding components containing radioactive material, does not mention the PWMST, the RWAST or the Boron Recycle Evaporator Feed Demineralizer.

In accordance with RG 1.206, please revise chapter 12 Table 12.3-1 to include the shielding information for the Refueling Water Auxiliary Storage Tank, the Primary Water Makeup Water Storage Tanks (PWMUST) and the Boron Recycle Evaporator Feed Demineralizer.

**ANSWER:**

The Refueling Water Storage Auxiliary Tank (RWSAT) and the Primary Makeup Water Tanks (PMWSTs) will be located outside of the Auxiliary Building as shown in Figure 12.3-1 (Sheet 1 of 34) of the DCD. The RWSAT receives the spent fuel pit (SFP) water and the PMWSTs receive the distilled water from the Boron Recycle System.

Initially, it was planned to install a concrete wall around these tanks. However, instead of adopting the concrete shielding, it was decided to adopt procedures that will decrease the radioactivity level of these tanks. To be more specific, the activity of the tank water is to be controlled so that the dose rates will be less than 0.25 mrem/h (Zone I level) at 2 meters from the surface of the tank. Furthermore to prevent proximity to the tank, barriers such as fences, which

will enclose the tank at 2 meters from the surface of the tank, are to be adopted. Consequently, the shielding walls for the RWSAT and the PMWST have become unnecessary in the shielding design.

The RWSAT stores a portion of SFP water and the SFP water is to be transferred to the RWSAT after purification by the SFP demineralizer. For the PMWST, if the activity level of the distilled water processed by the Boron Recycle System will be as high so that it may cause the dose rates to be more than 0.25 mrem/h at 2 meters from the body of the tank, the water is not to be stored in the PMWST, but to be transferred and processed by the Liquid Waste Management System. Radioactivities and source strengths of the RWSAT and the PMWST which satisfy the radiation level of Zone I at 2 meters from the surface of the tank are shown as below.

<b>Refueling Water Storage Auxiliary Tank activity</b>	
Nuclide	Activity ( $\mu\text{Ci}/\text{cm}^3$ )
Co-60	1.8E-04
<b>Refueling Water Storage Auxiliary Tank source strength</b>	
Gamma Ray Energy (MeV)	Source Strength ( $\text{MeV}/\text{cm}^3/\text{sec}$ )
0.015	1.1E-05
0.3	1.6E-04
0.8	4.2E-04
1.0	6.8E+00
1.5	1.0E+01
2.0	1.5E-04
3.0	7.4E-07

<b>Primary Makeup Water Tank activity</b>	
Nuclide	Activity ( $\mu\text{Ci}/\text{cm}^3$ )
Co-60	2.2E-04
<b>Primary Makeup Water Tank source strength</b>	
Gamma Ray Energy (MeV)	Source Strength ( $\text{MeV}/\text{cm}^3/\text{sec}$ )
0.015	1.4E-05
0.3	1.9E-04
0.8	5.1E-04
1.0	8.3E+00
1.5	1.2E+01
2.0	1.8E-04
3.0	9.0E-07

In order to clarify this, Table 12.2-50 and 12.2-51 in the DCD will be modified as above.

The B.A. evaporator feed demineralizer is located on the ground floor in the Auxiliary Building. The wall thicknesses for the B.A. evaporator feed demineralizer room are shown as below.

Elevation	Room Name	North	East	South	West	Floor	Ceiling
Auxiliary Building							
3'-7"	B.A. evaporator feed demineralizer room	2'-0"	4'-0"	3'-4"	3'-4"	3'-2"	4'-2"

**Impact on DCD**

The DCD Subsection 12.2.1.1.10, Tables 12.2-50, 12.2-51 and 12.3-1 will be revised as described in the Attachment 5.

**Impact on COLA**

There is no impact on the COLA.

**Impact on PRA**

There is no impact on the PRA.

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#### 12.2.1.1.10 Miscellaneous Sources

The principal sources of activity outside the buildings include the following:

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

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

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

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

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

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

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

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

#### 12.2.1.2 Sources for Shutdown

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

Table 12.2-50 Miscellaneous Sources – Refueling Water Storage Auxiliary Tank

Refueling Water Storage Auxiliary Tank activity	
Nuclide	Activity ( $\mu\text{Ci}/\text{cm}^3$ )
Co-60	<u>5.71.8E-043</u>
Refueling Water Storage Auxiliary Tank source strength	
Gamma Ray Energy (MeV)	Source Strength ( $\text{MeV}/\text{cm}^3/\text{sec}$ )
0.015	<u>3.51.1E-0405</u>
0.3	<u>4.81.6E-0304</u>
0.8	<u>1.34.2E-0204</u>
1.0	<u>2.16.8E+0200</u>
1.5	<u>3.21.0E+0201</u>
2.0	<u>4.61.5E-0304</u>
3.0	<u>2.37.4E-0507</u>

Table 12.2-51 Miscellaneous Sources – Primary Makeup Water Tank

Primary Makeup Water Tank activity	
Nuclide	Activity ( $\mu\text{Ci}/\text{cm}^3$ )
Co-60	9.52.2E-04
Primary Makeup Water Tank source strength	
Gamma Ray Energy (MeV)	Source Strength (MeV/cm <sup>3</sup> /sec)
0.015	5.81.4E-05
0.3	8.11.9E-04
0.8	2.15.1E-0304
1.0	3.58.3E+0100
1.5	5.31.2E+01
2.0	7.1.8E-04
3.0	3.89.0E-0607

**Table 12.3-1 Thicknesses of Concrete walls that enclose the major components  
(Sheet 4 of 4)**

Elevation	Room Name	North	East	South	West	Floor	Ceiling
Auxiliary Building							
3'-7"	Mixed bed demineralizer Room	3'-4"	3'-4"	3'-4"	4'-8"	3'-2" <sup>1)</sup>	4'-8"
3'-7"	Cation-bed demineralizer Room	3'-4"	2'-10"	2'-10"	4'-0"	2'-10"	4'-8"
3'-7"	Spent fuel pit demineralizer Room	2'-10"	2'-0"	2'-0"	3'-4"	3'-2"	3'-4"
3'-7"	Valve Area <sup>2)</sup>	2'-10"	4'-2"	4'-2" <sup>3)</sup>	3'-4" <sup>4)</sup>	3'-2" <sup>5)</sup>	4'-8"
3'-7"	A-Reactor coolant filter Room	2'-0"	2'-8"	2'-0"	2'-0"	2'-8"	2'-2"
3'-7"	B-Reactor coolant filter Room	2'-0"	2'-8"	2'-0"	2'-0"	2'-8"	2'-2"
3'-7"	A-Spent fuel filter Room	1'-6"	2'-2"	2'-0"	2'-0"	2'-8"	2'-2"
3'-7"	B-Spent fuel filter Room	1'-6"	2'-2"	1'-6"	2'-0"	2'-8"	2'-2"
3'-7"	A,B-Waste demineralizer Room	2'-6"	2'-5"	2'-5"	3'-9"	3'-3"	3'-9"
3'-7"	C,D-Waste demineralizer Room	3'-4"	3'-9" <sup>3)</sup>	2'-6"	3'-9"	2'-10"	3'-9"
3'-7"	Valve Area <sup>6)</sup>	2'-10"	2'-8"	2'-8"	2'-5" <sup>7)</sup>	2'-8" <sup>3)</sup>	3'-4"
3'-7"	Waste Mobile Systems	2'-8"	--- <sup>8)</sup>	2'-8"	3'-4"	4'-1"	4'-0"
<u>3'-7"</u>	<u>B.A. evaporator feed demineralizer Room</u>	<u>2'-0"</u>	<u>4'-0"</u>	<u>3'-4"</u>	<u>3'-4"</u>	<u>3'-2"</u>	<u>4'-2"</u>
15'-9"	Piping Area <sup>9)</sup>	2'-6"	3'-4"	2'-1"	3'-4" <sup>3)</sup>	2'-8"	3'-4"
15'-9"	Hold up Tank Piping Area	2'-8"	4'-8" <sup>4)</sup>	3'-4"	2'-8" <sup>3)</sup>	2'-8"	3'-4"
15'-9"	Hold up Tank Valve Area	3'-4"	4'-8" <sup>4)</sup>	3'-4"	4'-0"	2'-8"	3'-4"
25'-3"	Steam generator blowdown demineralizer Room	3'-4"	2'-8"	2'-6"	2'-3" <sup>3)</sup>	3'-4"	2'-3"
25'-3"	Valve Area <sup>10)</sup>	3'-4"	1'-6"	1'-11"	2'-4"	3'-4"	1'-10"

1) Face to Spent resin storage tank Room

2) Adjacent to Mix bed demineralizer Room

3) Face to area of Zone III

4) Face to A,B-Mix bed demineralizer Room

5) Face to Valve Area

6) Adjacent to Waste demineralizer Room

7) Face to A,B-Waste demineralizer Room

8) Removable Shield is to be used, if necessary

9) East side of demineralizer Rooms

10) Adjacent to Steam generator blowdown demineralizer Room

Docket No. 52-021  
MHI Ref: UAP-HF-09047

Enclosure 6

UAP-HF-09047  
Docket No. 52-021

Response to Request for Additional Information  
No. 145-1736 Revision 0

February, 2009

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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2/6/2009

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

**RAI NO.:** NO. 145-1736 REVISION 0  
**SRP SECTION:** 12.02 – Radiation Sources  
**APPLICATION SECTION:** 12.2  
**DATE OF RAI ISSUE:** 1/9/2009

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**QUESTION NO.: 12.02-13**

10 CFR 20.2001, 10 CFR 61.55, Waste Form Technical Position, Rev. 1, RG-1.206 C.I.12.2.1

10 CFR 20.2001 requires that wastes be disposed of in accordance with 10 CFR 61. 10 CFR 61.55 limits the concentration of radionuclides contained in waste. Waste Form Technical Position, Revision 1, provides guidance on acceptable methods for demonstrating compliance with the waste form structural stability requirements of 10 CFR Part 61. Regulatory Position 3 "Radiation Stability of Organic Ion-Exchange Resins" of this guidance, notes that to ensure that organic ion exchange resins will not undergo adverse degradation effects from radiation, resins should not be generated having loadings that will produce greater than 1E+8 Rads total accumulated dose. For Cs-137 and Sr-90 a total accumulated dose of 1E+8 Rads is approximately equivalent to a 10 Ci/ft concentration in resins in the un-solidified, as-generated form. The guidance contained in Regulatory Guide 1.206 section C.I.12.2.1 notes that the applicant is to provide the models, parameters and bases used to calculate source magnitudes, including isotopic composition and the bases for all values.

**Question 1:**

The APWR DCD Section 12.2.1.1 notes that the design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power. The isotopic concentrations listed in Table 12.2-48 "Solid Waste Management System Radiation Sources Spent Resin Storage Tank Activity" result in activity concentrations that are higher than the activity guidance provided in the Waste Form Technical Position Revision 1.

In accordance with RG 1.206, please revise chapter 12 to provide the bases, methods, parameters and assumptions, justifying that the stated activity level will not cause resin degradation from radiation exposure or decay heat generated in the resin.

**ANSWER:**

The activity concentration and dose strength of the resin contained in the Spent Resin Storage Tank (SRST) as described in DCD Chapter 12 are the shield design dose strength determined according to Regulatory Guide 1.206 section C.I.12.2. Two SRSTs are installed in the A/B

building, which are used for temporary storage of spent resin from the demineralizers until it is discharged outside of the site. One unit is to receive high level radioactive spent resin from the demineralizers of the CVCS system (mixed bed demineralizer, cation bed demineralizer, B.A. evaporator feed demineralizer and deborating demineralizer), while the other unit is to accept low level radioactive spent resin from the Liquid Waste Management System (LWMS) and Spent Fuel Pit Cooling and Purification System (SFPCS).

The shielding requirement for the SRST uses the activity determined for the high level radioactive spent resin of the CVCS system as described in the DCD Chapter 12. The activity concentration of Cs-137 shown in Table 12.2-48 of  $2.4E+04 \mu\text{Ci}/\text{cm}^3$  is five times greater than the upper limit concentration of Cs-137 of  $4,600 \text{ Ci}/\text{cm}^3$  which is in Table 2 in 10 CFR 61.55. Note that this shield design assumes 1% fuel defects, and it is understood that the activity concentration determined by this assumption is significantly higher than that of the spent resin actually discharged. Mitsubishi-supplied PWR plant operation experience shows almost no fuel leakage, which means that the design basis of using an average value of 1% fuel defects for long term operation (60 years) will have excessive margin. Even when using this excessive margin value of 0.1% fuel defects, the activity concentration of spent resin would be about 1/10th the design basis value (approximately  $2400\mu\text{Ci}/\text{cm}^3$ ), well below even the Class A waste concentration shown in Table 2 of 10 CFR 61.55. Therefore, near surface disposal can be allowed.

Furthermore, since the activity concentration of the resin will be measured and checked at the stage of transfer of spent resin from the SRST to the high integrity container loading for disposal, so near surface disposal poses no special problem from the viewpoint of actual plant operation.

#### **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.