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TOKYO, JAPAN

June 14, 2012

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-12159

Subject: **MHI's Revised Response to US-APWR DCD RAI No. 427-2909 REVISION 1 (SRP 12.02)**

References: 1) "Request for Additional Information No. 427-2909 Revision 1, SRP Section: 12.02 - Radiation Sources, Application Section: 12.2," dated July 30, 2009 (ML092110669).

2) "MHI's Responses to US-APWR DCD RAI 427-2909 Rev. 1, 428-2910 Rev. 1 and 429-3178 Rev. 1," MHI letter UAP-HF-09473, dated September 30, 2009 (ML093340084).

3) "United States – Advanced Pressurized Water Reactor Design Certification Application – Safety Evaluation with Open Items for Chapter 12, 'Radiation Program'", dated March 21, 2011 (ML110770408).

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Revised Response to Request for Additional Information No. 427-2909 Revision 1." In Reference 2, MHI provided the response to the NRC's Request for Additional Information ("RAI") in Reference 1. MHI has revised the response to Question 12.02-19 of Reference 2 in order to address NRC concerns identified as Open Item 12.02-1 in Reference 3. This letter transmits only the revised response to RAI 427-2909 Question 12.02-19; the responses to the other questions transmitted by Reference 2 are not revised by this letter.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Revised Response to Request for Additional Information No. 427-2909 REVISION 1
   (Proprietary version)

3. Revised Response to Request for Additional Information No. 427-2909 REVISION 1
   (Non-proprietary version)

CC: J. A. Ciocco
    J. Tapia

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-12159

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Director, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Revised Response to Request for Additional Information No. 427-2909 REVISION 1" dated June 2012, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design information and analysis of Radiation Protection, developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of Research and Development and detailed design for its software and hardware extending over several years.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with development of the US-APWR Radiation Protection. Providing public access to such information permits competitors to duplicate or mimic the Radiation Protection information without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced US-APWR Radiation Protection development costs associated with the Radiation Protection design.

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

Executed on this 14th day of June, 2012.

Y. Ogata

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

Enclosure 3

UAP-HF-12159
Docket No. 52-021

Revised Response to Request for Additional Information No. 427-2909 REVISION 1

June 2012
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/14/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 427-2909 REVISION 1**

SRP SECTION: **12.02 – Radiation Sources**

APPLICATION SECTION: 12.2

DATE OF RAI ISSUE: **7/30/2009**

QUESTION NO.: 12.02-19

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

Supplemental Question SQ-3 derive from 143-1737

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

In summary, the Applicant Response indicated that:

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

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

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

2. Some of the information provided for Table 12.2-60 "Parameters and Assumptions for Calculating Airborne Radioactivity Concentrations is inaccurate or incomplete:

a. Regarding Table 12.2-60 (Containment) (Sheet 1 of 3):

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

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

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

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

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

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

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

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

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

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

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

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

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

Requested Information

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

References

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

ANSWER:

Answer to 1a

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

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

where:

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

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

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

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

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

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

Answer to 1b

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

Answer to 2a I)

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

This dilution effect is included in the RAM to free volume value provided in DCD Table 12.2-60, but the description is easily misunderstood. Therefore, MHI revised DCD Table 12.2-60.

Answer to 2a II)

The noted typographical error was corrected in DCD Revision 2 as previously indicated in MUAP-09003 Revision 1.

Answer to 2a III)

DCD Chapter 16, Technical Specification Surveillance Requirement (SR) 3.6.3.2 states:

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

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

Answer to 2b I)

The noted typographical correction was corrected in DCD Revision 2 as previously indicated in MUAP-09003 Revision 1.

Answer to 2b II)

See the answer to 2a I) above.

Answer to 2b III)

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

Answer to 2c I)

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

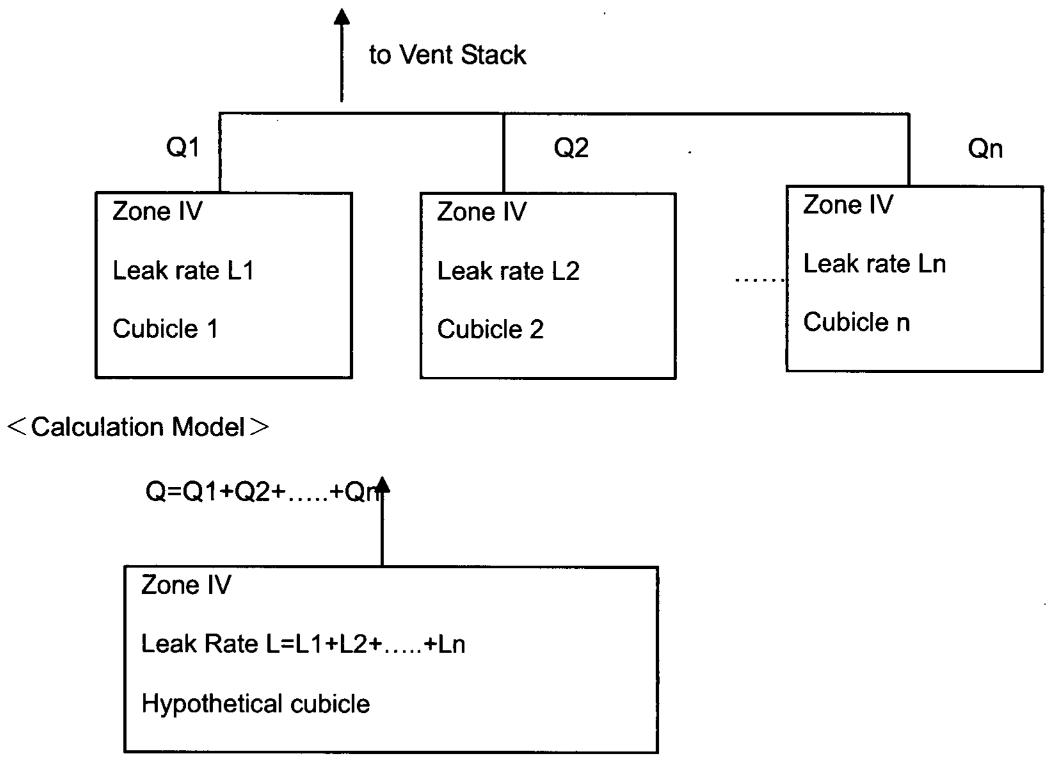

Figure 1 Treatment of the Cubicle in the Airborne Activity Calculation (example)

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

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

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

Answer to 3
Table 12.2-52 is the concentration of the reactor coolant at the time RHR is initiated following cooldown and includes purification and decay. During this four hour period, CVCS

purification is operating at 180 gpm. In calculating the source of the airborne concentration in the reactor cavity water after RHR initiation (after 4 hours), the radioactive concentration in the reactor cavity water, which is the source of airborne radioactivity, is based on 24 hours of radioactive decay and purification by the CVCS (at 400 gpm). No credit is conservatively taken for the decay and purification used in the initial 4-hour period prior to RHR initiation in the calculation. CVCS purification occurs at 400 gpm once RHR has been initiated. Therefore, the values in DCD Table 12.2-72 are different from the values in DCD Table 12.2-52 "RHR System Activity – 4 Hours After Shutdown" because no credit is taken for the decay and purification that occurs during the 4-hour period prior to RHRS initiation. The initial concentrations used in the Table 12.2-72 calculation do not conservatively include the decay and purification at 180 gpm during this 4-hour time period prior to RHRS initiation.

The methods, models, and assumptions used to derive the values presented in DCD Table 12.2-72 are shown in Attachment-1 of the response to this RAI.

Impact on DCD

DCD Table 12.2-60 was revised as previously indicated in MUAP-09003 Revision 1. In addition to those revisions, DCD Table 12.2-60 was also revised as described in the answer to 2aI), 2bII), 2cI) and 2c II) above. DCD Table 12.2-60 (sheet 2 of 3) was revised to include the information described in the answer to 2a III). DCD Table 12.2-61 was also revised as described in the answer to 2cI) above.

DCD Section 9.3.4.1.2.3 will be revised to include the following statement as shown in the attached markup (refer to Attachment-3):
"The 400 gpm letdown flow is purified in parallel with the two mixed bed demineralizer inlet filters, the two mixed bed demineralizers, and the two reactor coolant filters. The let down flow is cooled by the letdown heat exchanger before entering the VCT through the two spray nozzles, and finally returning to the RCS through the normal charging flow path. Two charging pumps are in service for purification during shutdown. The RHRS provides 400 gpm flow rate of all recirculation flow for CVCS low pressure letdown line for shutdown purification."

DCD Section 12.2.2.5 will be revised to include the equation 1 in the response to 1a as well as the equation 3 and the assumption described previous to this equation as shown in the attached markup (refer to Attachment-3).

DCD Table 12.2-72 will be revised to include a note with the information described in the answer to 3 as shown in the attached markup (refer to Attachment-3).

In the markup provided with this RAI, only the revision items that were not incorporated in DCD Revision 3 were included.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.

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

1. Reactor Cavity Water Activity

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

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

2. Primary Coolant Activity after Shutdown

Performing the necessary substitutions and solving Eq. A2-1 and A2-6 results in the following equations for the primary coolant activity of the parent and daughter radionuclide, A1 and B1, respectively.

The values of the input parameters for Eq. 2A-17 through Eq. 2A-23 are provided in Table 2 below.

Table 1 Parameters Used to Calculate Reactor Cavity Water Activity

Table 2 Parameters Used to Calculate Primary Coolant Activity After Shutdown

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

Attachment 2

Detailed Description of CVCS Letdown Flow Rate Following Reactor Shutdown.

1. System description

DCD Table 9.3.4-3 shows each component's design. Normal operating conditions are the bounding case in determining the design of the regenerative heat exchanger. . During purification, flow rate will increase from 180 gpm at normal operating conditions to 400 gpm once RHRS is initiated. However, inlet temperature decreases to less than normal operating conditions. The lower inlet temperature compensates for the augmentation of the flow rate during purification.

CVCS components used for purification of water in the refueling cavity, and letdown flow rate are identified in Subsection 9.3.4.1.2.3 "Purification".

Figure 9.3.4-1 Chemical and Volume Control System flow diagram (sheet 4 of 7) (See Figure A-3), the valve VLV-103 is shown above the VCT. When CVCS letdown flow rate is increased to 400 gpm, the valve is opened. Therefore, letdown flow of 200 gpm will pass through two spray nozzles. Detailed information will be incorporated in the operations procedure concerning piping line-up, etc.

Each spray nozzle is designed as 180 gpm rated flow rate, and gas partition factor is based on the condition of the rated flow rate of 180 gpm under normal operation. If the flow rate increases from 180 gpm to 200 gpm, the drop size becomes smaller than rated flow, and the gas partition factor increases beyond what is realized during normal operation.

There is a 5 psi increase at each spray nozzle during purification as a result of the flow rate increase. This increment is not a concern when viewing the over-all pressure balance.

Although the 400 gpm low-pressure letdown flow passes through the letdown pressure control valve PCV-014-N, there is no concern about the valve choking because the valve is designed to work at low pressure and the higher flow rate 400 gpm conditions. Therefore it's possible to pass water through the valve during purification.

2. Assumption for the airborne radioactivity calculation

Figure A-2 describes the assumptions used in the estimation of RCS concentrations for the US-APWR in comparison to the actual conditions expected.

During normal operation, the concentrations of radionuclides in primary coolant are rather stable due to a continuous CVCS purification flow rate of 180 gpm that compensates for the production of additional radionuclides due to fission and activation processes. After shutdown is initiated, the production of new radionuclides ceases. Between the initiation of shutdown and the start of the RHRS, the purification flow rate is maintained at 180 gpm and concentrations in primary coolant continue to decrease. This reduction in radionuclide concentration is intensified by a larger purification flow rate (400 gpm) when the RHRS is initiated.

In principle, there are 2 phases of reduction of primary coolant concentrations:
- Between the initiation of shutdown and the start of the RHRS: purification at 180 gpm
- After the start of the RHRS: purification at 400 gpm

As shown in Figure A-2, the US-APWR estimation conservatively assumes that the concentration of radionuclides in the RCS remains constant during the period between the beginning of shutdown and the start of the RHRS (i.e., no decay and no CVCS purification is credited prior to RHRS initiation). Both decay and purification (400 gpm) are credited for a 24 hour period between the connection of the RCS to the RHRS and the end of shutdown when operators can begin refueling operations. However, practical considerations of refueling outage risk management will prevent fuel movement until well after 24 hours, resulting in additional purification and decay beyond what is currently conservatively modeled in the US-APWR airborne activity concentration evaluation.

3. The impact for the radiation source for the shielding calculation
The impact of the increased flow rates on demineralizer activity concentration values is negligible.
The use of the mixed-bed demineralizer (which is installed at the uppermost stream of the CVCS) is considered using a flow rate parameter of 180 gpm over 731 days for the activity accumulation calculation. The calculation results show that the activity concentration of Cs-137, which has the longest half-life and is one of the most contributive nuclide for the activity concentration, is 7.66E+04 (Note) μCi/cm^3.

In order to confirm the impact of the increase of flow rate to 400 gpm, MHI evaluated the Cs-137 activity concentration with the condition that the flow rate parameter of 180 gpm is maintained for 731 days as normal operation and 400 gpm for 1 day in shutdown conditions, after initiation of RHR. As a result of the evaluation, the activity concentration of Cs-137 slightly increased by 0.3% to 7.68E+04 μCi/cm^3.

Even when considering that all activity concentrations of all long half-life nuclides in the mixed-bed demineralizer rose by 0.3%, it does not cause a significant rise in the source strength in the demineralizer. Therefore, it does not impact the dose rate at the outside of the concrete shielding.

Note: This value is rounded to two significant figures in Table 12.2-27 of the US-APWR DCD Chapter 12.

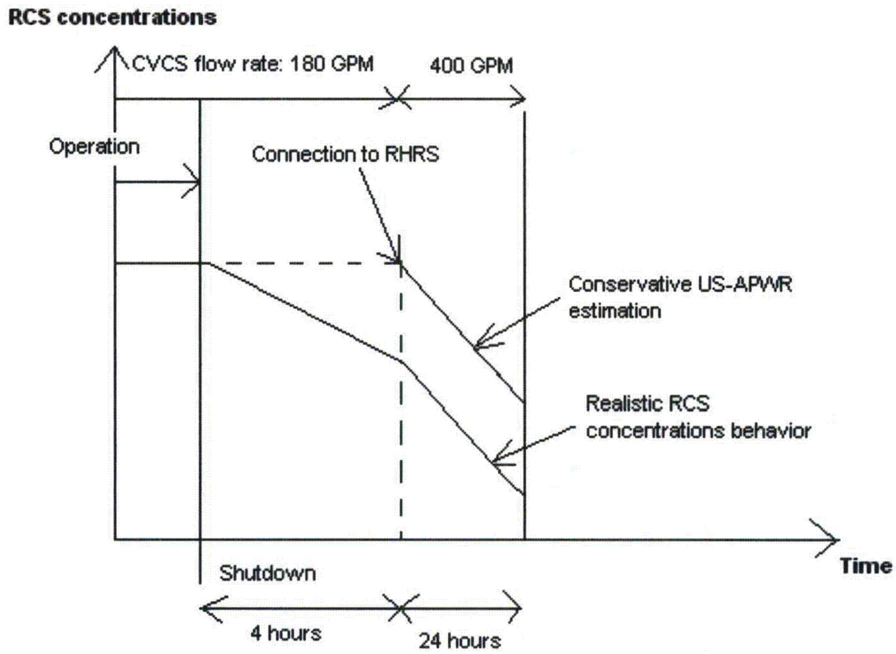

Figure A-2: Comparison between US-APWR estimation conditions and realistic conditions

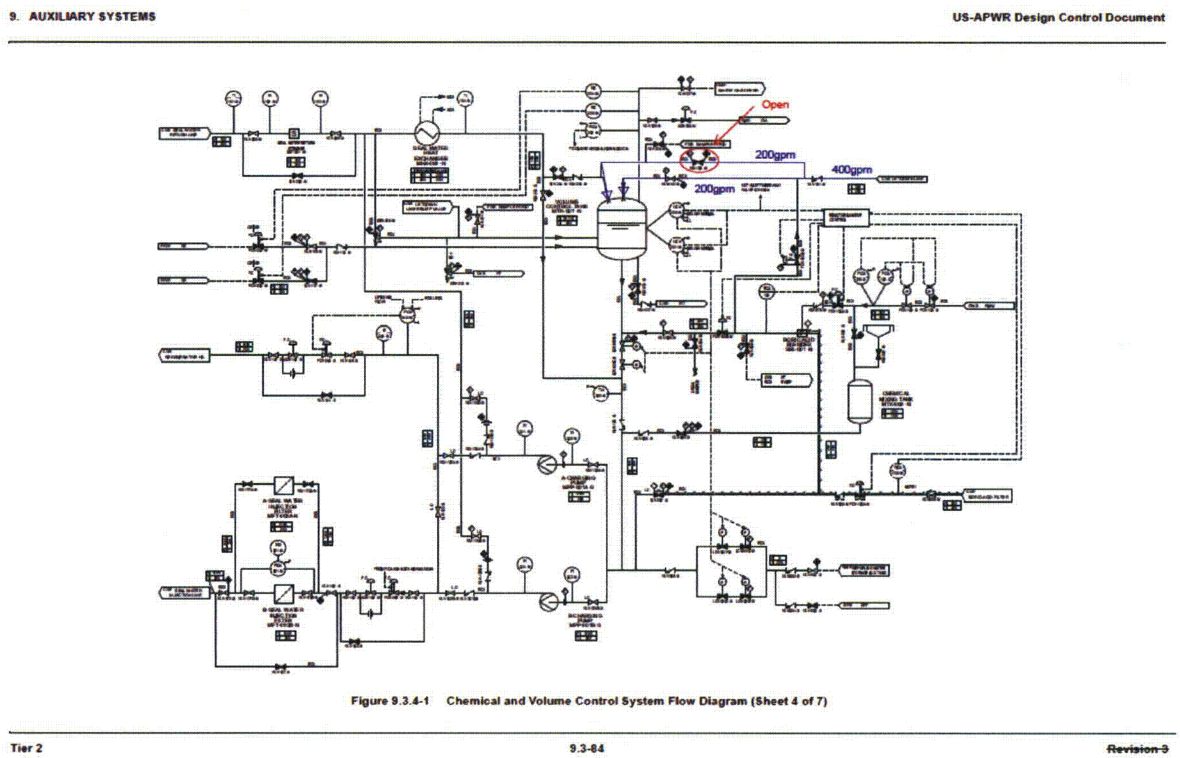

Figure A-3: Marked up of figure 9.3.4-1 for valve opening after connection to RHRS

RCS chemistry changes are accomplished with a feed and bleed operation. The letdown and makeup paths are operated simultaneously and appropriate chemicals are provided at the suction of the charging pumps.

The water chemistry specification for the reactor coolant during normal operation is shown in Table 9.3.4-1.

9.3.4.1.2.3 Purification

The CVCS removes radioactive corrosion products, ionic fission products, and fission gases from the reactor coolant to maintain low RCS activity levels. The CVCS purification capability takes into account occupational radiation exposure (ORE) to support ALARA goals.

The purification rate is based on minimizing ORE and providing access to the equipment for maintenance and inspection activities.

The CVCS has sufficient RCS purification and degasification capability to allow the reactor vessel head to be removed expeditiously during a refueling shutdown. In addition, purification during shutdowns has positive impact on reducing the ORE to workers during the outage. The CVCS is capable of providing purification flow up to 400 gpm when using the RHRS for letdown during shutdown. The 400 gpm letdown flow is purified in parallel with the two mixed bed demineralizer inlet filters, the two mixed bed demineralizers, and the two reactor coolant filters. The let down flow is cooled by the letdown heat exchanger before entering the VCT through the two spray nozzles, and finally returning to the RCS through the normal charging flow path. Two charging pumps are in service for purification during shutdown. The RHRS provides 400 gpm flow rate of all recirculation flow for CVCS low pressure letdown line for shutdown purification. The CVCS supports the plant ALARA goals with its shutdown purification function. DCD_12.02-19

9.3.4.1.2.4 Reactor Coolant Pump Seal Water Injection

The CVCS continuously supplies seal water to the reactor coolant pump seals, as required by the reactor coolant pump design. The seal water flow requirement is specified in Table 9.3.4-2. During a SBO, the reactor coolant pumps seal integrity is maintained until the charging pumps are powered from an alternate power source and seal water injection restarts.

9.3.4.2 System Description

The CVCS consists of charging pumps, regenerative heat exchanger, letdown heat exchanger, excess letdown heat exchanger, demineralizers, filters, pumps, tanks, and associated valves, piping, and instrumentation. The system parameters are given in Table 9.3.4-2. The piping and instrumentation diagram for the CVCS is included in Figure 9.3.4-1. The seismic category and quality group classification for CVCS components are specified in Chapter 3, Section 3.2.

equilibrium airborne concentration of the radioisotope in the regions is calculated using the following equation:

$$C_i(t) = \frac{(LR)_i A_i (PF)_i [1 - \exp(-\lambda_{Ti} t)]}{V \lambda_{Ti}}$$ Eq. 12.2-1

where:

$(LR)_i$ = Leak or evaporation rate of the ith radioisotope in the applicable region (g/s)

A_i = Radioactivity concentration of the ith leaking or evaporating radioisotope (μCi/g)

$(PF)_i$ = Partition factor or the fraction of the leaking radioactivity that is airborne for the ith radioisotope

λ_{Ti} = Total removal rate constant for the ith radioisotope from the applicable region (1/s)

λ_{Ti} = $\lambda_{di} + \lambda_e$, the removal rate constants in 1/s due to radioactive decay for the ith radioisotope and the exhaust from the applicable region, respectively

λ_e = the exhaust removal rate in 1/s defined as Q/V DCD_12.02-19

λ_{di} = the radioactive decay rate in 1/s for the i^{th} radioisotope

t = Time elapsed from the start of the leak and the time at which the concentration is evaluated (s)

V = Free volume of the region in which the leak occurs (cm^3)

Q = Ventilation flow rate (cm^3/s) DCD_12.02-19

$C_i(t)$ = Airborne concentration of the ith radioisotope at time t in the applicable region ($\mu Ci/cm^3$)

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

$$C_i(t) = \frac{(LR)_i A_i (PF)_i}{V \lambda_{Ti}}$$ Eq. 12.2-2

With high exhaust rates, this peak concentration will be reached within a few hours.

As a conservative assumption, radioactive decay of the i^{th} radioisotope is ignored. Using this assumption, Eq. 2 can be simplified as represented below: DCD_12.02-19

$$C_i = \frac{(LR)iAi(PF)i}{V\left(\frac{Q}{V}\right)} = \frac{(LR)iAi(PF)i}{Q}$$

12.2.2.6 Sources Resulting from Design-Basis Accidents

The radiation sources from DBAs include the design basis inventory of radioactive isotopes in the reactor coolant, plus the postulated fission product released from the fuel. DBA parameters and sources are discussed and evaluated in Chapter 15, Subsection 15.6.5.5.

12.2.3 Combined License Information

COL 12.2(1) *The COL Applicant is to list any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography.*

COL 12.2(2) *The COL Applicant is to address the radiation protection aspects associated with additional storage space for radwaste and/or additional radwaste facilities for dry active waste.*

COL 12.2(3) The COL Applicant is to include the conduct of regular surveillance activities and provisions to maintain the dose rate at 2 meters from the surface of both the RWSAT and the PMWTs under 0.25 mrem/h in the Radiation Protection Program.

COL 12.2(4) The COL Applicant is to implement a method of ensuring that the radioactivity concentration in both the RWSAT and the PMWTs remain under the specified concentration level described in the DCD.

12.2.4 References

12.2-1 "Standards for Protection Against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20, U.S. Nuclear Regulatory Commission, Washington, DC, May 1991.

12.2-2 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. RG 8.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

12.2-3 Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable. RG 8.10, Rev. 1-R, U.S. Nuclear Regulatory Commission, Washington, DC, May 1977.

12.2-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. RG 1.183, Rev. 0, U.S. Nuclear Regulatory Commission, Washington, DC, July 2000.

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

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

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Table 12.2-72 Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium) (Sheet 2 of 2)

Nuclide	Specific Activity (μCi/g)	Nuclide	Specific Activity (μCi/g)
Tc-101	-	Zn-65	2.0E-08
Ru-103	8.3E-09		
Rh-103m	2.3E-05		

Note: The initial concentrations used in the calculation that determined the values in this table conservatively exclude the decay and purification at 180 gpm during the 4-hour time period prior to RHRS initiation. Decay and purification are credited for 24 hours after connection of RCS to RHRS. DCD_12.02-19