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May 7, 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-09226

Subject: MHI's Response to US-APWR DCD RAI No. 262-1972 Revision 1

References: 1) "Request for Additional Information No. 262-1972 Revision 1, SRP Section: 12.03-12.04 – Radiation Protection Design Features, Application Section: 12.3," dated March 3, 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 eight RAIs contained within Reference 1 and 2.

The enclosed document is being submitted in two versions. One version (Enclosure 1) includes certain information, designated pursuant to the Commission guidance as sensitive unclassified non-safeguards information, referred to as security-related information ("SRI"), that is to be withheld from public disclosure under 10 C.F.R. § 2.390. The information that is SRI is identified by brackets. The second version (Enclosure 2) omits the SRI and is suitable for public disclosure. In the public version, the SRI is replaced by the designation "[Security-Related Information - Withheld Under 10 CFR 2.390]".

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,

Y. Ogata

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

Enclosure:

1. Response to Request for Additional Information No.262-1972 Revision 1 (SRI included version)
2. Response to Request for Additional Information No.262-1972 Revision 1 (SRI excluded version)

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CC: J. A. Ciocco
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Enclosure 2

UAP-HF-09226
Docket No. 52-021

Response to Request for Additional Information
No. 262-1972 Revision 1

May, 2009
(Security-Related Information Excluded)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**3/3/2009**

QUESTION NO.: 12.03-12.04-13

10 CFR 50 Criterion 64 "Monitoring radioactivity releases" states, in part, that means shall be provided for monitoring the spaces containing components for recirculation of loss-of-coolant accident fluids, and for anticipated operational occurrences.

10 CFR 20.1501(b) requires licensees to ensure that the instruments and equipment used for quantitative radiation measurements are calibrated periodically for the radiation measured.

Regulatory Guide 1.97 Rev-3 1983 notes that the applicant is to provide rate monitors for areas inside buildings containing components for the recirculation of loss-of-coolant accident fluid where access is required to service equipment important to safety. IEEE 497-2002 contains functional and design requirements for accident monitoring instrumentation for nuclear plants and RG 1.97 endorses IEEE 497-2002.

NUREG-0800 12.3 calls for the provision of Area and Airborne Radiation Monitoring equipment meeting the specifications of RG-1.97.

QUESTION 8866-Q1

Following an accident some areas in the RB and AB, such as the RHR & SI pump areas, the RHR heat exchangers and associated pipe areas, Boric Acid Transfer Pumps, CVCS HUT and WHUT will have elevated dose rates and require access. Other areas in the A/B such as the Waste Gas Compressors the Boric Acid Evaporator the Boric Acid Transfer pump the Solid Waste Dewatering Area) and the Charging Pumps will subject to large transient dose rates and will also require access during AOO or operational transients. Additionally, the Area for the Future Mobile System, which is adjacent to a major travel path to the RB and is unshielded on one side of the travel path, does not have a radiation monitor.

Contrary to the guidance noted, APWR Tier 2 Table 12.3-4 "Area Radiation Monitors" does not show radiation monitors at the locations discussed above.

In accordance with GDC 64, and NUREG-0800 Section 12.3, please modify Table 12.3-4 to include area monitors at these locations or provide the specific alternative approaches used and the associated justification.

QUESTION 8866-Q2

APWR FSAR Tier 2 section 12.3.4.2 and Table 12.3-5 "Airborne Radioactivity Monitors" includes radiation monitors for spaces containing components for recirculation of loss-of-coolant accident fluids. FSAR section 3.1.6.5 notes that radioactivity levels in the plant environs are continuously monitored during accident conditions by the plant Radiation Monitoring Systems. Some of the spaces monitored by these radiation detectors contain active components, which may require personnel access for equipment monitoring and condition assessment. In addition, some of the monitors noted in this table provide MCR alarms that require manual operator action. Contrary to the guidance in IEEE-497 sections 4.1 and 4.5, FSAR section 12.3.4.2 and Table 12.3-5 do not provide the type classification of these monitors.

In accordance with Criterion 64, NUREG-0800 12.3 and RG-1.97, please modify FSAR Tier 2 section 12.3.4.2 and Table 12.3-5 to include type classification and the basis for the classifications selected or provide the specific alternative approaches used and the associated justification.

QUESTION 8866-Q3

APWR FSAR Tier 2 section 12.3.6 "Combined License Information" COL 12.3(1) only addresses portable air sampling equipment during accident conditions, while Table 7.5-3 "PAM Variables" notes that in addition to portable air sampling equipment, portable survey instruments used for dose rate and radioactivity measurements. Contrary to the information provided in Table 7.5-3, FSAR Tier 2 section 12.3.6 does not address the Type Classification requirements for portable dose rate and activity monitoring instrumentation. IEEE-497 notes that portable equipment used as accident monitoring instrumentation shall meet the criteria for the applicable variable type.

In accordance with NUREG-0800 12.3 and RG-1.97, please modify FSAR Tier 2 section 12.3.6 to include portable dose rate monitoring instruments and to specify the type classification requirements portable for dose rate and activity measuring instrument or provide the specific

QUESTION 8866-Q4

The APWR FSAR Tier 2, section 12.3.4, "Area and Airborne Radioactivity Monitoring Instrumentation", states that radiation instrumentation assures compliance with 10CFR20. Chapter 12 only notes the types of sources that will be used for calibration. Contrary to the guidance provided in RG-1.206 C.I.12.3.4, FSAR Section 12.3 does not provide any information regarding the calibration methods, calibration frequencies and their bases.

In accordance with RG 1.206, provide information on the calibration methods, frequency and their bases, of the installed area and airborne monitors or provide the specific alternative approaches used and the associated justification.

QUESTION 8866-Q5

The APWR FSAR Tier 2, Table 12.3-4 "Area Radiation Monitors", states that Containment High Range Area Radiation Monitors are Safety Related. In addition, Table 7.3-3 "Engineered Safety Features Actuation Signals" notes that the MCR Outside Air Intake Airborne Radiation Monitors provide an ESF MCR ventilation isolation signal. FSAR section 15.6.2, regarding the SBLOCA outside containment, notes that the charging flow is sufficient to maintain Pressurizer level, so an automatic ESF ECCS signal would not be generated, but the filtration system for the MCR is still credited for limiting dose to the operators. The Containment High Range Radiation Monitors provide ESF actuation signal for containment purge isolation. RG-1.29 Rev 4 C.1 notes that all equipment needed to maintain the control room within safe habitability limits for personnel or that could result in off site dose should be classified as Seismic Category I. Contrary to this guidance, Item 48 "Radiation Monitoring System" in FSAR Tier 2 Table 3.2-2 "Classification of Mechanical and Fluid Systems, Components, and Equipment" is blank.

In accordance with RG 1.29 Rev 4, RG-1.97 and IEEE-497-2002, please clarify the Classification and Type information provided in Chapters 12, 3 and 7, including the basis for these determinations, or provide the specific alternative approaches used and the associated justification.

QUESTION 8866-Q6

APWR FSAR Tier 2 Reference 12.3-25 References ANSI-N13.1-1997. This should be ANSI-N13.1-1999.

Please revise Reference 12.3-25 to provide the correct revision date.

QUESTION 8866-Q7

Contrary to NUREG-0800 which states that each location have local audible alarm and that monitors located in high noise areas should also have visual alarms, the APWR FSAR section 12.3.4 "Area Radiation and Airborne Radioactivity Monitoring Instrumentation", states that all airborne radioactivity monitors have no local annunciation.

In accordance with NUREG-0800 Section 12.3 and RG-8.8 C.2.g, please provide revise FSAR Tier 2 section 12.3.4 to change the radiation monitor specifications, or provide the specific alternative approaches used and the associated justification.

QUESTION 8866-Q8

APWR FSAR Tier 2 Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 7 of 15) specifies the use of RG-1.97 Rev-4. Table 1.9.2-7 notes that plants licensed after June 2007 should reference RG-1.97 Rev-4. Contrary to the specification of the use of RG1.97 Rev-4, APWR FSAR Reference 12.3-23 and FSAR Table 1.9.2-14, reference RG-1.97 Rev-3. In addition, Chapter 12 does not contain a reference to IEEE-497-2002.

Please modify APWR FSAR Tier 2 chapters 1 and 12 to remove these inconsistencies.

ANSWER:

ANSWER to QUESTION 8866-Q1:
As committed to in DCD Section 12.3.4.1.2, portable Area Radiation Monitors (ARMs) will be used prior to entering and performing work in the RHR pump and heat exchanger areas. Prior to entering the other areas which have elevated dose rates and require access, such as the SI pump area, portable ARMs will also be used, if necessary.

In addition, removable shielding is to be used in the area of the waste mobile system, if necessary.

Thus, no additional permanent ARMs are considered necessary.

ANSWER to QUESTION 8866-Q2:
The Airborne Radioactivity Monitors included in DCD Section 12.3.4.2 and Table 12.3-5 are located in HVAC exhaust ducts within Radiological Controlled Areas where radioactivity may normally exist. The design objectives of these radiation monitors are to measure the airborne radioactivity in the HVAC exhaust ducts of the air exhausted from cubicles and to provide indication of an abnormal release of radioactive material from cubicles. Section 3.1.6.5.1 of the DCD further states that the plant Radiation Monitoring Systems continuously monitor the radioactivity levels in the facility effluent and discharge paths and in the plant environs during

normal and accident conditions, in accordance with 10 CFR 50 Criterion 64.

HVAC exhaust ducts within Radiological Controlled Areas are connected to the vent stack. Plant Vent Radiation Gas Monitors (normal range monitor, accident mid range monitor and accident high range monitor) are installed to measure the concentration of radioactive gases released through the vent stack described in DCD Section11.5.2.4 and Table 11.5-3. These monitors are Type E PAM instruments (see DCD Table 7.5-3).

Therefore, the Airborne Radioactivity Monitors described in DCD Section 12.3.4.2 and Table 12.3-5 are not PAM instruments.

ANSWER to QUESTION 8866-Q3:
Section 12.3.4 of the DCD will be revised to incorporate the comment in QUESTION 8868-Q3.

ANSWER to QUESTION 8866-Q4:
The calibration methods and the associated calibration frequencies of the area and airborne monitoring instrumentation are deferred to the detailed design phase. During the detailed design phase, the regulatory criteria used to establish the design will be incorporated into the procurement specifications and will be used to evaluate the vendor designs for the area and airborne radiation monitors. Design documents will be obtained from the selected vendor including design and fabrication drawings, calculations, bill of materials, reports, vendor specific inservice test procedures, etc. From these documents, the specific parameters for the acceptance test, calibrations, and inservice tests of the monitoring instrumentation will be evaluated and the calibration methods and the associated calibration frequencies will be developed.

The Operational Radiation Protection Program, as described in DCD Section 12.5, will be developed by the COL applicant to assure that the radiation exposures are within the limits of 10 CFR 20 and follow the ALARA principle. This program will include provisions for the definition and description of the monitoring instrumentation and equipment as well as procedures on methods to maintain exposures ALARA

ANSWER to QUESTION 8866-Q5:
For maintaining the control room within safe habitability limits, MCR outside air intake airborne radiation monitors provide an ESF MCR ventilation isolation signal. These radiation monitors are part of the I&C system and are described in Appendix 3D as indicated in DCD Section 3.2.1.2. In Table 3D-2 of Appendix 3D, on sheets 12 and 13, the MCR outside air intake radiation monitors are shown as Seismic Category I, along with other I&C equipment.

Item 48 of DCD Tier 2 Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment", on sheet 51, indicates that for the Radiation Monitoring System, the piping and valves between and including the containment isolation valves are specified as Seismic Category I. This is consistent with the information presented in DCD Table 12.3-4 which lists the Containment High Range Area Radiation Monitors as safety-related and as performing the control function of containment ventilation isolation. It is additionally consistent with DCD Table 7.3-3, on sheet 1, which shows that the Containment High Range Area Radiation Monitor provides an ESF actuation signal for Containment Purge Isolation.

Therefore, this categorization of the Radiation Monitoring System piping and valves as Seismic Category I is in accordance with the requirements of RG 1.29 Rev 4 Regulatory Position C. The information presented in Item 48 of DCD Table 3.2-2, on sheet 51, for the components of the Radiation Monitoring System regarding equipment class, location, quality group, codes and standards met, and seismic category is correct.

ANSWER to QUESTION 8866-Q6:
Reference 12.3-25 of the DCD will be revised to incorporate the comment in QUESTION 8866-Q6.

ANSWER to QUESTION 8866-Q7:
Whole of DCD Subsection 12.3.4.2.5 will be revised to incorporate the comment in QUESTION 8866-Q7.

ANSWER to QUESTION 8866-Q8:
Reference 12.3-23 of the DCD will be revised to incorporate the comment in QUESTION 8866-Q8. IEEE-497-2002 will be added to Reference 12.3-28 to incorporate the comment in QUESTION 8866-Q8. The mention of RG-1.97 Rev-3 in DCD Table 1.9.2-14 is an excerpt from SRP 14.3.9 which says to use Rev-3 or Rev-4 of RG-1.97. Since DCD Table 1.9.2-7 states that Rev-4 will be used for plants obtaining an operating license after June 2006, DCD Chapter 1 does not need to be modified.

Impact on DCD
According to the ANSWER to QUESTION 8866-Q3, DCD Subsection 12.3.4 will be revised as follows:

The PAM are described in Chapter 7, Section 7.5. The portable dose rate and activity monitoring instrumentations are Type E PAM.

According to the ANSWER to QUESTION 8866-Q6, DCD Subsection 12.3.7 will be revised as follows:

12.3-25 Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of nuclear facilities. American National Standard Institute N13.1-~~1997~~ 1999.

According to the ANSWER to QUESTION 8866-Q7, DCD Subsection 12.3.4 will be revised as follows:

12.3.4.2.5 Local Annunciation

All airborne radiation monitors have ~~no~~ local annunciation consisting of an audible alarm and a warning light at the local readout.

According to the ANSWER to QUESTION 8866-Q8, DCD Subsection 12.3.4 will be revised as follows:

The ARMS and Airborne Radioactivity Monitoring System supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and assure compliance with the personnel radiation protection requirements of 10 CFR 20 (Reference 12.3-2), 10 CFR 50 (Reference 12.3-7), 10 CFR 70 (Reference 12.3-21), and the guidelines of RGs 1.21 (Reference 12.3-22), 1.97 (Reference 12.3-23), 8.2 (Reference 12.3-24), and 8.8 (Reference 12.3-1), ~~and~~ ANSI N13.1-1999 (Reference 12.3-25), and IEEE 497-2002 (Reference 12.3-28).

According to the ANSWER to QUESTION 8866-Q8, DCD Subsection 12.3.7 will be revised as follows:

12.3-23 Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants . Regulatory Guide 1.97, Rev. ~~3~~ 4, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-28 IEEE Standards, IEEE 344-1975, IEEE 308-1974, IEEE-497-2002

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**3/3/2009**

QUESTION NO.: 12.03-12.04-14

10 CFR 20 requires licensees to keep occupational doses ALARA, that is to make every reasonable effort to maintain exposures as far as possible below regulatory limits, taking into account the state of technology. NUREG-0800 and RG 8.19 note that the Applicant should identify features were implemented for reducing dose, as a result of the analysis.

Question Q1

APWR FSAR Section 12.4.1 Tier 2 notes that dose reduction results from elimination of high maintenance components in waste processing equipment and the use of advanced technology in the refueling process, however, contrary to NUREG-0800 and RG-8.19, APWR FSAR Section 12.4 does not provide any detailed information regarding the ALARA initiatives adopted because of the Dose Assessment.

In accordance with NUREG-0800 and RG-8.19, please revise FSAR Section 12.4 to provide the ALARA improvements made because of the Dose Assessment, or provide the specific alternative approaches used and the associated justification.

Question Q2

APWR FSAR Section 12.4.1.5 Tier 2 notes that ASME specifies the ISI inspection frequency requirements. However, there are industry standard documents that provide information regarding the use of Risk Informed PRA processes to reduce ISI frequency for cost and dose savings.

In accordance with 10 CFR 20.1101 (b), please describe how the proposed ISI program is consistent with the ALARA requirements of 10 CFR 20.1101 (b), or provide your justification for use of the process noted in section 12.4.1.5.

Question Q3

APWR FSAR Tables 12.4-1 through 12.4-6 Tier 2, provide information regarding exposure estimates associated with routine activities during normal operation. Some of the assumptions and parameters noted in these tables do not appear to be consistent with expected plant operations, including:

- Activity frequencies are not consistent with normal operation, for instance, this table indicates daily inspections of components located in the containment building, such as Accumulators, Containment Cooling System and Pressurizer Valves.
- The dose rates associated with some components, such as Pressurizer Spray Valves, appear to differ significantly from industry norms.
- Exposure estimates associated with some potentially radiologically significant routine equipment such as the operation and maintenance of the Boric Acid Evaporator has been omitted.
- In some cases, the number of people performing the activity is inconsistent with the expected shift manning (e.g. 2 chemist or RP personnel, when there are 3 shifts/day)
- The description of some activities are unclear, such as "Decompositions of valves".

In accordance with NUREG-0800 and RG-8.19, please revise FSAR Section 12.4 to provide exposure estimate information that is consistent with expected normal operation of the plant, or provide the specific alternative approaches used and the associated justification.

ANSWER:

ANSWER to Question Q1:
DCD Section 12.4.1 discusses the Occupational Radiation Exposure associated with the following seven activity categories: routine operations and surveillance, non-routine operations and surveillance, routine maintenance, waste processing, refueling operations, in-service inspection, and special maintenance. The general design considerations incorporated into the design of the plant facility in order to make radiation exposure ALARA include the use of conservative source terms and shielding design, enhanced nuclear fuel performance, and equipment and piping layouts and access provisions that minimize radiation exposure to plant personnel. Additional features in the facility designed to reduce the occupational radiation exposure include the labyrinth structure between the reactor vessel and the primary shielding. Furthermore, MHI has developed remotely-operated technologies which will enable certain operational activities to be performed with minimal exposure. As described in DCD Sections 12.4.1.5 and 12.4.1.6, in-service inspection of the reactor vessel and inspection of the steam generator heat-transfer tube can be handled by the ultrasonic test machine and the eddy current test machine, respectively. The US-APWR implements a long-term refueling cycle (about 24 months) which reduces the annual occupational radiation exposure by reducing the frequency of refueling operations.

Operational programs that will further incorporate the ALARA principle will be developed as part of the Operational Radiation Protection Program, described in DCD Section 12.5, and developed by the COL applicant prior to start-up. The Operational Radiation Protection Program will include procedures on all plant activities included within the seven activity categories for which dose assessments were performed.

ANSWER to Question Q2:
US-APWR DCD Tier 2 Table 12.4-6 on Occupational Dose Estimates During ISI (repeated herein in the response to NRC QUESTION Q3) shows ISI occurring every refueling (once every 24 months) as a conservative assumption. As industry standard documents for using Risk Informed PRA processes to reduce ISI frequency for cost and dose savings become available and are utilized by the industry, they will be utilized for the US-APWR as well, which is consistent with the ALARA requirements of 10 CFR 20.1101 (b). While the dose estimates in DCD Chapter 12 assume ISI occurring every 2 years, it is likely that the COL applicant will develop a plant-specific ISI program.

ANSWER to Question Q3:

The exposure estimates for routine activities during normal operations for the US-APWR are presented in DCD Table 12.4-1 "Occupational Dose Estimates During Routine Operations and Surveillance". Some of the values presented for the dose rate exposure data are taken from Regulatory Guide 8.19 *Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants – Design Stage Man-Rem Estimates*, Rev 0, 1978, Table 2 "Occupational Dose Estimates During Routine Operations and Surveillance". Since the normal operation parameters listed in DCD Table 12.4-1 are from 1978 when RG 8.19 Rev 0 became effective, some of them may not be consistent with the present industry norms. Therefore, the occupational dose information in DCD Table 12.4-1 will be revised to be consistent with the present industry norms as follows:

- The activity frequencies listed in DCD Table 12.4-1 for the inspection of the accumulators and the Containment Cooling System in the CV will be changed from daily to monthly to be consistent with the activity frequencies of the present industry norms for U.S. PWRs.

- Inspections of pressurizer valves, which are performed periodically, was included in the periodic inspection doses (refurbishing of valves) in DCD Table 12.4-3. Therefore, the pressurizer valve activity in DCD Table 12.4-1 will be deleted so as not to double count inspections of pressurizer valves.

- As described in the response to RAI No. 142, which was submitted by MHI letter UAP-HF-09047, dated February 6, 2009, the dose rate of the boric acid evaporator during maintenance is not significant, based on Japanese PWR operating experience. It is also important to note that the exposure estimates for the boric acid evaporator are included in those listed for the boric acid makeup system in DCD Table 12.4-1.

The changes described above will affect the occupational dose estimates during routine operations and surveillance. As a result, the corresponding entry in Table 12.4-8 will be updated to match this new estimate for that item. The total annual dose estimate in DCD Table 12.4-8, which is also stated in DCD Section 12.4, will be revised to reflect the new sum.

For the expected shift manning, a footnote will be added to DCD Table 12.4-1 to show the number of shifts per day, which is three. The number of workers means the number for that particular frequency. For example, there are two workers performing routine patrols during each shift (so a total of six workers per day).

In addition, the description of some of the activities in the relevant tables will be revised for clarification. For example, "decomposition of valves" will be replaced with "refurbishing of valves". This change affects DCD Tables 12.4-1 and Table 12.4-3.

Impact on DCD

DCD Tables 12.4-1, 12.4-3, and 12.4-8 will be revised as described in the above response. The revised tables are shown below. It is important to note that changes to some Tables 12.4-1 through 12.4-8 were also recommended as part of the response to RAI No. 90, which was submitted by MHI letter UAP-HF-08274, dated November 26, 2008. The tables shown below also include those recommended changes.

The reference to Table 12.4-8 in the seventh paragraph of DCD Section 12.4 will be revised as follows:

Table 12.4-8 is a summation of the US-APWR's individual dose estimates in person-rem/year. The total annual station exposure is ~~72.63~~ 71.03 person-rem, which is substantially less than the 100 person-rem annual value cited in NUREG -0713 (Reference 12.4-2).

Table 12.4-1 Occupational Dose Estimates During Routine Operations and Surveillance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
~~Walking~~ Routine patrols	0.2	0.5	2	1/shift[*1]	**0.22**
Checking:					
Containment cooling system	1	1	1	1/~~day~~ month	~~**0.36**~~ **0.01**
Accumulators	1.5	1	1	1/~~day~~ month	~~**0.54**~~ **0.02**
~~Pressurizer valves~~	~~10~~	~~0.2~~	~~1~~	~~1/day~~	~~**0.73**~~
Boric acid makeup system	5	0.2	1	1/day	**0.36**
Fuel pool system	1	0.25	1	1/day	**0.09**
RHR pump	1	0.2	1	1/day	**0.07**
Total					~~**2.37**~~ **0.77**

[*1] There are 3 shifts per day.

Table 12.4-3 Occupational Dose Estimates During Routine Maintenance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
~~Mechanical:~~ Changing filters:					
Waste filter	100	0.5	1	6/year	**0.3**
Laundry filter	100	0.5	1	10/year	**0.5**
Boric acid filter	100	0.5	1	2/year	**0.1**
Mechanical:					
Mechanical components*1	10	100	5	1/2year	**2.5**
Electrical components*1	5	200	6	1/2year	**3**
~~Decompositions~~ Refurbishing of valves*1	50	40	4	1/2year	**4**
Reactor Coolant pump*1	10	100	4	1/2year	**2**
Other pumps*1	0.25	100	4	1/2year	**0.05**
Others:					
Radiation ~~management~~ surveillance by RP personnel*2	5	1	2	1/day	**3.65**
Washing / decontamination*2	1	1	3	1/day	**1.1**
Operation of equipment*2	1	1	2	1/day	**0.73**
Total					**17.93**

*1 The dose for this activity is taken from data from operating Japanese PWRs. The dose has been adjusted to account for the 24-month refueling cycle currently assumed for the US-APWR.

*2 This activity is part of routine maintenance that is done intermittently and not truly a daily activity. The total yearly dose for this activity is taken from data from operating Japanese PWRs. The remaining quantities were adjusted to match the total yearly dose given the assumption of the frequency being once per day.

Table 12.4-8 Annual Personnel Doses per Activity Categories

Category	Reference tables	Estimated Annual Person-rem Exposure
Occupational Dose Estimates During Routine Operations and Surveillance	Table 12.4-1	~~2.37~~ 0.77
Occupational Dose Estimates During Nonroutine Operations and Surveillance	Table 12.4-2	**8.61**
Occupational Dose Estimates During Routine Maintenance	Table 12.4-3	**17.93**
Waste processing	Table 12.4-4	**5.63**
Refueling	Table 12.4-5	**8.74**
Inservice Inspection	Table 12.4-6	**11.6**
Special maintenance	Table 12.4-7	**17.75**
Total		~~72.63~~ 71.03

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**3/3/2009**

QUESTION NO.: 12.03-12.04-15

The following questions arise as a result of the staff review of the **US-APWR DCD Tier1 Revision 0 to Revision 1 Change List.**

QUESTION 8688-Q1

10 CFR 20.1101(b) requires licensees to control external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. SRP Section 12.3-4 notes that the applicant is to describe the design measures provided to reduce time in radiation fields, and the provisions for potable shielding.

In the APWR FSAR Tier 2 section 12.1.2.1 3rd paragraph 4th bulleted item, the Applicant replaced "Provision of means and adequate space to use movable shielding" with "Provisions for movable shielding, such as adequate space for movable shielding". The explanation provided for this change was that it was an editorial change for clarity. However, the new wording implies that the only provision required for movable shielding is for available space, and not design methods for supporting or installing the movable shielding.

Please clarify the reason for the deletion of the word "means", and how this change remains consistent with 10 CFR 20.1101(b), ALARA requirements, and SRP Section 12.3-4 acceptance criteria, or provide the specific alternative approaches used and the associated justification.

QUESTION 8688-Q2

In the APWR FSAR Tier 2 section 12.3.4.2.2 4th paragraph, the Applicant replaced the 1st sentence "The sampling points of the airborne radioactivity monitors are shown in the Figure 12.3-10." with "The detailed sampling points of the airborne radioactivity monitors are shown in Figure 9.4.3-1." and deleted the old 2nd sentence. The explanation provided for this change was that it was an editorial change for clarity, and that it cited a more adequate figure. As part of this change the following sentence was deleted: "Details of design will be developed and, it is to be described in Combined License Information and verified it in ITAAC.". Another related change was made to FSAR Tier 2 12.3.6, which Deleted COL 12.3(2). This was also identified as an editorial change, and it was noted that that the more detailed designs are shown in Fig. 9.4.3-1. The deleted COL 12.3(2) stated that the COL Applicant is to provide more detailed air controlling process and airborne radioactivity monitor in Figure 12.3-10. However, the detail provided in

12.3-10 is equivalent to 9.4.3-1, which is insufficient for a complete evaluation of the radiation monitors.

Please provide a description of the information that is required to be provided by the COL Applicant to allow the staff to adequately assess the compliance of the plant Continuous Air Sampling equipment with the guidance provided by RG 8.8, RG 8.25 and ANSI N13.1 as noted in the SRP section 12.3-4, or provide the specific alternative approaches used and the associated justification.

QUESTION 8688-Q3

In the APWR FSAR Tier 2 Table 12.3-1 (Sheet 4 of 4) was modified to Add the concrete thicknesses of Area for future Waste Mobile Systems and added foot note 8) for the east side. Footnote 8 states that the Removable Shield is to be used, if necessary. However, the applicant does not:

- Provide information regarding what Zone Level should be used as the basis for installing Removable Shielding.
- Does not provide any criteria to the COL Applicant, for use in determining the need for removable shielding, and the required thickness of shielding to be provided.
- Does not provide a COL Action Item requiring the COL Applicant to determine if Removable Shielding should be installed in the area.
- Does not discuss the provisions provided in the design for the installation and removal of the removable shielding for the mobile waste processing system.

Please provide information that addresses the identified questions, or provide the specific alternative approaches used and the associated justification.

QUESTION 8688-Q4

Please address the following editorial observations:

1. FSAR Tier 2 section Reference 12.1-7 Deleted the old "Reference 12.1-7, However, a reference to RG 1.206 was not added to replace the deleted reference to RG 1.70. Please add a reference to RG 1.206.

2. FSAR Tier 2 section 12.3.1.2.1.3 Access Control for Personnel and Materials, states:
   (3) Regulations to be followed in the RCA:
   d. At the exit from the RCA, contamination is checked for radioactive materials on body surfaces with a radiation monitor, and persons are required to report to radiation control personnel and follow instructions if any contamination is found.

   This should read
   d. At the exit from the RCA, ~~contamination is~~ **personnel are** checked for radioactive materials on body surfaces with a radiation monitor, and ~~persons~~ **they** are required to report to radiation control personnel and follow instructions if any contamination is found.

3. FSAR Tier 2 section 12.3.4.2.2 Criteria for Location of Airborne Radioactivity Monitors, states: If the gas is detected, the existence of Iodine or other radioactive materials are to be recognized. It should state: If the gas is detected, the existence of Iodine or other radioactive materials are to be ~~recognized~~ **determined**.

4. FSAR Tier 2 section 12.1.4 4th paragraph COL 12.1(4) is deleted for an editorial change that Integrated COL 12.1(4) into COL 12.1(2). However, COL 12.1(2) was also deleted. Please revise the Revision change list to accurately reflect the purpose of the change.

ANSWER:

ANSWER to QUESTION 8688-Q1:
In this case, MHI did not intend to change the meaning of this sentence as the NRC pointed out. Therefore, MHI will fix the unintentional modification to the meaning of this sentence and revert to the original statement from Revision 0.

ANSWER to QUESTION 8688-Q2:
DCD Figure 12.3-10 is part of the system diagram for the Auxiliary Building HVAC System but does not show the entire system; consequently, it was not clear whether airborne monitors are installed at the exhaust air ducts in all the controlled areas for the Auxiliary Building HVAC System. On the other hand, DCD Figure 9.4.3-1 shows the entire system diagram for the Auxiliary Building HVAC System together with controlled areas and uncontrolled areas. It also clearly specifies whether airborne monitors are installed at the exhaust air ducts in all the controlled areas. Therefore, DCD Figure 9.4.3-1 provides the necessary detailed information. Based on this figure, airborne monitors will be installed at the exhaust air ducts in the following areas:

- Fuel Handling Area
- Annulus and Safeguard Area
- R/B
- A/B
- Sample and Lab Area

The acceptance criteria for the airborne radioactivity monitor ITAAC described in DCD Tier-1 Table 2.7.6.13-3 is that each of the as-built monitors exists. This change to the referenced figure does not affect the acceptance criteria for the airborne radioactivity monitor ITAAC.

ANSWER to QUESTION 8688-Q3:
Utilities that will utilize the US-APWR have the option of whether or not to install the Waste Mobile Systems in the future as additional equipment for the LWMS. Therefore, it is difficult to provide detailed information such as source strength and radiation level from the radioactivity contained in Waste Mobile Systems in the DCD.

On the other hand, if there is a contained source in addition to what is in DCD Section 12.2.1, COL 12.2(1) requires the COL Applicant to explain it in the FSAR. For this reason, the COL Applicant will be required to revise Section 12.2 of COLA Part 2 FSAR and explain the radioactivity concentrations of Waste Mobile Systems if this option system is included in the plant. Based on this information, it is assumed that the relevant parts of the COLA Part 2 FSAR Section 12.3 will also need to be updated.

Meanwhile, since Waste Mobile Systems are individually shielded, it is assumed that the east wall of the Waste Mobile Systems area is unnecessary. However, in consideration of an option to install Waste Mobile Systems with thin shields, footnote 8 was added to DCD Table 12.3-1 (sheet 4 of 4) to describe that removable shielding would be added, if necessary.

ANSWER to QUESTION 8688-Q4:
1. It is stated in Table 1.9.1-1 in DCD Chapter 1 Section 1.9 that all the chapters of the US-APWR DCD conform to RG-1.206. Also, RG-1.206 is stated as a reference in Subsection 1.9.7. Therefore, RG-1.206 is not explicitly stated as a reference in Chapter 12.
2. DCD Section 12.3.1.2.1.3 (3) d will be revised to incorporate the comment in Item 2 of QUESTION 8688-Q4.
3. DCD Section 12.3.4.2.2 will be revised to incorporate the comment in Item 3 of QUESTION

8688-Q4.

4. A typographical error was made in the reason for change in the change list. It should read "Integrated into COL 12.1(5)." COL 12.1(5) was added in the DCD Revision 1 as COL information related to the preparation of the Radiation Protection Program. Since the content of COL 12.1(4) is also related to the Radiation Protection Program, COL 12.1(4) is deleted in DCD Revision 1 and integrated into COL 12.1(5).

Impact on DCD

According to the ANSWER to QUESTION 8688-Q1, the 4th bulleted item in the 3rd paragraph of Subsection 12.1.2.1 will be changed again in the next revision as follows:

- Provision~~s for movable shielding, such as adequate space for~~ of means and adequate space to use movable shielding

According to the ANSWER to QUESTION 8688-Q2, the last paragraph of Subsection 12.3.4.2.2 will be revised to clarify of the statement as follows:

The sampling points of the airborne radioactivity monitors are shown in the Figure 12.3-10. The detailed ~~sampling points of~~ flow diagram of HVAC system installed the airborne radioactivity monitors are shown in Figure 9.4.3-1.

According to the ANSWER to the item 2 of QUESTION 8688-Q4, Subsection 12.3.1.2.1.3 Access Control for Personnel and Materials, (3) Regulations to be followed in the RCA: d. will be revised as follows:

At the exit from the RCA, ~~contamination is~~ personnel are checked for radioactive materials on body surfaces with a radiation monitor, and ~~persons~~ they are required to report to radiation control personnel and follow instructions if any contamination is found.

According to the ANSWER to the item 3 of QUESTION 8688-Q4, the 2nd sentence of the 2nd paragraph of subsection 12.3.4.2.2 will be revised as follows:

If the gas is detected, the existence of Iodine or other radioactive materials are to be ~~recognized~~ determined."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**3/3/2009**

QUESTION NO.: 12.03-12.04-16

QUESTION 8815-Q1

NUREG-0800 12.3-3 and Regulatory Guide 1.206 C.I.12.3.5 10, CFR 50.34(f)(2)(vii) and NUREG-0737 II.B.2, note that vital areas which require access by operators aiding, mitigating or recovering from the accident, need to be identified. NUREG-0737 II.B.2 notes that this is applicable to vital areas and equipment, other than the Control Room, such as the radwaste control stations, emergency power supplies, motor control centers, and instrument areas. These criteria are applicable even if the areas are accessed on an irregular basis and not continuous occupied for the duration of the event. Some of the areas that may reasonably be expected to require access include but are not limited to:

- RHR Pumps – - Figure 12.3-2 Sheet 1
- RHR Heat Exchangers - Figure 12.3-2 Sheet 3
- Safety Injection Pumps - Figure 12.3-2 Sheet 1
- Boric Acid Transfer Pumps - Figure 12.3-2 Sheet 1
- Waste Disposal Panel - Figure 12.3-2 Sheet 3
- SFP area - Figure 12.3-2 Sheet 8
- Plant filter areas - Figure 12.3-2 Sheet 7
- Charging Pumps - Figure 12.3-2 Sheet 1

US-APWR FSAR Tier 2 section 12.3.1.2.2 "Accident Conditions", and Figures 12.3-3 through 12.3-6 and Table 12.3-3 and Figure 12.3-2 "General Plant Arrangement with Post Accident Vital Areas" apparently only present accident exposure data for TSC, MCR and sampling activities. Insufficient information is provided to allow determination of the areas that plant operators or radiation protection personnel may need to access, as noted above, and the associated mission doses.

In accordance with 10 CFR 50 GDC-19, NUREG-0800 and RG-8.8 C.I.12.3.5, please revise section 12.3.3.1.2.2, and the associated figures and tables, to show mission paths and describe those design exposure values associated with plant operation and monitoring, for the duration of the event, or provide the specific alternative approaches used and the associated justification.

QUESTION 8815-Q2

US-APWR FSAR Tier 2 section 3.11 "Environmental Qualification of Mechanical and Electrical Equipment" notes that the equipment service times are specified in Appendix 3D. Appendix 3D, Table 3D-1 "Equipment Post-Accident Operability Times" notes that for equipment with a 2 week service requirement, that part of the evaluation criteria included consideration that the equipment is located outside containment, is accessible, and can be repaired, replaced, or recalibrated. Table 3D-2 "USAPWR Environmental Qualification Equipment List" shows a number of instruments that are located in areas exposed to radiation from system containing ESF fluids, however there is no discussion of mission doses related to these components, in chapter 12.

In accordance with 10 CFR 50 GDC-19, NUREG-0800 and RG-8.8 C.I.12.3.5, please revise section 12.3.3.1.2.2, and the associated figures and tables, to show mission paths and describe those design exposure values associated with accessing these instruments for the duration of the event, or provide the specific alternative approaches used and the associated justification.

QUESTION 8815-Q3

The US-APWR FSAR Tier 2 Section 12.3.1.2.2, "Accident Conditions" discusses vital area access for long term accident recovery. Projected dose rates and mission doses are listed in Table 12.3-3. The analysis presented is silent with respect to the assumed airborne activity concentrations, the use of respiratory protection equipment (if any), assumed protection factors (if used) to limit internal exposure, or the use of movable or temporary shielding material to limit external exposure.

Please describe in FSAR section 12.3.1.2.2 or in Table 12.3-3, the assumptions and parameters used to determine compliance with the requirements of NUREG-0737, GDC 19 and 10 CFR 50.34(f)(2)(vii), or provide the specific alternative approaches used and the associated justification.

ANSWER:

ANSWER to QUESTION 8815-Q1:
As described in DCD Section 12.3.1.2.2, the US-APWR vital areas which require personnel access within 30 days after the occurrence of an accident are limited to the main control room, technical support center, post accident sampling system, radiochemical laboratory, and hot counting room. It is not necessary to access the RHR pump room, RHR cooler room, or other rooms identified in the question. DCD Figures 12.3-3 through 12.3-6 present the radiation zone maps for times up to one month after the occurrence of an accident. The projected doses in the identified vital areas are shown in DCD Table 12.3-3 and correspond to these figures.

ANSWER to QUESTION 8815-Q2:
The instruments listed in DCD Table 3D-2 that have 2 week post-accident operational duration are the pressure transmitters for the Main Steam Line (A-D) Pressure instruments. These transmitters are located in the Main Steam Piping Area in the Reactor Building. According to RG 1.97, these instruments measure Type D variables, and are used to monitor the Steam Generators during plant operation. The repair, replacement, and recalibration of these instruments is not considered important to safety during post-accident periods. Therefore, these instruments are not considered to be in a vital area which must be accessed during a post-accident mission.

These instruments are in the Main Steam Piping Area, shown in DCD Section 12.3 in the figures at Elevation 101'-0", which is not expected to be a high radiation zone. Figure 12.3-5 (Sheet 9 of 10) shows that 1 day after the accident the area is ≤15 mrem/hr, and Figure 12.3-6 (Sheet 9 of 10) shows that 1 month after the accident the area is ≤1 mrem/hr. If any post-accident repair,

replacement, or recalibration becomes necessary, plant-specific procedures will ensure that a radiological assessment will be performed before the mission takes place. Based on the low radiation in this area, it is anticipated that a radiological assessment will show that a mission to these instruments would be allowable.

ANSWER to QUESTION 8815-Q3:
As discussed in DCD Section 12.3.4, portable instruments, including radiation monitors, will be used to accurately determine the airborne activity concentrations in plant areas where personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii)/(vii) and the criteria in Item III.D.3.3 of NUREG-0737. This is the responsibility of the COL Applicant and is specified in COL Item 12.3(1).

The COL Applicant will be developing the Operational Radiation Protection Program in order to provide policies and procedures concerning the operation of the plant which will ensure that the radiation exposures are maintained ALARA. The Operational Radiation Protection Program, as described in DCD Section 12.5, will include the definition and description of the monitoring instrumentation and equipment and of other protective equipment (e.g. portable ventilation systems, temporary shielding, etc.). It will also include procedures on radiological surveillance, dose control, and respiratory protection. The Operational Radiation Protection program will include procedures for accident conditions and post-accident recovery.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**3/3/2009**

QUESTION NO.: 12.03-12.04-17

10 CFR 20.1101(b) requires licensees to control 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 under normal and postulated accident conditions. NUREG-0800 notes that all accessible portions of the fuel transfer tube are shielded so that contact radiation levels are less than 100 Rad/h. If removable shielding is used for this purpose, then controls on the shielding and radiation monitoring instrumentation should be provided.

The APWR FSAR identifies a number of areas associated with fuel storage and handling, that either cause or have the potential to cause high dose rates in areas where people could be exposed. Please address the following questions regarding the physical access barriers, shielding and other design features provided to maintain personnel exposure ALARA.

QUESTION 8816-Q1

The APWR FSAR Tier 2 section 12.3.2.2.8 discusses the Spent Fuel Transfer Canal, and tube shielding design. This section notes that there are provisions for access and inspection. Unlike Figure 12.3-7 "Isometric View of Main Control Room Shielding", which allows the viewer to determine the arrangement of structures relative to the access points, Figure 12.3-8 "Labyrinth for radiation protection around Fuel Transfer Tube" does not clearly depict the area bounded by the shock absorber labyrinth. Insufficient information has been provided to allow the staff to determine the location of physical barriers or removable shielding provided to limit personnel radiation exposure.

Please revise FSAR chapter 12.3 to describe the design features noted in NUREG-0800, that are not apparent on Figure 12.3-8 or discussed in section 12.3.2.2.8 including, the access points, physical barriers preventing access, any shielding that is other than permanent, radiation monitor locations, or provide the specific alternative approaches used and the associated justification.

QUESTION 8816-Q2

The APWR FSAR Tier 2 section 12.3.2.2.8 discusses the Spent Fuel Transfer Canal, and tube shielding design. FSAR Figure 12.3-8 "Labyrinth for radiation protection around Fuel Transfer Tube" indicates that there is a valve operator assembly for the Fuel Transfer Tube Gate valve,

running through the 52' penetration area. FSAR Section 12.1.3.1 notes that some penetrations have been provided with an offset between the source and occupied areas. However, there is insufficient detail on FSAR Figure 12.3-8 to determine what design features ensure that no streaming can occur around or through the valve reach rod assembly. In addition, insufficient information has been provided to determine if there are any other potential streaming sources (e.g. the transfer canal drain line) in the area.

In accordance with 10 CFR 20, RG 1.206 and NUREG-0800, please revise chapter 12.3 to include information that demonstrates that potential streaming paths have been identified and the appropriate ALARA design features provided, or provide the specific alternative approaches used and the associated justification.

QUESTION 8816-Q3

The APWR FSAR Tier 2 section 12.3.2.2.8 discusses the Spent Fuel Transfer Canal and tube shielding design. A Cask Load Pit (CLP), a Fuel Inspection Pit (FIP) and the Spent Fuel Pit (SFP) are connected to the Fuel Transfer Canal (FTC). Weir Gates are installed between the FTC and the SFP, FIP and CLP. Neither chapter 9 or 12 discuss the design features provided to prevent moving an irradiated component near a weir gate connected to a drained pit. Industry experience shows that high dose rate areas result from moving irradiated material near a drained area.

In accordance with 10 CFR 20, RG 1.206 and NUREG-0800, please revise section 12.3 to describe the design features provided to prevent movement of irradiated components to areas connected to the SFP that may be empty, or provide the specific alternative approaches used and the associated justification.

QUESTION 8816-Q4

The APWR FSAR Tier 2 section 12.3.2.2.8 discusses the Spent Fuel Transfer Canal and tube shielding design. While section 9.1.2 notes that there are no drains connected to the SFP, neither chapter 9, or section 12.3.2.2.8 discuss the design features provided to prevent deliberate or inadvertent draining of the CLP or the FIP, while an irradiated fuel bundle is in the area.

In accordance with 10 CFR 20, RG 1.206 and NUREG-0800, please revise section 12.3 to describe the design features provided to prevent draining of the CLP or FIP while irradiated components are in the area or provide the specific alternative approaches used and the associated justification.

QUESTION 8816-Q5

The APWR FSAR Tier 2 section 12.3.2.2 discusses shielding designs, including shielding for fuel and irradiated material handling equipment. Industry experience shows that very high dose rate areas result from moving hollow tools or equipment over irradiated material. Insufficient information has been provided to determine if there is a possibility of very high dose rates from irradiated material located underwater.

In accordance with 10CFR 20, RG 1.206 and NUREG-0800, please revise section 12.3 to that require submersible tools to have flood ports, or provide the specific alternative approaches used and the associated justification.

QUESTION 8816-Q6

INPO SOER 85-1 "Reactor Cavity Seal Failure" notes a number of events, besides Refueling Cavity (RC) seal failure, could result in the loss of RC water level. FSAR section 9.1.4.2.2.2 "Reactor Refueling Operations" notes that the lower levels of the refueling cavity are flooded using

a fill line entering through the refueling cavity floor. Unlike the Spent Fuel Pool, which does not have any drain points below the level of the fuel, the refueling cavity appears to have a potential drain path below the potential elevation of the fuel. Chapter 12 contains insufficient information to determine the safe lay down area for the fuel bundle, and the resultant dose rate to personnel in the area because of reduced water shielding, in the event an RC leak occurred while moving an irradiated fuel bundle.

Please revise chapter 12.3 to provide the design features provide preclude an inadvertent loss of water from the RC, from other than a seal leak. Provide the safe lay down location for a fuel bundle and an estimate of the dose rate to personnel in the area. Identify any COL action items required to prevent an event (i.e. restrictions on use of nozzle dams, or valves that need to be locked closed during fuel movement). If this information is not provided then provide the specific alternative approaches used and the associated justification.

QUESTION 8816-Q7

The APWR FSAR Tier 2 section 12.3.2.2.4 "Fuel Handling Area Shielding Design", notes that the minimum depth of water above the fuel limits the dose at the water surface to less than 2.5 mrem/h for an assembly in a vertical position. FSAR Section 12.2.1.1.5 "Spent Fuel Pit Cooling and Purification System" notes that the activity of SFP water is determined assuming the presence of only Cobalt-60, present at the level specified in Table 12.2-32. The Co-60 generates a dose rate at the pit surface of up to 15 mrem/h. Table 12.4-5 only reflects the dose rate from the fuel and does not reflect the dose rate from activity entrained in the liquid, therefore the dose estimate for the refueling activity presented in table 12.4-5, is not consistent with the information provided in other sections of the FASR.

Please revise FSAR Table 12.4-5 to accurately reflect the total dose the workers will receive while engaged in fuel movement activities, or provide the specific alternative approaches used and the associated justification.

ANSWER:

ANSWER to QUESTION 8816-Q1:
The access points for accessing and inspecting the area around the fuel transfer tube will be added to DCD Figure 12.3-1 on sheets 9 and 10, as indicated in the "Impact on DCD" section of this response. Since a high dose rate is expected to be present in the area around the fuel transfer tube during fuel transfer, the area is not usually accessed during that time and access to the area is tightly controlled by means of an entrance lock. Therefore, it is not necessary to install items such as removable shielding or a permanent radiation monitor.

ANSWER to QUESTION 8816-Q2:
The valve operator assembly for the Fuel Transfer Tube gate valve is hollow, as shown in the figure in this response, and is filled with water up to the same level as the refueling cavity during fuel transfer. Therefore, due to the shielding provided by the water in the sleeve, there is no streaming from the valve operator assembly to the penetration area that would increase the dose rate of the penetration area significantly.

Like the spent fuel pool (SFP), the fuel transfer canal does not have a drain line and is instead drained by a temporary pump. The refueling cavity, however, does have a drain line, but it is located a sufficient distance away from the fuel transfer route. Consequently, there will be no potential streaming paths, which could become a problem in this area during fuel transfer.

ANSWER to QUESTION 8816-Q3:

Management of the radiation exposure for personnel working near weir gates which are connected to drained pits will be controlled by the operational procedures. During refueling, the Fuel Inspection Pit (FIP), which is along the route for moving irradiated fuel from the Reactor Vessel to the Spent Fuel Pit (SFP), is usually filled with water. Therefore, lack of shielding and high dose rates for personnel in this area would not occur. However, if an irradiated component is moved near a weir gate connected to the drained FIP or Cask Load Pit (CLP), access into the pit or near the weir gate will be administratively prohibited.

ANSWER to QUESTION 8816-Q4:
Similar to the Spent Fuel Pit (SFP), the Cask Loading Pit (CLP) and the Fuel Inspection Pit (FIP) do not have a floor drain and drained by a temporary pump. Therefore, deliberate or inadvertent draining of the CLP or the FIP while an irradiated fuel bundle is in the area will not occur.

ANSWER to QUESTION 8816-Q5:
All submersible tools or equipment for handling irradiated material underwater are designed with penetrations to ensure their filling with water which will act as shielding, thus preventing high dose rates to personnel above the pit when handling irradiated materials.

ANSWER to QUESTION 8816-Q6:
There are several fill and drain lines in the refueling cavity (RC). The drain lines at the bottom of the RC drain to the Refueling Water Storage Pit (RWSP) and are addressed in DCD Section 5.4.7.2.3.5. The drain lines that flow from the RC to the RWSP and all other paths are administratively locked after flooding and prior to fuel movement. Similarly, the fill line, which is located at an elevation one foot above the reactor flange, is administratively closed during fuel movement.

If the water level of the RC were to decrease, the low water level alarm would be announced both in the local area and in the main control room, which would result in the appropriate actions being taking to maintain the water level of the RC by adding water. Therefore, an inadvertent loss of water from the RC is precluded.

As a result, no safe lay down location or estimate of dose rates from irradiated fuel or components for an inadvertent loss of RC water is necessary.

ANSWER to QUESTION 8816-Q7:
The doses shown in DCD Table 12.4-5 are meant to be estimates of typical doses. Therefore, if operational radiation exposure is evaluated at the maximum design dose rate shown in DCD Figure 12.3-1, the result will be excessively conservative. Instead, the exposure estimate is carried out based on a more realistic actual dose rate. The fuel handling work shown in DCD Table 12.4-5 does not refer to work at the water surface of the Spent Fuel Pit (SFP) but to the work on the manipulator crane about one meter above the refueling cavity, resulting in a lower dose rate. The dose rate indicated in the table is consistent with the dose rate recorded during fuel handling activities in Japan. Consequently, the dose rate and the total dose the workers will receive while engaged in fuel handling includes the contribution from both the Spent Fuel (SF) and the contribution from the SFP water.

It is also worth noting that the pit water surface during fuel handling is controlled so that the total dose rate from both the SF and from the SFP water will be less than 15 mrem/h (within zone IV). However, the activity of SFP water (DCD Table 12.2-32) used for designing the shielding of the spent fuel pit cooling and purification system is set in a conservative manner by assuming that the dose rate at the water surface is 15 mrem/h from the SFP water alone.

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

Figure: The Valve Operator Assembly for the Fuel Transfer Tube Gate Valve

Impact on DCD

According to the ANSWER to QUESTION 8816-Q1, the DCD Figure 12.3-1 (sheets 9 and 10) will be revised as indicated in the attachment.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 9 of 34)

Reactor Building at Elevation 35'-2"

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

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 10 of 34)

Reactor Building at Elevation 50'-2"

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
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QUESTION NO.: 12.03-12.04-18

Question 1972-8817-Q1

10 CFR 20.1101(b) requires licensees to control external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. In 10 CFR 20, the definition for ALARA includes guidance to make every reasonable effort to maintain exposures below regulatory limits, taking into account the state of technology. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment.

The APWR FSAR Tier 2 12.2.1.2 "Sources for Shutdown" notes that in the reactor shutdown condition one of the significant sources requiring permanent shielding is the incore instrumentation system (ICIS). The incore instrumentation consists of movable neutron detectors (MDs) which are inserted into the core though the in-core instrumentation system nozzles located on the closure head. Figure 7.7-1 "Basic System for Insertion of Movable Neutron Detectors" indicates that at least some portion of the transit tube between the drive mechanism and the reactor head is not shielded. Industry experience shows that detectors stick in the core, and require personnel to work at the location of the movable drive system to rectify the problem. There have been personnel over exposures due to working on in core detector systems. Insufficient information is available to determine the adequacy of the shielding and personnel protective features provided for the ICIS system between the reactor head and the ICIS area, and the in the ICIS area.

In accordance with 10 CFR 20.1101(b) and RG 1.206, please update chapter 12.3 to describe the designed shielding and personnel protective features provided for work on the ICIS during an AOO, or provide the specific alternative approaches used and the associated justification.

ANSWER:

Same as the ANSWER to Question 8818-Q3 of QUESTION NO. 12.03-12.04-19, if the MDs become stuck in the core, radiation workers will have to work near the MD drive unit in order to resolve the problem. Under this condition, temporary shielding and portable ARMs will be used as necessary to protect personnel working near the MD drive unit.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 262-1972 REVISION 1**
SRP SECTION:	**12.03-12.04 – Radiation Protection Design Features**
APPLICATION SECTION:	**12.3**
DATE OF RAI ISSUE:	**3/3/2009**

QUESTION NO.: 12.03-12.04-19

RG-1.206, Part C.I.12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation, states that the applicant should describe the criteria for selection and placement of the fixed radiation monitors in accordance with ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation monitoring Systems for Light-Water Nuclear Reactors."

APWR FSAR Tier 2 section 12.3.4 "Area Radiation and Airborne Radioactivity Monitoring Instrumentation" discusses the plant Area Radiation Monitoring system (ARM). In this section, it states that the ARMS are in conformance with ANSI/ANS HPSSC-6.8.1. The information provided in this subsection of the FSAR appears to be inconsistent with the stated commitment.

Question 8818-Q1

ANSI /ANS-6.8.1 section 4.3 "Monitoring Range" notes that the selection criteria for the range should ensure that upper end of the scale is high enough to ensure that the reading remains on scale for dose rates approaching 10 times the maximum expected dose rate for Anticipated Operational Occurrences. Some of the radiation monitors listed in FSAR Table 12.3-4 "Area Radiation Monitors" may not meet these range criteria. Examples include RMS-RE-2, which is located in a Zone VII (> 1R/h & < 10 R/h) and RMS-RE-7, which is located in a Zone VI area (> 0.1R/h & < 1 R/h) but is also subjected to elevated dose rates from ICIS detectors. Another example is the MCR monitor, RMS-RE-1 located in a Zone I area (< 0.25 mrem/h), but based on the accident analysis information presented in Chapter 15, may be exposed to dose rates greater than the upper range of the instrument.

In accordance with RG 1.206, please revise section 12.3 to provide radiation monitors that meet the applicable range requirement, or provide the specific alternative approaches used and the associated justification.

Question 8818-Q2

APWR FSAR Tier 2 9A.3.1 FA1-101 "Containment Vessel" (CV) notes that there are two personnel air locks and one equipment hatch provided to access the CV. FSAR section 12.3.4.1.8 states that there is an area radiation monitor (ARM) at the containment air lock. FSAR Figure 12.3-1 "Radiation Zones for Normal Operation/Shutdown" shows an ARM at the at the Elevation

25'-3 personnel air lock but this figure does not show an ARM at the other personnel air lock or the Equipment Hatch.

ANSI/ANS-6.8.1, Section 4.2, Detector Locations, has the following criteria for locating instrumentation: "... 4.2.2 Special consideration should be given to those normally accessible and occasionally accessible areas which can experience significantly greater exposure rates resulting from operational transients or maintenance activities."

In accordance with NUREG-0800 Section 12.3, please modify Table 12.3-4 to include area monitors at these locations or discuss how workers will be alerted of increasing radiation levels in these and other areas where radiation levels could increase significantly due to accidents, transients or maintenance, and where personnel may be present.

Question 8818-Q3

The incore instrumentation consists of movable neutron detectors (MDs) which are inserted into the core though the in-core instrumentation system nozzles located on the closure head. US-APWR FSAR tier 2 Figure 7.7-1 indicates that at least some portion of the transit tube between the drive mechanism and the reactor head is not shielded. Table 12.3-4 "Area Radiation Monitors" indicates that there is one ICIS Area Radiation Monitor located in the area, but due to the detector location, it is not capable of responding to a source of radiation present in the unshielded portion of the transit tube depicted in figure 7.7-1. Industry Operating Experience indicates that movable incore detectors occasionally become stuck during transit from the core to the storage location. If these detectors stick during withdrawal, would present a high dose rate on the elevation of the containment by the unshielded portion of the MD transit tube.

ANSI/ANS-6.8.1, Section 4.2, Detector Locations, has the following criteria for locating instrumentation: "... 4.2.2 Special consideration should be given to those normally accessible and occasionally accessible areas which can experience significantly greater exposure rates resulting from operational transients or maintenance activities."

In accordance with NUREG-0800 Section 12.3, and RG-8.8 C.2.g, please modify Table 12.3-4 to include area monitors at these locations or discuss how workers will be alerted of increasing radiation levels in an area where radiation levels could increase significantly due to an Anticipated Operational Occurrence (equipment malfunction), and where personnel may be present.

ANSWER:

ANSWER to Question 8818-Q1:
The justification for the stated ranges of area radiation monitors (ARMs) RMS-RE-1, RMS-RE-2, and RMS-RE-7 are as follows;

<RMS-RE-1>
DCD Figures 12.3-3, 12.3-4, 12.3-5 and 12.3-6 demonstrate that the post-accident dose rate due to the gamma penetrated the shielding of the Main Control Room is less than 1 mrem/h. RMS-RE-1 is capable of detecting radiation dose rates up to 1 R/h (about 1000 mrem/h). Therefore, the range of this ARM is sufficient for its intended function.

<RMS-RE-2>
The function of this ARM is to continuously indicate the radiation level prior to entry into containment. If RMS-RE-2 indicates a very high radiation level (10 R/h (about 10 rem/h)), entry into containment is not allowed. Additionally, the range of this ARM is consistent with the range described in ANSI/ANS–HPSSC-6.8.1-1981 Table 2. Therefore, the range of this ARM is sufficient for its intended function.

<RMS-RE-7>
The function of this ARM is to continuously indicate the radiation level in the incore instrumentation system (ICIS) area. The ICIS consists of movable neutron detectors (MDs) which are inserted into the core though the ICIS nozzles located on the closure head. During the period of time when the MDs are moving from the closure head to the MD drive unit, the MD is an intense source of radiation and the ICIS area becomes a significantly high radiation area. However, to limit excessive radiation exposure, the radiation workers are prohibited from being in containment prior to moving the MDs. Additionally, the range of this ARM is consistent with the range described in ANSI/ANS–HPSSC-6.8.1-1981 Table 2. Therefore, the range of this ARM is sufficient for its intended function.

ANSWER to Question 8818-Q2:

The normal personnel airlock, located at floor level below the operating floor, is described in DCD Section 1.2.1.7.2.1. RMS-RE-2 is installed at the normal personnel airlock. The airlock located on the operating floor is the emergency airlock. This airlock is used only for emergency situations, so no ARM is needed. The equipment hatch is closed during operating Modes 1 to 4 and during the movement of irradiated fuel assemblies within containment, so no ARM is needed. Normal access to the containment during operation, maintenance, repair, and refueling is via the normal personnel airlock. Therefore, RMS-RE-2 is installed to continuously indicate the radiation level prior to normal entry to containment.

ANSWER to Question 8818-Q3:

As described in the ANSWER to QUESTION 8818-Q1 above, during the period of time when the movable neutron detectors (MDs) are moving from the core to the MD drive unit, the MD is an intense source of radiation and the ICIS area becomes a significantly high radiation area. RMS-RE-7 is installed near the ICIS drive unit in order to assure that the instrument reading response will indicate the MDs approach to the MD drive unit. However, to limit excessive radiation exposure, radiation workers are prohibited from being in containment prior to moving the MDs.

If the MDs become stuck during transit, radiation workers will have to work near the MD drive unit in order to resolve the problem. Under this condition, temporary shielding and portable ARMs will be used as necessary to protect personnel working near the MD drive unit.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/7/2009

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QUESTION NO.: 12.03-12.04-20

Question 8819-Q1

10 CFR 20.1101(b), 1201and 1202 require licensees to control internal and external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA.

APWR FSAR Tier 2 section 12.3.3 "Ventilation" discusses the how the plant HVAC system is designed to meet the requirements of 10 CFR 20, however, section 12.3.3.2 does not describe the design features provided to control airborne particulate and iodine contamination during refueling activities, resulting from evaporation of the Refueling Cavity water volume, or evaporation from exposed reactor components.

Please revise FSAR Tier 2 section 12.3.3.2 to include the design features for minimizing airborne contamination around the Refueling Cavity during refueling operations, or provide the specific alternative approaches used and the associated justification.

ANSWER:

Regulatory Guide 1.206 C.I.12.3.3 "Ventilation" noted that Section 12.3.3 of the FSAR should describe any ventilation system personnel protective features that are not addressed in Chapter 11 or described in Chapter 9.

The feature to maintain low airborne radioactivity in the containment is described in Chapter 9 Subsection 9.4.6. Therefore, these are not described in Chapter 12.

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.