MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

March 30, 2011

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-11085

Subject: **MHI's Responses to US-APWR DCD RAI 711-5533 Rev.2 (SRP 11.02)**

Reference: 1) "Request for Additional Information No.711-5533 Revision 2, SRP Section: 11.02 –Liquid Waste Management System, Application Section: 11.2" dated March 7,2011.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") documents as listed in Enclosures.

Enclosed are the responses to RAIs contained within Reference 1.

As indicated in the enclosed materials, the attachment data of this document (Enclosure 2) contains information that MHI considers proprietary,

Related in this RAI responses, MHI revised the technical report entitled "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS(proprietary ver. and non-proprietary ver.), MUAP-10019 Revision 1". The report and attachment data shown above are being submitted electronically in compact disc ("CD").

The enclosed report contains information that MHI considers proprietary, and therefore the report should be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. Accordingly, the report is being submitted in two versions, in separate compact discs. One version (in CD 2) contains the complete proprietary version of the report. A non-proprietary version of the report is enclosed in CD 3. In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]". In accordance with the NRC submittal procedures, this letter includes an Affidavit that identifies the reasons why the proprietary version of the report should be withheld from disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

D081
NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Responses to Request for Additional Information No.711-5533 Rev.2 (non-proprietary)

3. CD1:"Attachment of Responses to RAI's item 11.02-34 of NRC Requests"

   CD2: Technical Report "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS, MUAP-10019P Revision 1"
   – Version containing Proprietary information

4. CD3: Technical Report "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS, MUAP-10019NP Revision 1"
   – Version containing Non-Proprietary information

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

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-11085

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, Ltd ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from 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 "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS" which is the technical report dated March 2011, and "Attachment of Responses to RAI's item 11.02-34 of NRC Requests", have determined that the portions of the document contain proprietary information that should be withheld from public disclosure. Those pages of the technical report 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 methodology for evaluation to comply with Regulatory Guide 1.109 and 1.112, developed by MHI. This methodology was developed to significant cost to MHI, and with knowledge and know-how about using the PWR-GALE and RATAF code.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's Application for certification of its US-APWR Standard Plant Design.

6. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design of new fuel systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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

Executed on this 30th day of March, 2011.

Y. Ogata

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

Enclosure 2

UAP-HF-11085

Responses to Request for Additional Information No.711-5533
Revision 2

March 2011
(Non Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

03/30/2011

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

RAI NO.:	**NO. 711-5533 REVISION 2**
SRP SECTION:	**11.02 – Liquid Waste Management System**
APPLICATION SECTION:	**11.2**
DATE OF RAI ISSUE:	**03/07/2011**

QUESTION NO. : 11.02-34

Staff review of MHI Technical Report (TR) MUAP-10019[Proprietary]P, TR MUAP- 10019[Non-Proprietary ML102850687]NP (Revision 0), "Calculation Methodology for Radiological Consequences in Normal Operation and Tank Failure Analysis," September 2010, provided as a result of the staff's audit (ML102810271) on the MHI PWR-GALE code and in response to item 2 of RAI 629-4973, Question 11.03-18 and RAI 624-4972, Questions 11.02-33 found that the requested calculation packages were not included. Although TR MUAP-10019P (R0) describes the calculation methodology, gaseous and liquid effluent and dose results, basis for input design parameter values, gas and liquid tank failure analysis and summarizes the quality assurance and validation procedures for MHI's proprietary version of the NRC PWR-GALE code used to calculate expected annual liquid and gaseous effluent releases during normal operation including AOOs for a plant referencing the US-APWR design, it does not include the calculation packages which show demonstration of compliance with 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2; 10 CFR 50, Appendix A; and 40 CFR Part 190. Please address the following items and provide a markup on the proposed DCD changes (where applicable).

1. Provide a copy of the following calculation packages: PWR-GALE code calculations on maximum liquid effluent releases and comparisons to the ECLs in 10 CFR Part 20, Appendix B, Table 2, and maximum gaseous effluent releases and comparisons to the ECLs in 10 CFR Part 20, Appendix B, Table 1; waste gas surge tank leak analysis; and charcoal bed leak analysis, as requested in item 2 of RAI 629-4973, Question 11.03-18 and RAI 624-4972, Questions 11.02- 33, and as discussed in the staff's audit report.

2. Section 4, "Radioactive Effluent Releases due to Liquid Containing Failures - Tank Activities," Appendix A, "Basis for PWR-GALE code inputs," and Tables 16, "Input parameters for the RATAF code," and 17,"Source term for Liquid Containing Tank Failures" of TR MUAP-10019P/NP (R0) describes the basis for input design parameter values and calculation results for the new approach on the liquid tank failure analysis. Under 10 CFR 2.390, in response to RAI 403- 3027, Question 11.02-20, item 4, by letter dated July 16, 2009, a copy of the RATAF code input/output files were provided as requested by the staff. However, this information has since been updated. Provide a copy of the RATAF code input/output files used in the new approach on the liquid tank failure analysis described in TR MUAP-10019P/NP (R0).

3. Reference TR MUAP-10019P/NP (R0) in the appropriate DCD sections (e.g., DCD Tier 2, Sections 11.2 and 11.3, etc.).

ANSWER:

Item 1

The calculation packages are provided in the attached documents:

- N0-EB30008 (gaseous effluent and dose)
- N0-EB30009 (liquid effluent and dose)
- N0-EB30010 (gaseous waste management system leak or failure).

These three documents are the same technical content as in the documents UAP-HKH-0016, UAP-HKH-0017 and UAP-HKH-0018 that were requested after the NRC audit for the PWR-GALE code analyses. In the document N0-EB30008, the calculated releases and concentrations of the design basis calculation of I-131 releases were corrected. This correction resolved an error in the calculation of design basis from the results of the realistic estimation using design basis RCS concentrations. Because the releases are proportional to the RCS source term, the design basis releases were calculated by multiplying the ratio of the design basis RCS concentrations over the realistic RCS concentrations, to the realistic releases (output of PWR-GALE code). In the case of I-131, the error was consisting in using the I-130 RCS concentrations when adjusting I131 design basis releases. This has been corrected in the present documents.

In this revision, other nuclides activities were corrected. In the calculation of total releases, the summation of the different waste lines is increased to include the releases due to Anticipated Operating Occurrences (AOO) (0.16 Ci). This total AOO release is distributed to every nuclide depending on the importance on the total releases. Changing the calculated releases of I-131, the distribution of AOO releases is also modified. This impacts some total releases of other nuclides.

These corrections will be incorporated in the US-APWR DCD Revision 3. The design basis liquid effluent releases and concentrations for R-COLA will be also revised.

There is no impact on the S-COLA calculations because these calculations are independent due to the boric acid evaporator departure and the independent calculations were performed with consideration not repeating the same error. Both realistic and design basis releases were revised. The occurrence of using I-130 concentrations instead of I-131 or any similar error was carefully checked and precluded.

The technical report MUAP-10019-P (Proprietary) / MUAP-10019-NP (Non-proprietary, ML102850687) Revision 0, "Calculation Methodology for Radiological Consequences in Normal Operation and Tank Failure Analysis," that was provided after the NRC audit on PWR-GALE code is also revised to take into account these corrections. It is attached to this response.

Item 2

The RATAF code input/outputs files are provided in the attached CD.

Item 3

The reference will be included in the next DCD revision. The mark-up is attached to this response letter.

Impact on DCD

DCD Table 11.2-11, Table 11.2-13, and the reference for the technical report will be revised. (See attached markup)

Impact on COLA

R-COLA Table 11.2-11R and Table 11.2-13R will be revised.
No-impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Table 11.2-11 Liquid Releases with Maximum Defined Fuel Defects (Ci/yr)

(Sheets 1 of 2)

Isotope	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Detergent Waste	TOTAL Releases (1)
Corrosion and Activation Products						
Na-24	0.00000	0.00029	0.00002	0.00031	0.00000	3.20E-04
P-32	0.00000	0.00000	0.00000	0.00000	0.00018	1.80E-04
Cr-51	0.00000	0.00008	0.00000	0.00008	0.00470	4.78E-03
Mn-54	0.00000	0.00004	0.00000	0.00004	0.00380	3.84E-03
Fe-55	0.00000	0.00003	0.00000	0.00003	0.00720	7.23E-03
Fe-59	0.00000	0.00001	0.00000	0.00001	0.00220	2.21E-03
Co-58	0.00000	0.00012	0.00000	0.00012	0.00790	8.02E-03
Co-60	0.00000	0.00001	0.00000	0.00001	0.01400	1.40E-02
Ni-63	0.00000	0.00000	0.00000	0.00000	0.00170	1.70E-03
Zn-65	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
W-187	0.00000	0.00002	0.00000	0.00002	0.00000	2.06E-05
Np-239	0.00000	0.00003	0.00000	0.00003	0.00000	~~3.10E-05~~ 3.09E-05
Fission Products						
Rb-88	0.00000	0.03849	0.00000	0.03849	0.00000	3.97E-02
Sr-89	0.00000	0.00000	0.00000	0.00000	0.00009	9.00E-05
Sr-90	0.00000	0.00000	0.00000	0.00000	0.00001	1.00E-05
Sr-91	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Y-91m	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Y-91	0.00000	0.00000	0.00000	0.00000	0.00008	8.00E-05
Y-93	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Zr-95	0.00000	0.00001	0.00000	0.00001	0.00110	1.11E-03
Nb-95	0.00000	0.00002	0.00000	0.00002	0.00190	1.92E-03
Mo-99	0.00000	0.01333	0.00000	0.01333	0.00006	1.38E-02
Tc-99m	0.00000	0.00527	0.00000	0.00527	0.00000	5.44E-03
Ru-103	0.00000	0.00001	0.00000	0.00001	0.00029	3.00E-04
Rh-103m	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
Ru-106	0.00000	0.00001	0.00000	0.00001	0.00890	8.91E-03
Rh-106	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
Ag-110m	0.00000	0.00000	0.00000	0.00000	0.00120	1.20E-03
Ag-110	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Sb-124	0.00000	0.00000	0.00000	0.00000	0.00043	4.30E-04
Te-129m	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Te-129	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Te-131m	0.00000	0.00033	0.00000	0.00033	0.00000	3.40E-04
Te-131	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
I-131	~~0.00113~~ 0.02891	~~0.00056~~ 0.01445	0.00000	~~0.00169~~ 0.04336	0.00160	~~3.34E-03~~ 4.63E-02
Te-132	0.00000	0.00526	0.00000	0.00526	0.00000	5.43E-03
I-132	0.00000	0.00015	0.00015	0.00030	0.00000	~~3.10E-04~~ 3.09E-04
I-133	0.00163	0.00327	0.00491	0.00981	0.00000	1.01E-02
I-134	0.00000	0.00005	0.00000	0.00005	0.00000	5.16E-05
Cs-134	0.73457	1.83643	0.00000	2.57100	0.01100	2.66E+00
I-135	0.00000	0.00083	0.00125	0.00208	0.00000	2.15E-03
Cs-136	0.12019	0.44873	0.00000	0.56892	0.00037	5.87E-01
Cs-137	0.43698	1.16528	0.00000	1.60226	0.01600	1.67E+00

Notes:

Table 11.2-11 Liquid Releases with Maximum Defined Fuel Defects (Ci/yr)

(Sheets 2 of 2)

Isotope	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Detergent Waste	TOTAL Releases (1)
Ba-137m	0.20917	0.00000	0.00000	0.20917	0.00000	2.16E-01
Ba-140	0.00000	0.00010	0.00000	0.00010	0.00091	1.01E-03
La-140	0.00000	0.00002	0.00000	0.00002	0.00000	2.06E-05
Ce-141	0.00000	0.00000	0.00000	0.00000	0.00023	2.30E-04
Ce-143	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Pr-143	0.00000	0.00000	0.00000	0.00000	0.00000	0.00E+00
Ce-144	0.00000	0.00001	0.00000	0.00001	0.00390	3.91E-03
Pr-144	0.00000	0.00001	0.00000	0.00001	0.00000	1.03E-05
TOTAL (except H-3)	~~1.50367~~ 1.53145	~~3.51883~~ 3.53272	**0.00633**	~~5.02883~~ 5.07050	**0.08975**	~~5.28E+00~~ 5.32E+00
H-3 release						1.60E+03

Notes:

1. The release totals include an adjustment of 0.16 Ci/yr added by the PWR-GALE Code to account for AOOs.
2. An entry of 0.00000 indicates that the value is less than 1.0E-5 Ci/yr.

Table 11.2-13 Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 (Maximum Releases) (Sheets 1 of 2)

Isotope[1]	Discharge Concentration (μ Ci/ml) [2]	Effluent Concentration Limit (μ Ci/ml) [3]	Fraction of Concentration Limit
Na-24	1.56E-11	5.00E-05	~~3.12E-07~~ 3.11E-07
P-32	8.77E-12	9.00E-06	9.74E-07
Cr-51	2.33E-10	5.00E-04	4.66E-07
Mn-54	1.87E-10	3.00E-05	6.24E-06
Fe-55	3.52E-10	1.00E-04	3.52E-06
Fe-59	1.08E-10	1.00E-05	1.08E-05
Co-58	3.91E-10	2.00E-05	1.95E-05
Co-60	6.82E-10	3.00E-06	2.27E-04
Ni-63	8.28E-11	1.00E-04	8.28E-07
Zn-65	~~5.03E-13~~ 5.02E-13	5.00E-06	~~1.01E-07~~ 1.00E-07
W-187	~~1.01E-12~~ 1.00E-12	3.00E-05	3.35E-08
Np-239	1.51E-12	2.00E-05	7.54E-08
Rb-88	1.93E-09	4.00E-04	4.84E-06
Sr-89	4.38E-12	8.00E-06	5.48E-07
Sr-90	4.87E-13	5.00E-07	9.74E-07
Sr-91	0.00E+00	2.00E-05	0.00E+00
Y-91m	0.00E+00	2.00E-03	0.00E+00
Y-91	3.90E-12	8.00E-06	4.87E-07
Y-93	0.00E+00	2.00E-05	0.00E+00
Zr-95	5.41E-11	2.00E-05	2.70E-06
Nb-95	9.35E-11	3.00E-05	3.12E-06
Mo-99	6.73E-10	2.00E-05	3.36E-05
Tc-99m	2.65E-10	1.00E-03	2.65E-07
Ru-103	1.46E-11	3.00E-05	4.88E-07
Rh-103m	~~5.03E-13~~ 5.02E-13	6.00E-03	~~8.38E-11~~ 8.37E-11
Ru-106	4.34E-10	3.00E-06	1.45E-04
Ag-110m	5.84E-11	6.00E-06	9.74E-06
Sb-124	2.09E-11	7.00E-06	2.99E-06
Te-129m	0.00E+00	7.00E-06	0.00E+00
Te-129	0.00E+00	4.00E-04	0.00E+00
Te-131m	1.66E-11	8.00E-06	2.07E-06
Te-131	0.00E+00	8.00E-05	0.00E+00
I-131	~~1.63E-10~~ 2.26E-09	1.00E-06	~~1.63E-04~~ 2.26E-03
Te-132	2.64E-10	9.00E-06	2.94E-05
I-132	1.51E-11	1.00E-04	1.51E-07
I-133	4.93E-10	7.00E-06	7.04E-05
I-134	2.51E-12	4.00E-04	6.28E-09
Cs-134	1.30E-07	9.00E-07	1.44E-01

Notes:
1. Rh-106,Ag-110,Ba-137m are not included in Table 2 of 10 CFR 20 Appendix B. Therefore, these nuclides are excluded from the calculation of the discharge concentration.
2. Annual average discharge concentration based on release of average daily discharge for 292 days per year with 12,900 gpm dilution flow.
3. 10 CFR 20 Appendix B, Table 2

Table 11.2-13 Comparison of Annual Average Liquid Release Concentrations with 10 CFR 20 (Maximum Releases) (Sheets 2 of 2)

Isotope (1)	Discharge Concentration (μCi/ml) (2)	Effluent Concentration Limit (μCi/ml) (3)	Fraction of Concentration Limit
I-135	~~1.05E-10~~1.04E-10	3.00E-05	3.48E-06
Cs-136	2.86E-08	6.00E-06	4.77E-03
Cs-137	8.13E-08	1.00E-06	8.13E-02
Ba-140	4.93E-11	8.00E-06	6.17E-06
La-140	~~1.01E-12~~ 1.00E-12	9.00E-06	1.12E-07
Ce-141	1.12E-11	3.00E-05	3.73E-07
Ce-143	0.00E+00	2.00E-05	0.00E+00
Pr-143	0.00E+00	2.00E-05	0.00E+00
Ce-144	1.90E-10	3.00E-06	6.35E-05
Pr-144	~~5.03E-13~~ 5.02E-13	6.00E-04	~~8.38E-10~~ 8.37E-10
H-3	7.79E-05	1.00E-03	7.79E-02
TOTAL			~~3.09E-01~~ 3.11E-01

Notes:

1. Rh-106,Ag-110,Ba-137m are not included in Table 2 of 10 CFR 20 Appendix B. Therefore, these nuclides are excluded from the calculation of the discharge concentration.
2. Annual average discharge concentration based on release of average daily discharge for 292 days per year with 12,900 gpm dilution flow.
3. 10 CFR 20 Appendix B, Table 2

particular equipment. The decontamination factors are taken from NUREG-0017 Rev.1(Ref.11.2-13) and are presented in Table 11.2-7. The calculations are made based on the assumption as follows:

- The liquid effluent from the primary system is processed by the LWMS and is discharged without reuse.
- The steam generator blowdown is treated and returned to the condenser.

The release physical location and configuration of the treated effluent is site specific. Detailed design information such as release point, effluent temperature and flow rate, and size and shape of flow orifices, is to be presented in the site specific detail design. The COL applicant is responsible for ensuring that the site-specific information of the LWMS, e.g., radioactive release points, effluent temperature, shape of flow orifices, etc., is to be provided in the COLA(COL11.2(2)).

The annual average release of radionuclides is estimated by the PWR-GALE Code (Ref.11.2-13) with the reactor coolant activities that ~~is~~ are described in Section 11.1. The version of the code is a proprietary modified versi~~on of the NRG~~ PWR-GALE code reflecting the design specificities of US-APWR design(Ref.11.2-27). The parameters used by the PWR-GALE Code are provided in Table 11.2-9, and the calculated effluents are provided in Table 11.2-10. The calculated effluents for the maximum releases are provided in Table 11.2-11.

The calculated effluent concentrations using annual release rates are then compared against the concentration limits of 10 CFR 20 Appendix B (Ref. 11.2-8); see Table 11.2-12 and Table 11.2-13.

The calculation uses 12,900 gpm [[cooling tower blowdown]] as dilution water (See Chapter 10, Subsection 10.4.5). The ratios to the concentration limits of 10 CFR 20 Appendix B (Ref 11.2-8) are 8.10E-02 (with expected releases) and 3.09E-01 (with maximum defined fuel defects), and these values are less than the allowable value of 1.0.

The individual doses are evaluated with LADTAP II Code (Ref. 11.2-14). The parameters used in the LADTAP II Code are listed in Table 11.2-14, and calculated doses are listed in Table 11.2-15. Based on these parameters, the dose to total body is 1.98 mrem/yr (Child) and the dose to organ is 2.54 mrem/yr (Child's liver). These values are less than the criteria of 3 mrem/y and 10 mrem/yr, respectively, as specified in 10 CFR 50 Appendix I (Ref. 11.2-2).

The COL applicant is to calculate doses to members of the public following the guidance of RG 1.109 (Ref 11.2-15) and RG 1.113 using site-specific parameters, and compares the doses due to the liquid effluents with the numerical design objectives of Appendix I to 10 CFR 50 (Ref 11.2-10) and compliance with requirements of 10 CFR 20.1302, 40 CFR 190.

11.2.3.2 Radioactive Effluent Releases due to Liquid Containing Tank Failures

In case of a failure of a tank containing radioactive liquid, the radioactive concentrations in the potable water supply located in the unrestricted area are evaluated with the procedure specified in NUREG-0133 Appendix A (Ref. 11.2-16), and the results are less than the limits of 10 CFR 20 Appendix B (Ref 11.2-8).

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

The source term is calculated in accordance with Branch Technical Position (BTP) 11-6 (Ref 11.2-17). ~~Table 11.2-17 shows summarizes the evaluation results of radioactivity concentration at the location of the potable water supply. Branch Technical Position (~~BTP ~~)~~11-6 ~~(Ref 11.2-17)~~ Subsection B.2, endorses Appendix A of NUREG-0133, which describes the RATAF code for PWR plants. Accordingly, the RATAF code is utilized. ~~in this evaluation.~~

Table 11.2-16 shows input parameters for the RATAF code calculation(Ref.11.2-27).
Table11.2-17 shows the source term calculated by the RATAF code. The concentrations of corrosion and activation products are the RATAF code output. And the concentrations of fission products are calculated by multiplying the RATAF code output by 0.12/1.0 to adjust the fuel defect level from 1% built-in the RATAF code to 0.12% recommended in BTP 11-6.

~~The liquid radioactivity concentration in the tank is calculated by RATAF assuming a primary coolant concentration based on 1% fuel failure. However, for the determination of the critical receptor concentration BTP 11-6 allows the use of a source term based on the expected failed fuel fraction, which is set to 0.12% of the core thermal power. The evaluation result obtained from the case of the failure of the boric acid tank, which has the largest value of 2.2E-01, indicates that the ratio of concentration is still less than the allowable value of 1.0, in accordance with 10CFR 20 Appendix B (Ref 11.2-8). Satisfying the concentration limits of 10 CFR 20 Appendix B (Ref 11.2-8) results in a~~

11.2-25 Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment, U.S. Nuclear Regulatory Commission, IE Bulletin No. 80-10, May 6, 1980.

11.2-26 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

11.2-27 Calculation Methodology for Radiological Consequences in Normal Operation and Tank Failure Analysis, MUAP-10019-P Rev.1 (Proprietary) and MUAP-10019-NP Rev.1 (Non-proprietary), March , 2011

- Only proven and qualified equipment from the nuclear industry is used.
- Steel piping with butt welded construction is used to minimize crud traps. Only qualified welders are employed.
- Cubicles containing radioactive liquid are steel-lined or lined with an epoxy coating to minimize the potential for contamination to the groundwater system and to ease maintenance and decontamination.
- Drains and overflows are routed directly to sumps to minimize the spread of radioactive fluid.
- Non-radioactive auxiliary subsystems are isolated from the radioactive process streams.

In order to reduce leakage and releases of radioactive material, the following design features are included:

- Equipment, piping, and instruments are subject to stricter leak rate testing and inspections
- Sumps are equipped with level switches that activate alarms for prompt operator action to minimize the spread of contamination

11.3.3 Radioactive Effluent Releases

11.3.3.1 Radioactive Effluent Releases and Dose Calculation in Normal Operation

The GWMS treats and releases radioactive gaseous waste generated from normal operation, including AOOs. Gaseous release data (isotope and activity) are presented in Tables 11.3-5 through 11.3-7. There are no liquid or solid waste releases from the GWMS.

The GWMS is designed to treat potentially radioactive gas to meet the concentration and dose limits of 10 CFR 20 (Ref. 11.3-4), the dose limits of 10 CFR 50, Appendix I (Ref. 11.3-3), and GDC 64. The main sources of plant radioactive gaseous inputs to the GWMS are the waste gases from the VCT, CVDT, boric acid evaporator, and HTs. Their flow rates are presented in Figure 11.3-1 (Sheet 3 of 3).

The treated gaseous waste is further diluted by HVAC ventilation flow before the gases are released from the vent stack. The vent stack runs alongside the containment with the release point above the top of the containment. The design information of the vent stack and release point is described in Section 11.3.2.

The release rates and isotopic compositions are calculated using the PWR-GALE Code, NUREG-0017 (Ref. 11.3-1). The version of the code is a proprietary modified version of the NRC PWR-GALE code reflecting the design specificities of US-APWR design (Ref. 11.3-28). Other parameters for the PWR-GALE Code calculation are listed in

11.3-19 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Regulatory Guide 1.109, Rev. 1, October 1977.

11.3-20 Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors. [This NUREG includes Generic Letter 89-01.], NUREG-1301.

11.3-21 U.S. Nuclear Regulatory Commission, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants. NUREG-0133, Washington, DC.

11.3-22 Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors. Regulatory Guide 1.111, Rev. 1,July 1977.

11.3-23 Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I. Regulatory Guide 1.113, Rev. 1, April 1977.

11.3-24 Compliance with dose limits for individual members of the public. NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 20.1302.

11.3-25 U.S. Environmental Protection Agency, "Environmental Radiation Protection Standards for Nuclear Power Operations," Protection of Environment. Title 40, Code of Federal Regulations, Part 190, Washington, DC.

11.3-26 Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors. Regulatory Guide 1.110, March 1976.

11.3-27 Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment, U.S. Nuclear Regulatory Commission, IE Bulletin No. 80-10, May 6, 1980.

11.3-28 Calculation Methodology for Radiological Consequences in Normal Operation and Tank Failure Analysis, MUAP-10019-P Rev.1 (Proprietary) and MUAP-10019-NP Rev.1 (Non-proprietary), March , 2011.

Enclosure 3

UAP-HF-11085

CD1:"Attachment of Responses to RAI's item 11.02-34 of NRC Requests"

CD2: Technical Report "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS, MUAP-10019P Revision 1"
- Version containing Proprietary information

March 2011
(Proprietary)

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ATTACHMENT 1

FILE CONTAINED IN CD1 and CD2

CD1: "Attachment of Responses to RAI's item 11.02-34 of NRC Requests"

Contents of CD

Calculation documents

File Name	Size	Sensitivity Level
N0EB30008.pdf	3428 kB	Proprietary
N0EB30009.pdf	3997 kB	Proprietary
N0EB30010.pdf	1201 kB	Proprietary

RATAF input/output files(each file is txt format)

File Name	Size	Sensitivity Level
RATAF_INP_HT_1Y.DAT	2 kB	Proprietary
RATAF_INP_HT_1Y.DAT.outlist	113 kB	Proprietary
RATAF_INP_WHT_1Y.DAT	2 kB	Proprietary
RATAF_INP_WHT_1Y.DAT.outlist	113 kB	Proprietary
RATAF_INP_BAT_1Y.DAT	2 kB	Proprietary
RATAF_INP_BAT_1Y.DAT.outlist	113 kB	Proprietary

CD 2: Technical Report "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS, MUAP-10019P Revision 1"
- Version containing Proprietary information

Contents of CD

File Name	Size	Sensitivity Level
MUAP-10019P_R1.pdf	674 KB	Proprietary

Enclosure 4

UAP-HF-11085

CD3: Technical Report "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS, MUAP-10019NP Revision 1"
- Version containing Non-Proprietary information

March 2011
(Non-Proprietary)

ATTACHMENT 2

FILE CONTAINED IN CD3

CD 3: Technical Report "CALCULATION METHODOLOGY for RADIOLOGICAL CALCULATION METHODOLOGY for RADIOLOGICAL and TANK FAILURE ANALYSIS, MUAP-10019NP Revision 1"
- Version containing Non-Proprietary information

Contents of CD

File Name Level	Size	Sensitivity
MUAP-10019NP_R1.pdf	330 KB	Non-Proprietary