Energy Harbor Nuclear Corp.
Beaver Valley Power Station
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April 14, 2020
L-20-097

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
Docket No. 50-334, License No. DPR-66
Docket No. 50-412, License No. NPF-73
Response to Request for Additional Information and Supplemental Information Regarding License Amendment Request to Modify Technical Specifications 3.4.16, "RCS Specific Activity," 3.7.13 "Secondary Specific Activity," 5.5.7, "Ventilation Filter Testing Program (VFTP)," and 5.5.14, "Control Room Envelope Habitability Program" (EPID L-2019-LLA-0223)

By letter dated October 20, 2019 (ADAMS Accession No. ML19293A367), Energy Harbor Nuclear Corp. submitted a license amendment request (LAR) to revise the Beaver Valley Power Station Unit Nos. 1 and 2 Technical Specifications (TS). The requested changes revise TS 3.4.16, "RCS Specific Activity," 3.7.13," Secondary Specific Activity," 5.5.7, "Ventilation Filter Testing Program (VFTP)," and 5.5.14, "Control Room Envelope Habitability Program." The proposed changes to TSs 3.4.16 and 3.7.13 would reduce the allowed reactor coolant system and secondary coolant specific activities for Unit 2 and make administrative changes to the Unit 1 TS. The proposed changes to TS 5.5.7 for the control room emergency ventilation system (CREVS) change the acceptance criteria for the CREVS penetration and system bypass requirement and CREVS charcoal adsorber removal efficiency. The proposed change to the Control Room Envelope Habitability Program in TS 5.5.14 would add a note allowing a one-time extension of three years to the unfiltered air inleakage test frequency.

On March 12, 2020, the Nuclear Regulatory Commission (NRC) staff requested additional information to complete its review. The Energy Harbor Nuclear Corp. response is included in Attachment 1 to this letter.

In response to questions three and four in the request for additional information, a proposed license amendment request TS markup has been updated. Attachment 2 to this letter hereby replaces the previously-submitted markup to TS 5.5.14 provided in the October 20, 2019 submittal.

In response to question five in the request for additional information, the inputs to the radiological dose consequence analyses are provided in Attachment 3 to this letter. The enclosure to this letter contains information supporting the response to question six in the request for additional information.

The information provided by this submittal does not invalidate the significant hazards consideration analysis provided in the October 20, 2019 submittal.

There are no regulatory commitments contained in this submittal. If there are any questions, or if additional information is required, please contact Mr. Phil H. Lashley, Acting Manager, Nuclear Licensing and Regulatory Affairs, at (330) 315-6808.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 14, 2020.

Sincerely,

Rod L. Penfield

Attachment:
1) Response to RAI
2) TS 5.5.14 Markup
3) Input to Radiological Dose Consequence Analyses

Enclosure: WECTEC Global Project Services, Inc. Technical Report

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

The NRC staff's request for additional information is provided in bold text followed by the Energy Harbor Nuclear Corp. response.

1) **The licensee stated the following in the second paragraph of Section 3.4:**

   "FENOC has high confidence that the CRE [control room envelope] has not continued to degrade based upon improving test results between 2015 and 2017 due to plant modifications and system rebalancing efforts for the normal and recirculation (isolation) modes."

   However the issue that triggered the need for a test after 3 years is the failure in 2017. It does not appear appropriate to justify the proposed test extension using test results prior to the failure in 2017. Please provide an appropriate justification for the proposed test extension or revise the request to not include a proposed test extension.

   Response:

   From the table in Section 3.4 of the license amendment request, in 2008, the CRE tracer gas test results for the Unit 1 and Unit 2 pressurization (emergency) modes met the acceptance criteria. However, for the next test in 2015, the Unit 1 and Unit 2 pressurization (emergency) modes did not meet the acceptance criteria. Therefore, between 2015 and 2017, plant modifications and system rebalancing efforts were implemented. Subsequent tracer gas tests were performed in 2016 and 2017 with essentially unchanged results for the Unit 1 and Unit 2 pressurization (emergency) modes. There is high confidence that the unfiltered air inleakage in the Unit 1 and Unit 2 pressurization modes was a step change between 2008 and 2015, and has not continued to degrade as indicated by the test results in 2015, 2016, and 2017.

2) **The licensee also made the following statements in Section 3.4 of the LAR:**

   > **For the Unit 1 and Unit 2 pressurization (emergency) modes, the step change that occurred between 2008 and 2015 has remained essentially unchanged over the most recent two-year test period; therefore, it is not anticipated to degrade any further. Additionally, the proposed acceptance criteria in this license amendment request are expected to provide sufficient margin for potential CRE boundary degradation in the future.**

   While the first sentence appears to support the justification for the need for the proposed changes, the second sentence appears to refute the statement that further boundary degradation is not expected and therefore refute the

justification for the need for the proposed changes. Please clarify whether or not the licensee expects the boundary to continue to degrade and why a test extension would be appropriate.

Response:

Energy Harbor Nuclear Corp. does not expect the CRE pressurization (emergency) mode boundary to continue to degrade. The CRE boundary will continue to be monitored for potential degradation through normal periodic assessments, maintained in accordance with site preventative maintenance programs, and verified to be acceptable through testing including the following: flow and filter efficiency tests for Unit 1 and Unit 2 pressurization fan subsystems; differential pressure readings between CRE areas and adjacent non-CRE areas in all modes of operation; and purge fan flow verifications.

A test extension is appropriate for two reasons:

1) The CRE pressurization (emergency) mode unfiltered air inleakage has remained consistent over the most recent three years of testing (2015 to 2017).

2) In the unlikely event of CRE boundary degradation, extending the test interval is not expected to significantly reduce a margin of safety because the proposed amendment would provide unfiltered air inleakage test margin compared to the most recent test results.

3) In the third paragraph of Section 3.4 of the LAR the licensee stated:

> **Therefore, with the last test performed in October 2017, the next CRE unfiltered air inleakage would be performed no later than April 2025 with the current provisions of TS 5.5.14.f included. FENOC intends to perform the three-year self-assessment that would be required by Figure 1 in RG 1.197 as if the periodic CRE testing were on the normal six-year cycle.**

Please provide an explanation regarding a proposed next test date of April 2025. Given the failure occurred in October 2017 and the program suggests a test 3 years later (October 2020) and the request for a 3 year extension in addition to the program suggested test date would appear to indicate a proposed nominal test date of October 2023.

Response:

The provisions of TS 5.5.14.f are removed from this request. The first sentence of the third paragraph in Section 3.4 of the LAR is hereby changed to: “Therefore, with the last test performed in October 2017, the next CRE unfiltered air inleakage test would be performed no later than October 2023.”

Furthermore, the proposed change to TS 5.5.14 in Section 2.3 of the LAR is hereby changed from:

- NOTE -

The three-year test frequency for the CRE unfiltered air inleakage test failure that occurred in October 2017 may be extended an additional three years.

To:

- NOTE -

The three-year test frequency for the CRE unfiltered air inleakage test failure that occurred in October 2017 may be extended an additional three years, not to exceed October 2023.

An updated TS 5.5.14 markup is provided in Attachment 2 to this submittal.

4) It does not appear appropriate to use the 25% extension of TS 5.5.14.f for any amount of time associated with the proposed request for a three year extension. Please provide further justification for use of the 25% extension for the entire proposed nominal 6 year interval or revise the request to not use the extension.

Response:

With the changes described in the response above, Energy Harbor Nuclear Corp. is no longer requesting that the provisions of TS 5.5.14.f be applicable to this LAR.

5) The NRC staff intends to perform independent calculations, as discussed above, to support its evaluation of this LAR to facilitate the staff's review, please provide additional information describing, for each design basis accident and unit affected by the proposed LAR changes, all the basic parameters used in the radiological dose consequence analyses. To support a timely NRC review, this information should include the CLB value, the revised value where applicable, the basis for any changes to the CLB on all basic parameters whether or not the individual parameter is being changed in separate tables for each affected design basis accident and unit.

Response:

The requested information is provided in Attachment 3 to this submittal. Unless a parameter is identified as applicable to Unit No. 1 (BV1), or Unit No. 2 (BV2), then it applies to both units.

6) **Please provide additional information describing the models, assumptions, and parameter inputs used in BVPS' calculations of the radiological dose consequence analyses affected by the proposed LAR changes for the revised LOCA, SGTR, and RCP LRA. Alternatively, to support a timely NRC review, the calculation packages which typically contain this information and in sufficient detail may be provided for the selected design basis accidents.**

Response:

Additional information describing the models, assumptions, and parameter inputs for the revised radiological dose consequence analyses affected by the proposed LAR is provided in the technical report enclosure to this submittal.

In addition, the calculation packages for the revised loss of coolant accident analysis, the steam generator tube rupture analyses, and the reactor coolant pump locked rotor analysis will be provided within 60 days from April 13, 2020. The additional time is needed to identify potential proprietary information contained in the vendor analyses and provide the redacted versions on the docket, if necessary.

Attachment 2
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TS 5.5.14 Markup
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5.5 Programs and Manuals

5.5.14 Control Room Envelope Habitability Program (continued)

c. --

- NOTE -

The three-year test frequency for the CRE unfiltered air inleakage test failure that occurred in October 2017 may be extended an additional three years, not to exceed October 2023.

Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.15 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

Attachment 3
L-20-097

Input to Radiological Dose Consequence Analyses

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
1	Accident Induced Leakage - Faulted Steam Generator	MSLB BV2	2.1 gpm	8.1 gpm	Maximizing Accident Induced Leakage allowance is part of LAR strategy
2	Activity Concentration Limit: Pre-accident Iodine Spike (= 60 x Tech Spec Equilibrium DE I-131 RCS Iodine Activity Limit)	MSLB BV1	21 µCi/gm DE I-131	No Change	
3	Activity Concentration Limit: Pre-accident Iodine Spike (= 60 x Tech Spec Equilibrium DE I-131 RCS Iodine Activity Limit)	MSLB BV2	21 µCi/gm DE I-131	6 µCi/gm DE I-131	Imposing lower activity limit is related to AIL allowance change
4	Activity Concentration Tech Spec Limit - Primary Coolant (Iodines)	MSLB BV2	0.35 µCi/gm DE I-131	0.10 µCi/gm DE I-131	Imposing lower activity limit is related to AIL allowance change
5	Activity Concentration Tech Spec Limit - Primary Coolant (Iodines)	MSLB BV1	0.35 µCi/gm DE I-131	No Change	
6	Activity Concentration Tech Spec Limit - Secondary Coolant liquid (Iodines)	MSLB BV2	0.10 µCi/gm DE I-131	0.05 µCi/gm DE I-131	Imposing lower activity limit is related to AIL allowance change
7	Activity Concentration Tech Spec Limit - Secondary Coolant liquid (Iodines)	MSLB BV1	0.10 µCi/gm DE I-131	No Change	
8	Activity of Primary Coolant Iodine and Noble Gas Concentrations at Tech Spec Limits (principal dose contributors)	MSLB BV2	based on 0.35 µCi/gm DE I-131 2.74E-01 µCi/gm (I-131) 1.08E-01 µCi/gm (I-132) 4.10E-01 µCi/gm (I-133) 6.00E-02 µCi/gm (I-134) 2.36E-01 µCi/gm (I-135) 3.89E-02 µCi/gm (Kr-83m) 1.35E-01 µCi/gm (Kr-85m) 1.18E+01 µCi/gm (Kr-85) 9.00E-02 µCi/gm (Kr-87) 2.52E-01 µCi/gm (Kr-88) 4.84E-01 µCi/gm (Xe-131m) 3.99E-01 µCi/gm (Xe-133m) 2.95E+01 µCi/gm (Xe-133) 9.09E-02 µCi/gm (Xe-135m) 9.16E-01 µCi/gm (Xe-135)	based on 0.10 µCi/gm DE I-131 7.79E-02 µCi/gm (I-131) 3.22E-02 µCi/gm (I-132) 1.19E-01 µCi/gm (I-133) 1.85E-02 µCi/gm (I-134) 7.04E-02 µCi/gm (I-135) 1.17E-02 µCi/gm (Kr-83m) 4.22E-02 µCi/gm (Kr-85m) 3.71E+00 µCi/gm (Kr-85) 2.77E-02 µCi/gm (Kr-87) 7.85E-02 µCi/gm (Kr-88) 1.58E-01 µCi/gm (Xe-131m) 1.31E-01 µCi/gm (Xe-133m) 9.55E+00 µCi/gm (Xe-133) 2.82E-02 µCi/gm (Xe-135m) 2.93E-01 µCi/gm (Xe-135)	Imposing lower activity limit is related to AIL allowance change

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
9	Activity of Primary Coolant Iodine and Noble Gas Concentrations at Tech Spec Limits (principal dose contributors)	LOCA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1	based on 0.35 µCi/gm DE I-131 2.74E-01 µCi/gm (I-131) 1.08E-01 µCi/gm (I-132) 4.10E-01 µCi/gm (I-133) 6.00E-02 µCi/gm (I-134) 2.36E-01 µCi/gm (I-135) 3.89E-02 µCi/gm (Kr-83m) 1.35E-01 µCi/gm (Kr-85m) 1.18E+01 µCi/gm (Kr-85) 9.00E-02 µCi/gm (Kr-87) 2.52E-01 µCi/gm (Kr-88) 4.84E-01 µCi/gm (Xe-131m) 3.99E-01 µCi/gm (Xe-133m) 2.95E+01 µCi/gm (Xe-133) 9.09E-02 µCi/gm (Xe-135m) 9.16E-01 µCi/gm (Xe-135)	based on 0.35 µCi/gm DE I-131 2.73E-01 µCi/gm (I-131) 1.13E-01 µCi/gm (I-132) 4.17E-01 µCi/gm (I-133) 6.47E-02 µCi/gm (I-134) 2.46E-01 µCi/gm (I-135) 4.09E-02 µCi/gm (Kr-83m) 1.48E-01 µCi/gm (Kr-85m) 1.30E+01 µCi/gm (Kr-85) 9.68E-02 µCi/gm (Kr-87) 2.74E-01 µCi/gm (Kr-88) 5.54E-01 µCi/gm (Xe-131m) 4.59E-01 µCi/gm (Xe-133m) 3.34E+01 µCi/gm (Xe-133) 9.87E-02 µCi/gm (Xe-135m) 1.02E+00 µCi/gm (Xe-135)	Analysis of coolant concentrations was updated
10	Activity of Primary Coolant Iodine Concentrations with Pre-accident Spike (principal dose contributors)	SGTR BV1 SGTR BV2 MSLB BV1	based on 21 µCi/gm DE I-131 1.64E+01 µCi/gm (I-131) 6.46E+00 µCi/gm (I-132) 2.46E+01 µCi/gm (I-133) 3.60E+00 µCi/gm (I-134) 1.41E+01 µCi/gm (I-135)	based on 21 µCi/gm DE I-131 1.64E+01 µCi/gm (I-131) 6.77E+00 µCi/gm (I-132) 2.50E+01 µCi/gm (I-133) 3.88E+00 µCi/gm (I-134) 1.48E+01 µCi/gm (I-135)	Analysis of coolant concentrations was updated
11	Activity of Primary Coolant Iodine Concentrations with Pre-accident Spike (principal dose contributors)	MSLB BV2	based on 21 µCi/gm DE I-131 1.64E+01 µCi/gm (I-131) 6.46E+00 µCi/gm (I-132) 2.46E+01 µCi/gm (I-133) 3.60E+00 µCi/gm (I-134) 1.41E+01 µCi/gm (I-135)	based on 6 µCi/gm DE I-131 4.67E+00 µCi/gm (I-131) 1.93E+00 µCi/gm (I-132) 7.15E+00 µCi/gm (I-133) 1.11E+00 µCi/gm (I-134) 4.22E+00 µCi/gm (I-135)	Imposing lower activity limit is related to AIL allowance change
12	Activity of Secondary Coolant Iodine Liquid based on Tech Spec Limit (principal dose contributors)	MSLB BV2	based on 0.10 µCi/gm DE I-131 8.33E-02 µCi/gm (I-131) 1.40E-02 µCi/gm (I-132) 9.39E-02 µCi/gm (I-133) 1.95E-03 µCi/gm (I-134) 3.39E-02 µCi/gm (I-135)	based on 0.05 µCi/gm DE I-131 4.17E-02 µCi/gm (I-131) 6.93E-03 µCi/gm (I-132) 4.66E-02 µCi/gm (I-133) 9.52E-04 µCi/gm (I-134) 1.67E-02 µCi/gm (I-135)	Imposing lower activity limit is related to AIL allowance change
13	Activity of Secondary Coolant Iodine Liquid based on Tech Spec Limit (principal dose contributors)	LACP SGTR BV1 SGTR BV2 MSLB BV1	based on 0.10 µCi/gm DE I-131 8.33E-02 µCi/gm (I-131) 1.40E-02 µCi/gm (I-132) 9.39E-02 µCi/gm (I-133) 1.95E-03 µCi/gm (I-134) 3.39E-02 µCi/gm (I-135)	based on 0.10 µCi/gm DE I-131 8.34E-02 µCi/gm (I-131) 1.39E-02 µCi/gm (I-132) 9.32E-02 µCi/gm (I-133) 1.90E-03 µCi/gm (I-134) 3.34E-02 µCi/gm (I-135)	Analysis of coolant concentrations was updated

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
14	Aerosol and Elemental Iodine Removal Coefficients - Sprayed Region of Containment	LOCA	Values are listed in BV1 UFSAR Table 14.3-14a & BV2 UFSAR Table 15.6-11	Values are listed in Technical Report Table 7.2-2	Analysis was updated to bound a proposed plant modification
15	Aerosol and Elemental Iodine Removal Coefficients - Unsprayed Region of Containment	LOCA	Values are listed in BV1 UFSAR Table 14.3-14a & BV2 UFSAR Table 15.6-11	Values are listed in Technical Report Table 7.2-2	Analysis was updated to bound a proposed plant modification
16	Appearance Rate - Concurrent Iodine Spike	SGTR BV1 SGTR BV2	335 x Iodine Appearance Rate based on Equilibrium Tech Spec Limit	No Change	
17	Appearance Rate - Concurrent Iodine Spike	SLBOC MSLB BV1 MSLB BV2	500 x Iodine Appearance Rate based on Equilibrium Tech Spec Limit	No Change	
18	Break Flow from RCS to ruptured SG - before reactor trip	SGTR BV1	21,900 lbm	No Change	
19	Break Flow from RCS to ruptured SG - before reactor trip	SGTR BV2	9,200 lbm	No Change	
20	Break Flow from RCS to ruptured SG - following reactor trip	SGTR BV2	197,400 lbm	No Change	
21	Break Flow from RCS to ruptured SG - following reactor trip	SGTR BV1	128,000 lbm	No Change	
22	Breathing Rate - Control Room	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	3.5E-04 m^3/sec (0 to 30 day)	No Change	
23	Breathing Rate - ERF	LOCA	3.5E-04 m^3/sec (0 to 30 day)	No Change	
24	Breathing Rate - Offsite	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	3.5E-04 m^3/sec (0 to 8 hr) 1.8E-04 m^3/sec (8 to 24 hr) 2.3E-04 m^3/sec (1 to 30 day)	No Change	
25	Containment Free Volume - minimum of BV1 or BV2	LOCA CREA	1.750E6 ft^3	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
26	Containment leak rate (volume fractions per day) - maximum	LOCA CREA	0.10% (0 to 1 day) 0.05% (1 to 30 day)	No Change	
27	Containment Mixing Rate	LOCA	2 unsprayed volumes per hour	No Change	
28	Containment Spray Coverage	LOCA	63%	60%	Analysis was updated to bound a proposed plant modification
29	Containment Vacuum Relief Line (bounding) release rate - maximum	LOCA	2,200 scfm for 5 sec	No Change	
30	Control Room Envelope Free Volume - minimum	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	173,000 ft^3	No Change	
31	Control Room Purge flowrate and time - minimum	SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA (BV1)	16,200 cfm for 30 min	No Change	
32	Control Room ventilation emergency intake flow rate (filtered)	LOCA CREA MSLB BV1 MSLB BV2	600 to 1,030 cfm	800 to 1,000 cfm	Change in ventilation flow parameter is proposed in LAR
33	Control Room ventilation emergency recirculation flow rate (filtered)	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	Not credited in analysis	No Change	
34	Control Room ventilation infiltration during Isolation (Recirculation) Mode	LOCA	300 scfm	450 cfm	Change in ventilation flow parameter is proposed in LAR

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
35	Control Room ventilation intake/inleakage flow rate - maximum normal operation	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	500 cfm	1,250 cfm	Change in ventilation flow parameter is proposed in LAR
36	Control Room ventilation unfiltered inleakage during emergency ventilation mode	LOCA CREA MSLB BV1 MSLB BV2	30 cfm	165 cfm	Change in ventilation flow parameter is proposed in LAR
37	Core Activity of Dose Significant Isotopes in the Gap - Alkali Metals & Ba-137M	LRA	1.69E+05 Ci (Rb-86) 5.57E+07 Ci (Rb-88) 7.26E+07 Ci (Rb-89) 6.69E+07 Ci (Rb-90) 2.11E+07 Ci (Rb-90M) 1.57E+07 Ci (Cs-134) 3.69E+06 Ci (Cs-134M) 4.39E+06 Ci (Cs-135M) 4.97E+06 Ci (Cs-136) 9.81E+06 Ci (Cs-137) 1.48E+08 Ci (Cs-138) 1.37E+08 Ci (Cs-139) 1.23E+08 Ci (Cs-140) 9.35E+06 Ci (Ba-137M)	No Change	
38	Core Activity of Dose Significant Isotopes in the Gap - Halogens	CREA LRA	3.02E+05 Ci (Br-82) 9.37E+06 Ci (Br-83) 1.95E+07 Ci (Br-85) 2.86E+00 Ci (I-129) 2.07E+06 Ci (I-130) 7.78E+07 Ci (I-131) 1.14E+08 Ci (I-132) 1.60E+08 Ci (I-133) 1.77E+08 Ci (I-134) 1.52E+08 Ci (I-135) 6.99E+07 Ci (I-136)	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
39	Core Activity of Dose Significant Isotopes in the Gap - Noble Gases	CREA LRA	9.46E+06 Ci (Kr-83M) 8.27E+05 Ci (Kr-85) 1.95E+07 Ci (Kr-85M) 3.91E+07 Ci (Kr-87) 5.43E+07 Ci (Kr-88) 6.75E+07 Ci (Kr-89) 7.24E+07 Ci (Kr-90) 1.08E+06 Ci (Xe-131M) 1.60E+08 Ci (Xe-133) 5.05E+06 Ci (Xe-133M) 4.84E+07 Ci (Xe-135) 3.36E+07 Ci (Xe-135M) 1.46E+08 Ci (Xe-137) 1.36E+08 Ci (Xe-138)	No Change	
40	Core Activity of Dose Significant Isotopes in the Gap with 100 hours decay	FHA	8.27E+05 Ci (Kr-85) 3.77E+00 Ci (Kr-85M) 9.50E+00 Ci (Xe-127) 4.49E+03 Ci (Xe-129M) 1.00E+06 Ci (Xe-131M) 1.11E+08 Ci (Xe-133) 2.07E+06 Ci (Xe-133M) 2.13E+05 Ci (Xe-135) 6.51E+02 Ci (Xe-135M) 4.25E+04 Ci (Br-82) 2.86E+00 Ci (I-129) 7.64E+03 Ci (I-130) 5.62E+07 Ci (I-131) 4.74E+07 Ci (I-132) 5.86E+06 Ci (I-133) 3.98E+03 Ci (I-135)	No Change	
41	Core inventory release fractions, by radionuclide groups, during the Early In-Vessel release phase into containment atmosphere	LOCA	0.95 Noble Gases 0.35 Halogens 0.25 Alkali Metals 0.05 Tellurium Metals 0.02 Barium, Strontium 0.0025 Noble Metals 0.0005 Cerium Group 0.0002 Lanthanides	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
42	Core inventory release fractions, by radionuclide groups, during the Early In-Vessel release phase into containment sump water	LOCA	0.00 Noble Gases 0.35 Halogens 0.25 Alkali Metals 0.05 Tellurium Group 0.02 Barium, Strontium 0.0025 Noble Metals 0.0005 Cerium Group 0.0002 Lanthanides	No Change	
43	Core inventory release fractions, by radionuclide groups, during the Gap release phase into containment atmosphere	LOCA	0.05 Noble Gases 0.05 Halogens 0.05 Alkali Metals	No Change	
44	Core inventory release fractions, by radionuclide groups, during the Gap release phase into containment sump water	LOCA	0.00 Noble Gases 0.05 Halogens 0.05 Alkali Metals	No Change	
45	Core inventory release timing for Early In-Vessel release phase	LOCA	Onset: 30.5 min Duration: 1.3 hr	No Change	
46	Core inventory release timing for Gap release phase	LOCA	Onset: 30 sec Duration: 30 min	No Change	
47	Core Power Level (with power uncertainty) -- Rated Thermal Power plus 0.6% uncertainty	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	2,918 MWt	No Change	
48	CVCS Letdown Line Break - isolation time (manual operator action time)	SLBOC	15 min	No Change	
49	CVCS Letdown Line Break - mass flow rate	SLBOC	16.79 lbm/sec	No Change	
50	Decay time prior to fuel movement	FHA	100 hr	No Change	
51	Decontamination Factor (DF) for Aerosols - maximum	LOCA	No Restriction	No Change	
52	Decontamination Factor (DF) for elemental Iodine - maximum	LOCA	200	No Change	
53	Duration of Concurrent Iodine Spike	SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2	4 hr	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
54	Duration of Control Room Purge credited in analysis	FHA (BV1)	2 to 2.5 hr - BV1 only	No Change	
55	Duration of Control Room Purge credited in analysis	MSLB BV1 MSLB BV2	24 to 24.5 hr	No Change	
56	Duration of Control Room Purge credited in analysis	SGTR BV1 SGTR BV2	8 to 8.5 hr	No Change	
57	Duration of Engineered Safety Features leakage	LOCA	1,200 sec to 30 day	No Change	
58	Duration of Environmental Release	CREA	30 day (Containment) 8 hr (Secondary Side)	30 day (Containment) 30 day (Secondary Side)	RHR is not credited for terminating release in CREA analysis supporting LAR
59	Duration of Environmental Release	LRA LACP	8 hr	No Change	
60	Duration of Environmental Release	MSLB BV1	8 hr (Intact SGs) 19 hr (Faulted SG)	No Change	
61	Duration of Environmental Release	MSLB BV2	8 hr (Intact SGs) 21 hr (Faulted SG)	No Change	
62	Engineered Safety Features leak rate	LOCA	5,700 cc/hr [analysis to use 2x value]	No Change	
63	Equilibrium Core Inventory	LOCA	List of isotopes shown on BV1 UFSAR Table 14.2-12 = BV2 UFSAR Table 15.0-7a	No Change	
64	Filter Efficiency - Control Room intake (Iodine removal)	LOCA CREA MSLB BV1 MSLB BV2	98% (elemental) 98% (organic) 99% (particulate)	No Change	
65	Filter Efficiency - SLCRS particulate and carbon filter	LOCA SLBOC	0% (not credited in analysis)	No Change	
66	Flash Fraction - break flow	SLBOC	37%	No Change	
67	Flash Fraction of Break Flow from RCS to ruptured SG - before reactor trip	SGTR BV1	0.2227	No Change	
68	Flash Fraction of Break Flow from RCS to ruptured SG - following reactor trip	SGTR BV1	0.1645	No Change	
69	Flash portion of Break Flow from RCS to ruptured SG - before reactor trip	SGTR BV2	1,730.2 lbm	No Change	
70	Flash portion of Break Flow from RCS to ruptured SG - following reactor trip to flashing termination	SGTR BV2	6,814.5 lbm	No Change	
71	Form of Iodine activity - after scrubbing	FHA	57% elemental 43% organic	No Change	
72	Form of Iodine activity - before scrubbing	FHA	99.85% elemental 0.15% organic	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
73	Form of Iodine activity available for environmental release	CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2	97% elemental 3% organic	No Change	
74	Form of Iodine activity released from sump water or RCS from melted and failed fuel	LOCA	97% elemental 3% organic	No Change	
75	Form of Iodine activity released to containment atmosphere from melted and failed fuel	LOCA	95% cesium Iodide 4.85% elemental 0.15% organic	No Change	
76	Fraction of Core Activity in the Fuel Gap	CREA	0.10 Noble Gases 0.10 Halogens	No Change	
77	Fraction of Core Activity in the Fuel Gap	LRA FHA	based on RG 1.183 Table 3 0.05 for Noble Gases (except Kr-85) 0.05 for Halogens (except I-131) 0.10 for Kr-85 0.08 for I-131 0.12 for Alkali Metals (Cs, Rb)	based on DG-1199 Table 4 0.04 for Noble Gases (except Kr-85) 0.05 for Halogens (except I-131 & I-132) 0.35 for Kr-85 0.08 for I-131 0.23 for I-132 0.46 for Alkali Metals (Cs, Rb)	LAR proposes using release fractions from DG-1199 in place of RG 1.183 for certain analyses
78	Fraction of Core Activity in the melted Fuel - Containment Leakage	CREA	1.0 Noble Gases 0.25 Halogens	No Change	
79	Fraction of Core Activity in the melted Fuel - Secondary System Release	CREA	1.0 Noble Gases 0.50 Halogens	No Change	
80	Fraction of ESF Iodine activity leakage that flashes when the liquid temperature is less than 212 degF or if the calculated flash fraction is less than 10%	LOCA	0.10	No Change	
81	Frequency of natural deposition of elemental Iodine in the sprayed region	LOCA	2 per hour	0.5358 per hour	Sprayed versus unsprayed surface areas and Containment volume were recharacterized during transition from LOCTIC code to MAAP-DBA code and when addressing GSI-191 while updating analysis

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
82	Iodine Appearance Rate based on Equilibrium Tech Spec Limit	MSLB BV2	based on 0.35 µCi/gm DE I-131 2.53E+03 µCi/sec (I-131) 2.66E+03 µCi/sec (I-132) 4.42E+03 µCi/sec (I-133) 3.00E+03 µCi/sec (I-134) 3.41E+03 µCi/sec (I-135)	based on 0.10 µCi/gm DE I-131 6.48E+02 µCi/sec (I-131) 8.06E+02 µCi/sec (I-132) 1.19E+03 µCi/sec (I-133) 9.71E+02 µCi/sec (I-134) 9.86E+02 µCi/sec (I-135)	Imposing lower activity limit is related to AIL allowance change
83	Iodine Appearance Rate based on Equilibrium Tech Spec Limit	SLBOC SGTR BV1 SGTR BV2 MSLB BV1	based on 0.35 µCi/gm DE I-131 2.53E+03 µCi/sec (I-131) 2.66E+03 µCi/sec (I-132) 4.42E+03 µCi/sec (I-133) 3.00E+03 µCi/sec (I-134) 3.41E+03 µCi/sec (I-135)	based on 0.35 µCi/gm DE I-131 2.27E+03 µCi/sec (I-131) 2.83E+03 µCi/sec (I-132) 4.17E+03 µCi/sec (I-133) 3.39E+03 µCi/sec (I-134) 3.44E+03 µCi/sec (I-135)	Analysis of coolant concentrations was updated
84	Main steam flow (maximum) to condenser before reactor trip	SGTR BV1	1,207.407 lbm/sec per SG	No Change	
85	Main steam flow (maximum) to condenser before reactor trip	SGTR BV2	142,300 lbm (Ruptured SG) 281,900 lbm (Intact SGs)	No Change	
86	Mass - Reactor Coolant - initial and minimum post-accident	CREA LRA LACP SLBOC	340,711 lbm	341,331 lbm	BV1 OSG (340,711 lbm) was replaced by BV1 RSG (345,097 lbm), so BV2 OSG (341,331 lbm) became limiting (minimum).
87	Mass - Reactor Coolant - initial and minimum post-accident	SGTR BV2	368,000 lbm	No Change	
88	Mass - Reactor Coolant - initial and minimum post-accident	SGTR BV1	373,100 lbm	No Change	
89	Mass - Reactor Coolant - minimum post-accident	MSLB BV1	340,711 lbm	345,097 lbm	BV1 OSG (340,711 lbm) was replaced by BV1 RSG (345,097 lbm)
90	Mass - Reactor Coolant - minimum post-accident	MSLB BV2	341,331 lbm	No Change	
91	Mass - Secondary Coolant - initial and minimum post-accident	CREA LRA LACP MSLB BV1	101,799 lbm/SG	No Change	
92	Mass - Secondary Coolant - initial and minimum post-accident	MSLB BV2	105,076 lbm/SG	No Change	
93	Mass - Steam Generator liquid - initial	SGTR BV2	95,150 lbm/SG	No Change	
94	Mass - Steam Generator liquid - initial	SGTR BV1	96,000 lbm/SG	No Change	
95	Mass - Steam Generator liquid - minimum post-accident	SGTR BV1	91,000 lbm/SG (ruptured and intact SGs)	No Change	
96	Mass - Steam Generator liquid - minimum post-accident	SGTR BV2	95,150 lbm/SG (ruptured and intact SGs)	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
97	Moisture Carryover Fraction in SGs (to determine carryover in particulates in fuel gap release)	LRA	0.0025	No Change	
98	Number of Fuel Assemblies in Core	FHA	157	No Change	
99	Number of Rods in a Fuel Assembly	FHA	264	No Change	
100	Number of ruptured Fuel Rods due to dropping an assembly in the reactor cavity or in the spent fuel pool	FHA	137	No Change	
101	Occupancy Factors - Control Room	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	1.0 (0 to 1 day) 0.6 (1 to 4 day) 0.4 (4 to 30 day)	No Change	
102	Occupancy Factors - ERF	LOCA	1.0 (0 to 1 day) 0.6 (1 to 4 day) 0.4 (4 to 30 day)	No Change	
103	Partition Coefficient in Condenser/Air Ejector	SGTR BV1 SGTR BV2	Noble Gas: 1 (all released) Organic Iodine: 1 (all released) Elemental Iodine: 100 (1/100th released)	No Change	
104	Partition Coefficient in Steam Generators - Iodine	MSLB BV1 MSLB BV2	100 (Intact SGs) No retention (faulted SG)	No Change	
105	Partition Coefficient in Steam Generators - Noble Gases	MSLB BV1 MSLB BV2	No retention	No Change	
106	Partition Coefficient in Steam Generators with tubes totally submerged	CREA LRA LACP	Noble Gas: released freely without retention Iodine: 100	No Change	
107	Partition Coefficient in Steam Generators: flashed portion of the rupture flow	SGTR BV1 SGTR BV2	Noble Gas: released freely without retention Iodine: released freely without retention	No Change	
108	Partition Coefficient in Steam Generators: non-flashed portion of the rupture flow and leakage flow in intact SG	SGTR BV1 SGTR BV2	Noble Gas: released freely without retention Iodine: 100	No Change	
109	Percentage - Failed Fuel, maximum	CREA	10%	No Change	
110	Percentage - Failed Fuel, maximum	LRA	20%	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
111	Percentage - Failed Fuel, maximum	LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2	None	No Change	
112	Percentage - Melted Fuel, maximum	CREA	0.25%	No Change	
113	Percentage - Melted Fuel, maximum	LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2	None	No Change	
114	pH (Long Term) sump water, minimum	LOCA	> 7	No Change	
115	Pool Decontamination Factors (DFs) based on 23 ft of water in the reactor cavity/spent fuel pool during fuel handling	FHA	200 (overall Iodine) 1 (Noble Gases) Infinite (Particulates)	No Change	
116	Primary to Secondary Steam Generator tube leakage	SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2	150 gpd/SG	No Change	
117	Primary to Secondary Steam Generator tube leakage	LRA LACP	450 gpd/3 SGs	No Change	
118	Primary to Secondary Steam Generator tube leakage	CREA	450 gpd/3 SGs (0 to 2500 sec) 0 gpd (> 2500 sec, when primary pressure < secondary pressure)	No Change	
119	Radial Peaking Factor to be applied to Failed or Melted Fuel	CREA LRA FHA	1.75	1.70	Peaking factor margin was reduced, but the 1.70 value continues to bound the limit of 1.62 in the BV1 LRM and BV2 LRM
120	Release Rate Coefficients of Gaseous activity from RWST vent	LOCA	See RWST Vent Gas Release Rate chart on BV1 UFSAR Table 14.3-14a = BV2 UFSAR Table 15.6-11	No Change	
121	Release Rate Coefficients of Iodine activity from RWST vent	LOCA	See RWST Vent Iodine Release Rate chart on BV1 UFSAR Table 14.3-14a = BV2 UFSAR Table 15.6-11	No Change	
122	Steam Release (maximum) from intact SGs	MSLB BV1	345,000 lbm (0 to 2 hr) 734,000 lbm (2 to 8 hr)	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
123	Steam Release (maximum) from intact SGs	MSLB BV2	350,000 lbm (0 to 2 hr) 730,000 lbm (2 to 8 hr)	No Change	
124	Steam Release (maximum) from intact SGs via MSSVs/ADVs	SGTR BV2	163,500 lbm (116 to 4076 sec) 216,800 lbm (4076 to 7200 sec) 798,500 lbm (7200 to 28,800 sec)	No Change	
125	Steam Release (maximum) from intact SGs via MSSVs/ADVs	SGTR BV1	417,100 lbm (Rx Trip to 7200 sec) 979,500 lbm (2 to 8 hr) 658,400 lbm (8 to 16 hr) 546,700 lbm (16 to 24 hr)	No Change	
126	Steam Release (maximum) from ruptured SG via MSSVs/ADVs	SGTR BV2	67,300 lbm (116 to 4076 sec) 0 lbm (4076 to 7200 sec) 46,800 lbm (7200 to 28,800 sec)	No Change	
127	Steam Release (maximum) from ruptured SG via MSSVs/ADVs	SGTR BV1	68,900 lbm (Rx Trip to 1800 sec)	No Change	
128	Steam Release (maximum) from SGs	LRA LACP	348,000 lbm 778,000 lbm - BV1 773,000 lbm - BV2	348,000 lbm 778,000 lbm	Bounding values of 348,000 lbm (BV2 OSG) and 778,000 lbm (BV1 RSG) were selected
129	Steam Release via MSSV/ADVs	CREA	900 lbm/sec (0 to 150 sec) 300 lbm/sec (150 to 300 sec) 150 lbm/sec (300 to 2500 sec) 778,000 lbm (2500 sec to 8 hr)	900 lbm/sec (0 to 150 sec) 300 lbm/sec (150 to 300 sec) 150 lbm/sec (300 to 2500 sec) 778,000 lbm (2500 sec to 8 hr) Rate same as previous period (8 hr to 30 day)	RHR is not credited for terminating release in CREA analysis supporting LAR
130	Sump back-leakage into RWST	LOCA	1 gpm [analysis to use 2x value]	No Change	
131	Temperature of sump water after recirculation initiates, maximum	LOCA	250 degF	No Change	
132	Time after LOCA when Control Room is isolated due to Containment Isolation phase B (CIB) signal	LOCA	77 sec	No Change	
133	Time for manual initiation of Emergency Ventilation	LOCA CREA MSLB BV1 MSLB BV2	30 min	No Change	
134	Time of Break Flow Termination	SGTR BV2	4,076 sec	No Change	
135	Time of Containment Spray initiation	LOCA	85.4 sec	77.4 sec	Containment response analysis for LOCA was updated to address GSI-191 changes; BV2 calculated value is bounding for BV1
136	Time of Containment Spray termination	LOCA	96 hr	No Change	
137	Time of Flashing Termination	SGTR BV2	1,932.5 sec	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
138	Time of initiation of RWST Release post accident, minimum	LOCA	3,055 sec	3,039 sec	Containment response analysis for LOCA was updated to address GSI-191 changes; BV1 calculated value is bounding for BV2
139	Time of initiation of sump back-leakage into RWST, minimum	LOCA	1,782 sec	1,768 sec	Containment response analysis for LOCA was updated to address GSI-191 changes; BV1 calculated value is bounding for BV2
140	Time of Reactor Trip	SGTR BV2	116 sec	No Change	
141	Time of Reactor Trip	SGTR BV1	225 sec	No Change	
142	Volume of Sump Water - minimum	LOCA	19,111 ft^3 (20 to 30 min) 25,333 ft^3 (30 min to 2 hr) 43,577 ft^3 (2 hr to 30 day)	No Change	
143	x/Q - BV1 Air Ejector (Turbine Building SE Corner) to BV1 CR Intake	SGTR BV1 MSLB BV1	1.05E-02 sec/m^3 (0 to 2 hr) 7.72E-03 sec/m^3 (2 to 8 hr) 3.01E-03 sec/m^3 (8 to 24 hr) 2.14E-03 sec/m^3 (24 to 96 hr) 2.00E-03 sec/m^3 (96 to 720 hr)	No Change	
144	x/Q - BV1 Containment Edge Release to BV1 CR Intake	LOCA CREA	7.48E-04 sec/m^3 (0 to 2 hr) 5.77E-04 sec/m^3 (2 to 8 hr) 2.53E-04 sec/m^3 (8 to 24 hr) 2.00E-04 sec/m^3 (24 to 96 hr) 1.78E-04 sec/m^3 (96 to 720 hr)	No Change	
145	x/Q - BV1 Containment Top Release to BV1 CR Intake	LOCA CREA	8.16E-04 sec/m^3 (0 to 2 hr) 5.78E-04 sec/m^3 (2 to 8 hr) 2.27E-04 sec/m^3 (8 to 24 hr) 1.71E-04 sec/m^3 (24 to 96 hr) 1.47E-04 sec/m^3 (96 to 720 hr)	No Change	
146	x/Q - BV1 MSSV/ADV to BV1 CR Intake	CREA LRA LACP SGTR BV1 MSLB BV1	1.24E-03 sec/m^3 (0 to 2 hr) 9.94E-04 sec/m^3 (2 to 8 hr) 4.08E-04 sec/m^3 (8 to 24 hr) 3.03E-04 sec/m^3 (24 to 96 hr) 2.51E-04 sec/m^3 (96 to 720 hr)	No Change	
147	x/Q - BV1 RWST Vent to BV1 CR Intake	LOCA	7.34E-04 sec/m^3 (0 to 2 hr) 6.17E-04 sec/m^3 (2 to 8 hr) 2.54E-04 sec/m^3 (8 to 24 hr) 1.96E-04 sec/m^3 (24 to 96 hr) 1.57E-04 sec/m^3 (96 to 720 hr)	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
148	x/Q - BV1 Ventilation Vent Release to BV1 CR Intake	SLBOC FHA	4.75E-03 sec/m^3 (0 to 2 hr) 3.66E-03 sec/m^3 (2 to 8 hr) 1.43E-03 sec/m^3 (8 to 24 hr) 1.02E-03 sec/m^3 (1 to 4 day) 8.84E-04 sec/m^3 (4 to 30 day)	No Change	
149	x/Q - BV2 Air Ejector (Turbine Building NW Corner) to BV2 CR Intake	SGTR BV2 MSLB BV2	1.03E-03 sec/m^3 (0 to 2 hr) 7.84E-04 sec/m^3 (2 to 8 hr) 3.57E-04 sec/m^3 (8 to 24 hr) 2.64E-04 sec/m^3 (24 to 96 hr) 1.86E-04 sec/m^3 (96 to 720 hr)	No Change	
150	x/Q - BV2 Containment Top to ERF	LOCA	7.22E-05 sec/m^3 (0 to 2 hr) 6.43E-05 sec/m^3 (2 to 8 hr) 2.96E-05 sec/m^3 (8 to 24 hr) 2.48E-05 sec/m^3 (24 to 96 hr) 2.15E-05 sec/m^3 (96 to 720 hr)	No Change	
151	x/Q - BV2 MSSV/ADV to BV2 CR Intake	SGTR BV2 MSLB BV2	5.01E-04 sec/m^3 (0 to 2 hr) 3.58E-04 sec/m^3 (2 to 8 hr) 1.61E-04 sec/m^3 (8 to 24 hr) 1.19E-04 sec/m^3 (24 to 96 hr) 8.32E-05 sec/m^3 (96 to 720 hr)	No Change	
152	x/Q - BV2 RWST Vent to ERF	LOCA	9.42E-05 sec/m^3 (0 to 2 hr) 8.37E-05 sec/m^3 (2 to 8 hr) 3.81E-05 sec/m^3 (8 to 24 hr) 2.97E-05 sec/m^3 (24 to 96 hr) 2.58E-05 sec/m^3 (96 to 720 hr)	No Change	
153	x/Q - BV2 Ventilation Vent Release to BV2 CR Intake	FHA	9.39E-04 sec/m^3 (0 to 2 hr)	No Change	
154	x/Q Exclusion Area Boundary	SGTR BV1 MSLB BV1	1.04E-03 sec/m^3	No Change	
155	x/Q Exclusion Area Boundary	FHA	1.04E-03 sec/m^3 (BV1) 1.25E-03 sec/m^3 (BV2)	No Change	
156	x/Q Exclusion Area Boundary	LOCA CREA LRA LACP SLBOC SGTR BV2 MSLB BV2	1.25E-03 sec/m^3	No Change	

ID	Parameter Description	Analysis	Current Licensing Basis Parameter Value	License Amendment Request Parameter (Proposed) Value	Reason for Change
157	x/Q Low Population Zone	LOCA CREA LRA LACP SLBOC SGTR BV1 SGTR BV2 MSLB BV1 MSLB BV2 FHA	6.04E-05 sec/m^3 (0 to 8 hr) 4.33E-05 sec/m^3 (8 to 24 hr) 2.10E-05 sec/m^3 (24 to 96 hr) 7.44E-06 sec/m^3 (96 to 720 hr)	No Change	

Term	Description
AIL	Accident Induced Leakage
BV1	Beaver Valley Unit 1
BV2	Beaver Valley Unit 2
CREA	Control Rod Ejection Accident
DE	Design Equivalent
FHA	Fuel Handling Accident
GSI	Generic Safety Issue
LACP	Loss of AC Power
LAR	License Amendment Request
LOCA	Loss Of Coolant Accident
LRA	Locked Rotor Accident
MSLB	Main Stream Line Break
OSG	Original Steam Generator
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RSG	Replacement Steam Generator
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SLBOC	Small Line Break Outside Containment

Analysis	Computer Code
CREA	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
FHA	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
LACP	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
LOCA	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
LOCA	SW-QADCGGP, NU-222, Ver. 00, Lev. 03, A Combinatorial Geometry Version of QAD-5A
LRA	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
MSLB BV1	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
MSLB BV2	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
SGTR BV1	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
SGTR BV2	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code
SLBOC	PERC2, NU-226, Ver.00, Lev. 02, Passive Evolutionary Regulatory Consequence Code

Enclosure
L-20-097

WECTEC Global Project Services, Inc. Technical Report
(114 pages follow)

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TECHNICAL REPORT

WECTEC Global Project Services, Inc.

IMPACT ON DOSE CONSEQUENCES

PROPOSED CHANGES TO PROMOTE OPERATIONAL FLEXIBILITY

BEAVER VALLEY POWER STATION

Prepared for:
FENOC

August 12, 2019
QA Category 1
Safety Related

Sreela Ferguson ALL (except as noted below)	Prepared By	08-12-2019 Date
Keith Ferguson ALL Tables (except Tables 7.2-2, 7.2-3, 7.2-6)	Prepared By	08-12-2019 Date
Joon Cho Tables 7.2-2, 7.2-3, 7.2-6 & Fig 7.2-1, 7.2-2, 7.2-3	Prepared By	08-12-2019 Date
Joon Cho Sections 7.2.1.2 & 7.2.1.3	Reviewed By	08-12-2019 Date
Keith Ferguson	Reviewed By	08-12-2019 Date
Joseph Baron	Reviewed By	08-12-2019 Date
Sreela Ferguson	Lead Engineer	08-12-2019 Date
Norman Hanley	Project Manager	08-12-2019 Date

RECORD OF REVISIONS

Rev. No.	Description of Changes	Pages Revised	Pages Added	Pages Replaced
0	Original Issue	N/A	N/A	N/A
1	Closure of Open Items	2 & 3	N/A	N/A

OPEN ITEMS

No.	Description	Status
1	Section 2.1: Per NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", licensees who wish to take credit for reduced reactor coolant system iodine activities (below 0.35 microcuries per gram dose equivalent I-131) in the radiological dose calculation, should provide a justification supporting the request that evaluates the release rate data described in Reference 6 (J. P. Adams and C. L. Atwood, "The Iodine Spike Release Rate During a Steam Generator Tube Rupture", Nuclear Technology, Vol. 94, p. 361 (1991). FENOC needs to complete this evaluation. This task will need to be completed prior to submitting the License Amendment Request.	CLOSED
2	Section 7.4: For safe shutdown, BVPS-1 is licensed as a Hot Standby plant and BVPS-2 is licensed as an Approach to Cold Shutdown plant. The Residual Heat Removal Systems, which are used to achieve cold shutdown, are safety related; however, the inside containment RHR pump motors and RHR valve motor operators are not environmentally qualified for a Loss of Coolant Accident (specifically spray). Historically, the RHR system has been credited for shutdown cooling in the BVPS dose consequence analyses of record. Section 7.4 addresses a Main Steam Line Break outside containment. A statement is made that the dose consequences of a MSLB inside containment are not analyzed since they would clearly be bounded by the dose consequences of a MSLB outside containment. Based on engineering judgment, this statement is true if RHR is available for shutdown cooling. However, because the RHR Systems may not be available to achieve cold shutdown following an accident that impacts the inside containment environment, FENOC requires confirmation of this assumption that a MSLB outside containment remains bounding. This task will need to be completed prior to submitting the LAR.	CLOSED

No.	Description	Status
3	Section 7.6: Table 6 of NRC Regulatory Guide 1.183 assumes the Analysis Release Duration for a PWR Locked Rotor Accident is "until cold shutdown is established"; for safe shutdown, however, BVPS-1 is licensed as a Hot Standby plant and BVPS-2 is licensed as an Approach to Cold Shutdown plant, so the Residual Heat Removal Systems may not be available to achieve cold shutdown following a LRA that results in fuel damage and consequential high radiation fields inside containment. Historically, the RHR system has been credited for shutdown cooling in the BVPS dose consequence analyses of record. With up to 20% failed fuel (contained in the Reactor Coolant System) for a LRA, the dose field may result in a significant cumulative dose, especially after RCS flow is initiated through the RHR pumps. Taking into consideration the potential post-LRA dose field, FENOC requires confirmation of the assumption that shutdown cooling via initiation of the RHR System is available at t = 8 hours following a LRA. Section 7.5: Similarly, taking into consideration the potential post-SGTR dose field, FENOC requires confirmation of the assumption that shutdown cooling via initiation of the RHR System is available following a Steam Generator Tube Rupture. This task will need to be completed prior to submitting the LAR.	CLOSED

TABLE OF CONTENTS

1.0 INTRODUCTION

The licensing basis methodology currently utilized to estimate dose consequences following design basis accidents at Beaver Valley Power Station (BVPS) reflects use of Alternative Source Terms (AST) in accordance with the Code of Federal Regulations 10 CFR 50.67 (Reference 1), NUREG-0800, Section 15.0.1 (Reference 2), and NRC Regulatory Guide 1.183 (Reference 3).

Summarized below is the licensing history associated with implementation of AST at BVPS. Also included is a summary of subsequent licensing applications that affected the reported dose consequences of the Loss-of-Coolant Accident (LOCA).

AST was implemented at BVPS in 3 phases - originally via a selective application to revise the Fuel Handling Accident (FHA), followed by the update of the LOCA and the Control Rod Ejection Accident (CREA) in support of operation with atmospheric containments (referred to herein as Atmospheric Containment Conversion), and then finally, update of the remaining design basis accidents, including the FHA, in support of the Extended Power Uprate (EPU). Specifically:

- The licensing application addressing Atmospheric Containment Conversion at BVPS was submitted to the Nuclear Regulatory Commission (NRC) via License Amendment Request (LAR) Nos. 300 and 172 (Reference 4). This application used AST to determine the dose consequences of the affected accidents, i.e., the LOCA and the CREA. The referenced analyses were performed at EPU conditions. The application was approved by the NRC Safety Evaluation Report (SER) for Facility Operating License (OL) Nos. DPR-66 (BVPS-1) and NPF-73 (BVPS-2), via Amendment Nos. 257 and 139, respectively (Reference 5).

- The licensing application addressing Replacement Steam Generators at BVPS-1 was submitted to the NRC via LAR No. 320 (Reference 42). The application was approved by the NRC SER for OL No. DPR-66, via Amendment No. 273 (Reference 15).

- The licensing application addressing Extended Power Uprate at BVPS was submitted to the NRC via LAR Nos. 302 and 173 (Reference 6). This application used AST to determine the dose consequences of all the remaining design basis accidents at EPU conditions. The application was approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 275 and 156, respectively (Reference 7).

- The licensing application addressing the changes in the recirculation spray system pump start signal due to the containment sump screen modification was submitted to the NRC via LAR Nos. 334 and 205 (Reference 8). This application updated the previously submitted AST dose consequences following the LOCA. The application was approved by the NRC SERs for OL Nos. DPR-66 and NPF-73, via Amendment No. 280 (Reference 9) and Amendment No. 164 (Reference 10), respectively.

It is noted that the licensing applications addressing the change in the buffering agent used post-LOCA to maintain a minimum pH of 7 in the sump water at BVPS-1 and BVPS-2, respectively, were submitted to the NRC via LAR No. 10-021 (Reference 26) and LAR No. 08-006 (Reference 27); it was concluded that the previous dose consequences for the LOCA, submitted as part of the recirculation spray system pump start signal due to the containment sump screen modification, remained unaffected. The applications were approved by the NRC

SERs for OL Nos. DPR-66 and NPF-73, via Amendment No. 289 (Reference 28) and Amendment No. 168 (Reference 29), respectively.

As part of a long term objective, and with the intent of providing operational margin, FENOC is updating the BVPS design basis site boundary and control room dose consequence analyses to facilitate relaxation of certain operational limits that have significant effect on plant operation. To that end, FENOC is proposing the following changes with this License Amendment Request:

- Increase in the allowable unfiltered air inleakage into the BVPS Control Room Envelope (CRE). This change is intended to address the fact that recent CRE Tracer Gas Tests indicate unfiltered CRE inleakage that challenge the values used in the current design basis dose consequence analyses.

- Increase flexibility in core design methodology (currently limited by BVPS licensing commitment to Note 11 of NRC Regulatory Guide 1.183, Revision 0). Per this commitment, the current linear heat generation rate is limited to < 6.3 kw/ft peak rod average power for burnups exceeding 54,000 MWD/MTU. It is expected that future BVPS fuel management schemes will be adversely affected by this limitation.

- Increase in the maximum allowed accident-induced steam generator tube leakage at BVPS-2. BVPS-2 is currently operating with its original steam generators.

To that end, FENOC is updating the BVPS dose consequence analyses.

The proposed changes will affect the dose consequences associated with the following accidents that represent the current BVPS licensing basis and are discussed in BVPS-1 UFSAR Chapter 14 and BVPS-2 UFSAR Chapter 15, respectively.

1. Loss of Coolant Accident (LOCA)
2. Control Rod Ejection Accident (CREA)
3. Main Steam Line Break (MSLB) outside Containment
4. Steam Generator Tube Rupture (SGTR)
5. Locked Rotor Accident (LRA)
6. Loss of AC Power (LACP)
7. Fuel Handling Accident (FHA) in the Fuel Pool or in Containment
8. Small Line Break Outside Containment (SLBOC)

2.0 REGULATORY APPROACH

2.1 Proposed Changes to Current Licensing Basis

FENOC proposes to revise the BVPS licensing basis through a) reanalysis of the radiological dose consequences of the design basis accidents listed in BVPS-1 UFSAR Chapter 14 and BVPS-2 UFSAR Chapter 15, respectively, and b) implementation of the following changes in plant operations.

1. Proposed changes to the BVPS Technical Specifications are identified below. The related TS Bases will also be affected.

   - TS 3.4.16 – The BVPS-2 reactor coolant Dose Equivalent (DE) I-131 specific activity shall be reduced from ≤ 0.35 µci/gm to ≤ 0.10 µci/gm DE I-131. This reduction in the allowable activity in the reactor coolant is intended to support increased allowable accident-induced steam generator tube leakage (OPEN ITEM 1).
   - TS Figure 3.4.16-1 – The existing reactor coolant DE I-131 Specific Activity Limit versus Percent Rated Power Figure shall be updated to note that it is being replaced by two Figures. The existing Figure shall be applicable to BVPS-1. The new second Figure shall be applicable to BVPS-2.
   - TS 3.7.13 – The BVPS-2 secondary coolant DE I-131 specific activity shall be reduced from ≤ 0.10 µci/gm to ≤ 0.05 µci/gm DE I-131. This reduction in the allowable activity in the secondary coolant is intended to support increased allowable accident-induced steam generator tube leakage.
   - TS 5.5.7b - The acceptance criteria associated with the in-place test for penetration and system bypass, for the charcoal adsorber in the Control Room Emergency Ventilation System (CREVS) intake filter, shall be increased from < 0.05% to < 0.5%. This change is intended to support operational flexibility.
   - TS 5.5.7c - The acceptance criteria associated with determining the methyl iodide removal efficiency via a laboratory test of a sample of the CREVS intake filter charcoal adsorber shall be increased from ≥ 99% to ≥ 99.5%. This change is intended to ensure that there is no impact on the dose consequence analyses due to the change in TS 5.5.7b.

2. In support of operational flexibility, the allowable unfiltered air inleakage into the CRE during the listed modes of operation of the CR ventilation system is increased as noted below:

   - CR Ventilation Isolation mode – The allowable unfiltered air inleakage will be increased from the current maximum value of 300 cfm to 450 cfm (updated value represents an upper bound analytical value which includes test measurement uncertainties and a 10 cfm allowance for ingress/egress)

   - CR Ventilation Emergency mode – The allowable unfiltered air inleakage will be increased from the current maximum value of 30 cfm to 165 cfm (the updated value represents an upper bound analytical value which includes test measurement uncertainties and a 10 cfm allowance for ingress/egress).

3. In support of operational flexibility, during normal plant operation, the air inflow to the BVPS common control room due to unfiltered intake plus inleakage will be increased from 500 cfm to a maximum of 1250 cfm. The proposed maximum normal operation unfiltered air intake to the control room is an analytical upper bound value that is intended to include a) the CR ventilation intake flow rate (including test measurements uncertainties), and b) all unfiltered air inleakage including a 10 cfm allowance for ingress/egress.

4. Use of fractions of fission product inventory in the gap from Table 3 of NRC Draft Regulatory Guide DG-1199 (Reference 11) when assessing the dose consequences of BVPS Non-LOCA events other than reactivity initiated accidents, where only the fuel clad is postulated to be breached. This change in licensing basis is intended to support flexibility in future BVPS fuel management schemes and is deemed to be acceptable since BVPS falls within, and intends to operate within, the maximum allowable power operating envelope for PWRs shown in Figure 1 of DG-1199.

2.2 Planned Design Modifications

There are no design modifications associated with this application.

2.3 Planned Procedural and Process Updates

Provided below are the key procedures that will be updated prior to implementation, but not before receipt, of an amendment.

1. Update of the BVPS Control Room Ventilation System Tracer Gas Test Procedure to include the new CR air inleakage test acceptance criteria and the range of CRVS ventilation flows deemed to be acceptable by the AST dose consequence analyses. In addition, the Control Room Emergency Ventilation System charcoal adsorber efficiencies will be confirmed to be within specification.

2. Update of the BVPS core design process to include additional verification of core power peaking. Specifically, the limit on the peak rod linear heat generation rate in fuel assemblies during normal operation will be confirmed to remain within the nodal power envelope depicted for Pressurized Water Reactors in Figure 1 of DG-1199. In addition, the radial peaking factor will be confirmed to be less than or equal to the lowered value of 1.70.

3. Update of the Chemistry procedures related to the lowered reactor/secondary coolant activities for BVPS-2.

2.4 Dose Acceptance Criteria

In accordance with the current BVPS licensing basis, the acceptance criteria for dose consequences at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are based on 10 CFR 50.67 and Section 4.4 & Table 6 of Regulatory Guide 1.183 (also noted in Table 1 of

NUREG-0800, Section 15.0.1):

(i) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release should not receive a radiation dose in excess of the accident specific total effective dose equivalent (TEDE) value noted in Table 6 of RG 1.183.

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), should not receive a radiation dose in excess of the accident specific TEDE value noted in Table 6 of RG 1.183.

RG 1.183 (regulatory guidance for accident analyses using AST) does not specifically address ANS Condition II and Condition III scenarios. However, per RG 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of RG 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of RG 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

Thus, the acceptance criteria utilized for the EAB and LPZ doses following a LACP or SLBOC represent the most limiting dose criterion in Table 6 of RG 1.183, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

The acceptance criteria for the Control Room dose is based on 10 CFR 50.67:

i. adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

In accordance with the current BVPS licensing basis, the habitability of the BVPS Emergency Response Facility (which houses the Technical Support Center (TSC)) is assessed for the LOCA using a methodology similar to that utilized for the control room. The assessment is intended to demonstrate compliance from a radiological perspective with paragraph IV.E.8 of Appendix E to 10 CFR Part 50 (Reference 38), regulatory guidance provided in Section 8.2.1, item f, of NUREG-0737 Supplement No. 1 (Reference 14), and the current BVPS licensing basis.

Based on the above, the acceptance criteria for the dose to any person working in the TSC is ≤ 5 rem TEDE for the duration of the accident.

It is noted that, in accordance with the current BVPS licensing basis, the ERF is only assessed for the LOCA.

3.0 COMPUTER CODES

The computer codes utilized in the dose consequence analyses specifically developed to support this application are listed below. The referenced computer codes have been verified and validated for QA Category 1 use under the WECTEC 10CFR50 Appendix B Quality Assurance Program. They have been used extensively to support nuclear power plant design and are a part of the current BVPS licensing basis.

1. WECTEC Proprietary Computer Program ACTIVITY2, "Fission Products in a Nuclear Reactor," NU-014, V01, L04.

2. WECTEC Proprietary Computer Program IONEXCHANGER, "Gamma Activities in Ion Exchangers or Tanks as a Function of Time for Constant Feed Activity," NU-009, V01, L04.

3. WECTEC Proprietary Computer Code, PERC2, "Passive/Evolutionary Regulatory Consequence Code," NU-226, V00, L02.

4. WECTEC Proprietary Computer Code, SW-QADCGGP, "A Combinatorial Geometry Version of QAD-5A," NU-222, V00, L02.

5. WECTEC Proprietary Computer Code, SWNAUA, "Aerosol Behavior in Condensing Atmosphere," NU-185, V02, L01.

4.0 RADIATION SOURCE TERMS

4.1 Core Activity Inventory

The current BVPS licensing basis equilibrium core inventory utilized to support this application is based on a core power level of 2918 MWt (which is derived from a Rated Thermal Power of 2900 MWt plus 0.6% uncertainty), and current licensed values of fuel enrichment and burnup. The methodology used to develop the core inventory, and the associated isotopic listing, is presented in Section 5.3.3.1 and Table 5.3.3-1 of LAR Nos. 300 and 172 and was approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139.

This application does not impact the licensing basis core inventory used for the dose consequence analyses. The design basis core activity of isotopes significant to dose consequences previously provided in Table 5.3.3-1 of LAR Nos. 300 and 172 is reproduced herein in Table 4.1-1.

4.2 Coolant Activity Inventory

The BVPS design basis (1% fuel defects) reactor and secondary coolant isotopic activity inventory discussed in Section 5.3.3.2 of LAR Nos. 300 and 172 was updated to account for minor changes in design input values. The methodology used to establish the updated coolant activity inventory was approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139 and remains unchanged.

4.2.1 <u>Technical Specification Reactor and Secondary Coolant Activity Concentrations</u>

The Technical Specification (TS) reactor and secondary coolant isotopic activity inventory was updated using methodology discussed in Section 5.3.3.2 of LAR Nos. 300 and 172 and approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139.

The updated TS coolant concentrations reflect the updated design basis reactor coolant isotopic mix of fission products discussed above. The TS coolant concentrations are based on the current BVPS TS Limiting Conditions for Operation (LCOs) for BVPS-1 and the proposed BVPS TS LCOs for BVPS-2.

Current BVPS TS LCO 3.4.16 limits the specific activity for iodines in the reactor coolant to 0.35 μCi/gm Dose Equivalent (DE) I-131. With this application, the above LCO will be retained for BVPS-1. The proposed LCO for BVPS-2 will be $\leq$ 0.10 μCi/gm DE I-131.

Current BVPS TS LCO 3.4.16 also limits the non-iodine nuclides that make up > 95% of the reactor coolant gross specific activity with half-lives greater than 15 minutes to 100/$\bar{E}$ μCi/gm. This LCO remains unchanged by this application and remains applicable to both units.

In accordance with the current BVPS licensing basis, the estimated isotopic TS concentration of fission products in the reactor coolant is based on the design basis fission product mix, adjusted to reflect the more limiting of the failed fuel percentages associated with the two limits identified in TS LCO 3.4.16. As noted in Section 5.3.3.2 of LAR Nos. 300 and 172 and approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139, this approach is reasonable as the mix of isotopes in the reactor coolant is determined by the leakage of core activity from the defective fuels and the escape coefficients of the isotopes and its precursors. The above is already factored into the design basis coolant isotopic mix, and the referenced mix is not expected to change (i.e., between iodine vs non-iodine isotopes) such that both TS LCOs co-exist at the same time.

In accordance with the current BVPS licensing basis, the maximum allowable failed fuel percentage in the reactor coolant associated with TS LCO 3.4.16 for iodines is developed using thyroid dose conversion factors (DCF) for I-131, I-132, I-133, I-134, and I-135 obtained from ICRP Publication 30 (Reference 16), as noted in the definition of Dose Equivalent I-131 in Section 1.1 of the BVPS Technical Specifications. The average beta and gamma energies per disintegration used to develop the failed fuel percentage in the reactor coolant associated with TS LCO 3.4.16 for non-iodines are based on References 17, 18, 19 and 20. Consistent with the current BVPS licensing basis, the updated analysis continues to demonstrate that the failed fuel percentage associated with TS Section 3.4.16 for the iodines in the reactor coolant is more limiting.

Current BVPS TS LCO 3.7.13 limits the specific activity for iodines in the secondary coolant to 0.10 µCi/gm DE I-131. With this application, the above LCO will be retained for BVPS-1. The proposed LCO for BVPS-2 will be ≤ 0.05 µCi/gm DE I-131. Using methodology discussed in Section 5.3.3.2 of LAR Nos. 300 and 172 and approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139, the TS iodine dose equivalent I-131 concentrations per iodine nuclide in the secondary coolant are calculated using the same methodology discussed earlier for the reactor coolant.

The noble gas and halogen reactor and secondary coolant TS isotopic activity concentrations for BVPS-1 and BVPS-2 are presented in Table 4.2-1A and Table 4.2-1B, respectively.

Note: The coolant concentrations based on the current BVPS-1 TS LCOs are utilized for all accidents with the exception of the BVPS-2 MSLB, which uses coolant concentrations based on the proposed LCOs for BVPS-2.

4.2.2 Pre-Accident & Concurrent Iodine Spike and Equilibrium Iodine Appearance Rates

In accordance with the current TS LCO for BVPS-1 and the proposed TS LCO for BVPS-2, the pre-accident iodine spike concentration limit in the reactor coolant is 21 µCi/gm DE I-131 (transient Technical Specification limit for full power operation) and 6 µCi/gm DE I-131, respectively (i.e., 60 times the reactor coolant iodine TS concentrations).

The activity in the reactor coolant resulting from a concurrent iodine spike is based on an accident dependent multiplier, times the equilibrium iodine appearance rate. Consistent with the current BVPS licensing basis, the equilibrium appearance rates are conservatively calculated based on the Technical Specification reactor coolant activities, along with the maximum design letdown rate, maximum Technical Specification allowed leakage, and an ion-exchanger iodine removal efficiency of 100%. Maximizing the reactor coolant cleanup results in maximizing the equilibrium iodine appearance rates.

The pre-accident iodine spike concentrations and the equilibrium iodine appearance rates (utilized to develop concurrent iodine spike values) for BVPS-1 and BVPS-2 are presented in Table 4.2-

2A and Table 4.2-2B, respectively.

4.3 Gap Fractions for Non-LOCA Events

Table 3 of RG 1.183 provides the fractions of fission product inventory in the fuel gap for Non-LOCA events that are postulated to result in fuel damage. The referenced gap fractions are contingent upon meeting Note 11 of Table 3 of RG 1.183. Note 11 indicates that the release fractions listed in Table 3 are "acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU." As documented in NRC communications with other licensees (e.g., Millstone, Reference 39), the burnup criterion associated with the maximum allowable linear heat generation rate is applicable to the peak rod average burnup in any assembly and is not limited to assemblies with an average burnup that exceeds 54 GWD/MTU.

BVPS has three design basis non-LOCA accidents that are postulated to result in fuel damage, i.e., the Control Rod Ejection Accident (CREA), Locked Rotor Accident (LRA), and Fuel Handling Accident (FHA).

Consistent with the current BVPS licensing basis and in accordance with Appendix H of RG 1.183 (and Note 11 of Table 3), the gap fraction associated with the CREA remains as follows:

Noble Gases: 10%
Halogens: 10%

With this application, and similar to the recently approved AST LAR for Diablo Canyon Power Plant (DCPP), FENOC is proposing to use the gap fractions provided in Table 3 of DG-1199 for all BVPS Non-LOCA events other than reactivity initiated accidents, where only the fuel clad is postulated to be breached. In accordance with the NRC SER (Reference 12) issued in response to the DCPP AST LAR, this approach is acceptable as long as the licensee can demonstrate that plant operation falls within, and intends to operate within, the maximum allowable power operating envelope for PWRs shown in Figure 1 of DG-1199.

In summary, the gap fractions used to assess the dose consequences of the LRA and FHA are as follows:

Nuclide Group	Gap Fraction LRA/FHA (based on DG-1199)
I-131	0.08
I-132	0.23
Kr-85	0.35
Other Noble Gases	0.04
Other Halogens	0.05
Alkali Metals	0.46

The core inventory of noble gases, halogens and alkali metals is presented in Table 4.3-1. These values are consistent with the values presented for these isotopes herein in Table 4.1-1, and in LAR Nos. 300 and 172, Table 5.11.4-3.

Table 4.1-1
BVPS-1 & BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWt)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
AG-111			5.05E+06		PU-239			2.86E+04
	PARENT:	AG-111M	5.06E+06			PARENT:	NP-239	1.66E+09
	GRAND PARENT:	PD-111	5.04E+06			GRAND PARENT:	U-239	1.66E+09
AG-112			2.28E+06		PU-240			3.87E+04
	PARENT:	PD-112	2.27E+06			PARENT:	NP-240	4.32E+06
AM-241			1.17E+04		PU-241			1.13E+07
	PARENT:	PU-241	1.13E+07		PU-242			2.01E+02
BA-137M			9.35E+06			PARENT:	AM-242	7.04E+06
	PARENT:	CS-137	9.81E+06		RB-86			1.69E+05
	GRAND PARENT:	XE-137	1.46E+08		RB-88			5.57E+07
BA-139			1.41E+08			PARENT:	KR-88	5.43E+07
	PARENT:	CS-139	1.37E+08			GRAND PARENT:	BR-88	2.99E+07
	GRAND PARENT:	XE-139	1.01E+08		RB-89			7.26E+07
BA-140			1.42E+08			PARENT:	KR-89	6.75E+07
	PARENT:	CS-140	1.23E+08			GRAND PARENT:	BR-89	2.08E+07
	GRAND PARENT:	XE-140	7.06E+07		RB-90			6.69E+07
BA-142			1.21E+08			PARENT:	KR-90	7.24E+07
	PARENT:	CS-142	5.48E+07			GRAND PARENT:	BR-90	1.13E+07
	GRAND PARENT:	XE-142	1.07E+07			2ND PARENT:	RB-90M	2.11E+07
BR-82			3.02E+05		RB-90M			2.11E+07
	PARENT:	BR-82M	2.62E+05			PARENT:	KR-90	7.24E+07
BR-83			9.37E+06			GRAND PARENT:	BR-90	1.13E+07
	PARENT:	SE-83M	4.69E+06		RH-103M			1.26E+08
	2ND PARENT:	SE-83	4.42E+06			PARENT:	RU-103	1.26E+08
BR-85			1.95E+07		RH-105			8.16E+07

Table 4.1-1 (Continued)
BVPS-1 & BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
CE-141			1.30E+08			PARENT:	RH-105M	2.53E+07
	PARENT:	LA-141	1.29E+08			GRAND PARENT:	RU-105	8.90E+07
	GRAND PARENT:	BA-141	1.28E+08			2ND PARENT:	RU-105	8.90E+07
CE-143			1.21E+08		RH-105M			2.53E+07
	PARENT:	LA-143	1.20E+08			PARENT:	RU-105	8.90E+07
CE-144			9.82E+07			GRAND PARENT:	TC-105	8.76E+07
CM-242			4.22E+06		RH-106			5.13E+07
	PARENT:	AM-242	7.04E+06			PARENT:	RU-106	4.63E+07
CM-244			5.97E+05		RU-103			1.26E+08
	PARENT:	AM-244	1.89E+07			GRAND PARENT:	MO-103	1.24E+08
CS-134			1.57E+07		RU-106			4.63E+07
	PARENT:	CS-134M	3.69E+06			2ND PARENT:	SN-125M	1.20E+06
CS-134M			3.69E+06		SB-127			6.92E+06
CS-135M			4.39E+06			PARENT:	SN-127	2.78E+06
CS-136			4.97E+06			2ND PARENT:	SN-127M	3.76E+06
CS-137			9.81E+06		SB-129			2.52E+07
	PARENT:	XE-137	1.46E+08			PARENT:	SN-129	9.90E+06
	GRAND PARENT:	I-137	7.47E+07			2ND PARENT:	SN-129M	9.29E+06
CS-138			1.48E+08		SB-130			8.37E+06
	PARENT:	XE-138	1.36E+08		SB-130M			3.47E+07
	GRAND PARENT:	I-138	3.80E+07			PARENT:	SN-130	2.61E+07
CS-139			1.37E+08		SB-131			6.09E+07
	PARENT:	XE-139	1.01E+08			PARENT:	SN-131	2.24E+07
	GRAND PARENT:	I-139	1.83E+07		SB-132			3.67E+07
CS-140			1.23E+08			PARENT:	SN-132	1.81E+07
	PARENT:	XE-140	7.06E+07		SB-133			5.08E+07

Table 4.1-1 (Continued)
BVPS-1 & BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT:	I-140	4.81E+06		SE-83			4.42E+06
	PARENT:	SM-155	3.11E+06		SM-153			4.02E+07
EU-156			2.29E+07			PARENT:	PM-153	7.37E+06
	PARENT:	SM-156	1.93E+06		SN-127			2.78E+06
EU-157			2.41E+06		SR-89			7.61E+07
H-3			4.36E+04			PARENT:	RB-89	7.26E+07
I-129			2.86E+00			GRAND PARENT:	KR-89	6.75E+07
	PARENT:	TE-129	2.40E+07		SR-90			7.21E+06
	GRAND PARENT:	TE-129M	4.87E+06			PARENT:	RB-90	6.69E+07
	2ND PARENT:	TE-129M	4.87E+06			GRAND PARENT:	KR-90	7.24E+07
I-130			2.07E+06			2ND PARENT:	RB-90M	2.11E+07
	PARENT:	I-130M	1.10E+06		SR-91			9.50E+07
I-131			7.78E+07			PARENT:	RB-91	8.85E+07
	PARENT:	TE-131	6.54E+07			GRAND PARENT:	KR-91	4.98E+07
	GRAND PARENT:	TE-131M	1.57E+07		SR-92			1.01E+08
	2ND PARENT:	TE-131M	1.57E+07			PARENT:	RB-92	7.83E+07
I-132			1.14E+08			GRAND PARENT:	KR-92	2.66E+07
	PARENT:	TE-132	1.12E+08		SR-93			1.14E+08
	GRAND PARENT:	SB-132	3.67E+07			GRAND PARENT:	KR-93	9.04E+06
I-133			1.60E+08		SR-94			1.14E+08
	PARENT:	TE-133	8.66E+07			GRAND PARENT:	KR-94	4.18E+06
	GRAND PARENT:	SB-133	5.08E+07		TC-99M			1.29E+08
	2ND PARENT:	TE-133M	7.12E+07			PARENT:	MO-99	1.45E+08
I-134			1.77E+08			GRAND PARENT:	NB-99	8.50E+07
	PARENT:	TE-134	1.41E+08		TC-101			1.33E+08
	2ND PARENT:	I-134M	1.59E+07			PARENT:	MO-101	1.33E+08

Table 4.1-1 (Continued)
BVPS-1 & BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
I-135			1.52E+08		TC-104			1.05E+08
I-136			6.99E+07			PARENT:	MO-104	9.99E+07
KR-83M			9.46E+06		TC-105			8.76E+07
	PARENT:	BR-83	9.37E+06			PARENT:	MO-105	7.38E+07
	GRAND PARENT:	SE-83M	4.69E+06		TE-127			6.81E+06
KR-85			8.27E+05			PARENT:	TE-127M	1.13E+06
	PARENT:	KR-85M	1.95E+07			GRAND PARENT:	SB-127	6.92E+06
	GRAND PARENT:	BR-85	1.95E+07			2ND PARENT:	SB-127	6.92E+06
	2ND PARENT:	BR-85	1.95E+07		TE-127M			1.13E+06
KR-85M			1.95E+07			PARENT:	SB-127	6.92E+06
	PARENT:	BR-85	1.95E+07			GRAND PARENT:	SN-127	2.78E+06
KR-87			3.91E+07		TE-129			2.40E+07
	PARENT:	BR-87	3.09E+07			PARENT:	TE-129M	4.87E+06
KR-88			5.43E+07			GRAND PARENT:	SB-129	2.52E+07
	PARENT:	BR-88	2.99E+07			2ND PARENT:	SB-129	2.52E+07
KR-89			6.75E+07		TE-129M			4.87E+06
	PARENT:	BR-89	2.08E+07			PARENT:	SB-129	2.52E+07
KR-90			7.24E+07			GRAND PARENT:	SN-129	9.90E+06
	PARENT:	BR-90	1.13E+07		TE-131			6.54E+07
LA-140			1.46E+08			PARENT:	SB-131	6.09E+07
	PARENT:	BA-140	1.42E+08			GRAND PARENT:	SN-131	2.24E+07
	GRAND PARENT:	CS-140	1.23E+08			2ND PARENT:	TE-131M	1.57E+07
LA-141			1.29E+08		TE-131M			1.57E+07
	PARENT:	BA-141	1.28E+08			PARENT:	SB-131	6.09E+07
LA-142			1.26E+08			GRAND PARENT:	SN-131	2.24E+07
	PARENT:	BA-142	1.21E+08		TE-132			1.12E+08

Table 4.1-1 (Continued) BVPS-1 & BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWth)								
ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT:	CS-142	5.48E+07			PARENT:	SB-132	3.67E+07
LA-143			1.20E+08			GRAND PARENT:	SN-132	1.81E+07
MO-99			1.45E+08		TE-133			8.66E+07
	PARENT:	NB-99M	5.82E+07			PARENT:	TE-133M	7.12E+07
	2ND PARENT:	NB-99	8.50E+07			GRAND PARENT:	SB-133	5.08E+07
MO-101			1.33E+08			2ND PARENT:	SB-133	5.08E+07
NB-95			1.34E+08		TE-133M			7.12E+07
	PARENT:	ZR-95	1.33E+08			PARENT:	SB-133	5.08E+07
	GRAND PARENT:	Y-95	1.28E+08		TE-134			1.41E+08
	2ND PARENT:	NB-95M	1.52E+06		XE-131M			1.08E+06
NB-95M			1.52E+06			PARENT:	I-131	7.78E+07
	PARENT:	ZR-95	1.33E+08			GRAND PARENT:	TE-131M	1.57E+07
	GRAND PARENT:	Y-95	1.28E+08		XE-133			1.60E+08
NB-97			1.27E+08			PARENT:	I-133	1.60E+08
	PARENT:	NB-97M	1.19E+08			GRAND PARENT:	TE-133M	7.12E+07
	GRAND PARENT:	ZR-97	1.26E+08			2ND PARENT:	XE-133M	5.05E+06
	2ND PARENT:	ZR-97	1.26E+08		XE-133M			5.05E+06
NB-97M			1.19E+08			PARENT:	I-133	1.60E+08
	PARENT:	ZR-97	1.26E+08			GRAND PARENT:	TE-133M	7.12E+07
ND-147			5.22E+07		XE-135			4.84E+07
	PARENT:	PR-147	5.18E+07			PARENT:	I-135	1.52E+08
	GRAND PARENT:	CE-147	4.92E+07			2ND PARENT:	XE-135M	3.36E+07
NP-239			1.66E+09		XE-135M			3.36E+07
	GRAND PARENT:	PU-243	4.23E+07			PARENT:	I-135	1.52E+08
	2ND PARENT:	U-239	1.66E+09		XE-137			1.46E+08
PD-109			3.26E+07			PARENT:	I-137	7.47E+07

Table 4.1-1 (Continued)
BVPS-1 & BVPS-2 Equilibrium Core Inventory (Power Level: 2918 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)		ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
PM-147			1.38E+07		XE-138			1.36E+08
	PARENT:	ND-147	5.22E+07			PARENT:	I-138	3.80E+07
	GRAND PARENT:	PR-147	5.18E+07		Y-90			7.49E+06
PM-148			1.41E+07			PARENT:	SR-90	7.21E+06
	PARENT:	PM-148M	2.37E+06			GRAND PARENT:	RB-90	6.69E+07
PM-148M			2.37E+06		Y-91			9.87E+07
PM-149			4.82E+07			PARENT:	SR-91	9.50E+07
	PARENT:	ND-149	3.02E+07			GRAND PARENT:	RB-91	8.85E+07
	GRAND PARENT:	PR-149	2.80E+07			2ND PARENT:	Y-91M	5.51E+07
PM-151			1.60E+07		Y-91M			5.51E+07
	PARENT:	ND-151	1.58E+07			PARENT:	SR-91	9.50E+07
PR-142			5.57E+06			GRAND PARENT:	RB-91	8.85E+07
PR-143			1.18E+08		Y-92			1.02E+08
	PARENT:	CE-143	1.21E+08			PARENT:	SR-92	1.01E+08
	GRAND PARENT:	LA-143	1.20E+08			GRAND PARENT:	RB-92	7.83E+07
PR-144			9.89E+07		Y-93			7.73E+07
	PARENT:	CE-144	9.82E+07			PARENT:	SR-93	1.14E+08
	2ND PARENT:	PR-144M	1.38E+06		Y-94			1.23E+08
PU-238			3.40E+05			PARENT:	SR-94	1.14E+08
	2ND PARENT:	NP-238	3.98E+07		Y-95			1.28E+08
					ZR-95			1.33E+08
						PARENT:	Y-95	1.28E+08
					ZR-97			1.26E+08

Table 4.2-1A
BVPS-1 Reactor and Secondary Coolant
Technical Specification Iodine and Noble Gas Activity Concentrations

	0.35 µCi/gm DE I-131	0.10 µCi/gm DE I-131
Nuclide	Reactor Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	2.73E-01	8.34E-02
I-132	1.13E-01	1.39E-02
I-133	4.17E-01	9.32E-02
I-134	6.47E-02	1.90E-03
I-135	2.46E-01	3.34E-02
Kr-83m	4.09E-02	--
Kr-85m	1.48E-01	--
Kr-85	1.30E+01	--
Kr-87	9.68E-02	--
Kr-88	2.74E-01	--
Xe-131m	5.54E-01	--
Xe-133m	4.59E-01	--
Xe-133	3.34E+01	--
Xe-135m	9.87E-02	--
Xe-135	1.02E+00	--

Table 4.2-1B
Proposed BVPS-2 Reactor and Secondary Coolant
Technical Specification Iodine and Noble Gas Activity Concentrations

	0.10 µCi/gm DE 1-131	0.05 µCi/gm DE I-131
Nuclide	Reactor Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	7.79E-02	4.17E-02
I-132	3.22E-02	6.93E-03
I-133	1.19E-01	4.66E-02
I-134	1.85E-02	9.52E-04
I-135	7.04E-02	1.67E-02
Kr-83m	1.17E-02	--
Kr-85m	4.22E-02	--
Kr-85	3.71E+00	--
Kr-87	2.77E-02	--
Kr-88	7.85E-02	--
Xe-131m	1.58E-01	--
Xe-133m	1.31E-01	--
Xe-133	9.55E+00	--
Xe-135m	2.82E-02	--
Xe-135	2.93E-01	--

Table 4.2-2A
BVPS-1 Reactor Coolant 0.35 µCi/gm DE I-131 Pre-Accident Iodine Spike Activity Concentrations and Equilibrium Iodine Appearance Rates

Nuclide	Pre-Accident Iodine Spike Activity Concentrations (µCi/gm)	Equilibrium Iodine Appearance Rates (µCi/sec)
I-131	16.4	2.27E+03
I-132	6.77	2.83E+03
I-133	25.0	4.17E+03
I-134	3.88	3.39E+03
I-135	14.8	3.44E+03

Table 4.2-2B
Proposed BVPS-2 Reactor Coolant 0.10 µCi/gm DE I-131 Pre-Accident Iodine Spike Activity Concentrations and Equilibrium Iodine Appearance Rates

Nuclide	Pre-Accident Iodine Spike Activity Concentrations (µCi/gm)	Equilibrium Iodine Appearance Rates (µCi/sec)
I-131	4.67E+00	6.48E+02
I-132	1.93E+00	8.06E+02
I-133	7.15E+00	1.19E+03
I-134	1.11E+00	9.71E+02
I-135	4.22E+00	9.86E+02

Table 4.3-1 BVPS Core Inventory of Dose Significant Isotopes in the Gap (2918 MWt)					
Noble Gases		Halogens		Alkali Metals & Ba-137M	
Nuclide	Core Activity (Ci)	Nuclide	Core Activity (Ci)	Nuclide	Core Activity (Ci)
Kr-83M	9.46E+06	Br-82	3.02E+05	Rb-86	1.69E+05
Kr-85	8.27E+05	Br-83	9.37E+06	Rb-88	5.57E+07
Kr-85M	1.95E+07	Br-85	1.95E+07	Rb-89	7.26E+07
Kr-87	3.91E+07	–	–	Rb-90	6.69E+07
Kr-88	5.43E+07	–	–	Rb-90M	2.11E+07
Kr-89	6.75E+07	–	–	–	–
Kr-90	7.24E+07	I-129	2.86E+00	Cs-134	1.57E+07
		I-130	2.07E+06	Cs-134M	3.69E+06
Xe-131M	1.08E+06	I-131	7.78E+07	Cs-135M	4.39E+06
Xe-133	1.60E+08	I-132	1.14E+08	Cs-136	4.97E+06
Xe-133M	5.05E+06	I-133	1.60E+08	Cs-137	9.81E+06
Xe-135	4.84E+07	I-134	1.77E+08	Cs-138	1.48E+08
Xe-135M	3.36E+07	I-135	1.52E+08	Cs-139	1.37E+08
Xe-137	1.46E+08	I-136	6.99E+07	Cs-140	1.23E+08
Xe-138	1.36E+08	–	–	Ba-137M	9.35E+06

Note: Values reported reflect the isotopic gap activity in the equilibrium core immediately after shutdown. These values have to be adjusted for a) the failed fuel percentage, b) radial peaking factor and c) number of fuel rods damaged (if applicable), prior to assessing the associated dose consequences of gap releases following the CREA, LRA and FHA. It is noted that the gap activity utilized for a FHA will also need to reflect a decay of 100 hours, which is the earliest allowable time for fuel movement after reactor shutdown.

5.0 ACCIDENT ATMOSPHERIC DISPERSION FACTORS (χ/Q)

5.1 Site Boundary Atmospheric Dispersion Factors

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are consistent with the current BVPS licensing basis. These values are presented in Table 5.1-1.

In accordance with the current BVPS licensing basis, with the exception of the MSLB, SGTR and FHA which have been analyzed specific to each unit, for the purposes of performing bounding analyses representative for both units, the BVPS-2 EAB χ/Q values are utilized. As noted in Table 5.1-1, the LPZ χ/Q values are the same for both units.

Table 5.1-1
BVPS Site Boundary Atmospheric Dispersion Factors (sec/m^3)

Exclusion Area Boundary Averaging Period				
Release Point	**0 to 2 hr**			
BVPS-1 Release Points	1.04E-3			
BVPS-2 Release Points	1.25E-3			
Low Population Zone Averaging Period				
Release Point	**0 to 8 hr**	**8 to 24 hr**	**1 to 4 day**	**4 to 30 day**
BVPS-1 & BVPS-2 Release Points	6.04E-5	4.33E-5	2.10E-5	7.44E-6

5.2 On-Site Atmospheric Dispersion Factors

The Control Room and Emergency Response Facility atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application. They were originally established by Atmospheric Containment Conversion LAR Nos. 300 and 172 (for the LOCA and the CREA), and EPU LAR Nos. 302 and 173 (for the remaining accidents). These χ/Q values were approved by the NRC SERs for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139 (Atmospheric Containment Conversion) and Amendment Nos. 275 and 156 (Extended Power Uprate), respectively, and are considered the current BVPS licensing basis. A complete listing of all the BVPS-1 and BVPS-2 χ/Q values for all potential release-receptor combinations applicable to the design basis accidents is discussed in Section 5.11.9.1 of LAR Nos. 302 and 173 and summarized therein in Table 5.11.9-2a and Table 5.11.9-2b, respectively.

The χ/Q values for the bounding CR and ERF release-receptor combinations utilized in the dose consequence analyses are summarized with the accident specific input tables provided in Section 7.

As noted previously in LAR Nos. 300 and 172 and LAR Nos. 302 and 173, and as confirmed below, the control room air intake χ/Q values remain representative of the worst case χ/Q values for control room unfiltered air in-leakage (including that associated with CR ingress/egress).

- Component tests performed as part of the 2017 CR Inleakage Tracer Gas Test indicated that potential sources of unfiltered air inleakage into the Control Room are the normal operation intake dampers - which can be assigned the same χ/Q as the Control Room air intakes.

- Regarding other potential locations of air inleakage, a χ/Q value that reflects the center of the Control Room boundary at roof level as a receptor could be considered the average value applicable to unfiltered inleakage locations around the CRE, and thus representative for all CR unfiltered leakage locations. However, since the post-accident release points are a) closer to the CR intakes, and b) the directions from the release points to the CR center and CR intakes are similar, use of χ/Q values associated with the CR intakes, for CR unfiltered air inleakage, would be conservative.

- The 10 cfm allowance for ingress/egress is assigned to the door leading into the Control Room that is considered the primary point of access. This door (2-S-35-71) is located at grade level on the side of the building facing the BVPS-1 Containment and between the CR air intakes. It is located close enough to the air intakes to allow the assumption that the χ/Q associated with this source of inleakage would be reasonably similar to that associated with the air intakes.

6.0 DOSE CALCULATION METHODOLOGY

6.1 Inhalation and Submersion Doses from Airborne Radioactivity

The dose calculation methodology is similar to that outlined in Section 5.3.5 of LAR Nos. 300 and 172 and approved by the NRC SER for OL Nos. DPR-66 and NPF-73, via Amendment Nos. 257 and 139; and is the current BVPS licensing basis.

As noted in LAR Nos. 300 and 172, computer program PERC2 is used to calculate the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from submersion due to halogens, noble gases and other nuclides transported to offsite locations and in the control room. The CEDE is calculated using Federal Guidance Report No. 11 (Reference 21) dose conversion factors which uses methodology provided in ICRP Publication 30 (Reference 16). The committed doses to other organs due to inhalation of halogens, noble gas, other nuclides and their daughters are also calculated. PERC2 is a multiple compartment activity transport code with the dose model consistent with the regulatory guidance. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included.

The PERC2 activity transport model first calculates the integrated activity (using a closed form integration solution) at the offsite locations and in the control room air region, and then calculates the cumulative doses as described below:

<u>Committed Effective Dose Equivalent (CEDE) Inhalation Dose</u> – The dose conversion factors by isotope and internal organ type are applied to the activity in the air space of the control room, or at the EAB/LPZ. The exposure is adjusted by the appropriate respiration rate and occupancy factors for the CR dose at each integration interval as follows:

$$Dh(j) = A(j) \times h(j) \times C2 \times C3 \times CB \times CO$$

Where:

Dh(j)	=	Committed Effective Dose Equivalent (rem) from isotope j
A(j)	=	Integrated Activity (Ci-s/m^3)
h(j)	=	Isotope j Committed Effective Dose Equivalent (CEDE) dose conversion factor (mrem/pCi) based on Federal Guidance Report No. 11
C2	=	Unit conversion of 1×10^{12} pCi/Ci
C3	=	Unit conversion of 1×10^{-3} rem/mrem
CB	=	Breathing rate (m^3/s)
CO	=	Occupancy factor

<u>Deep Dose Equivalent (DDE) from External Exposure</u> – According to the guidance provided in Section 4.1.4 and Section 4.2.7 of RG 1.183, the Effective Dose Equivalent (EDE) may be used in lieu of DDE in determining the contribution of external dose to the TEDE if the whole body is irradiated uniformly. The EDE in the control room is based on a finite cloud model that addresses

buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the control room. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes the integrated activity in the control room air space, the photon energy release rates per energy group from activity airborne in the control room, and ANSI/ANS 6.1.1-1991 (Reference 22).

The Deep Dose Equivalent at the EAB and LPZ locations is very conservatively calculated using the semi-infinite cloud model outlined in Section 7-5.2, Equation 7.36, of TID-24190 (Reference 23), where 1 rad is assumed to be equivalent to 1 rem.

$$\gamma D\infty(x,y,0) \text{ rad} = 0.25\ E\gamma_{BAR}\ \psi(x,y,0)$$

$E\gamma_{BAR}$	=	average gamma released per disintegration (Mev/dis)
$\psi(x,y,0)$	=	concentration time integral (Ci-sec/m^3)
0.25	=	$[1.11 \times 1.6\text{x}10^{-6} \times 3.7\text{x}10^{10}] / [1293 \times 100 \times 2]$

where:

1.11	=	ratio of electron densities per gm of tissue to per gm of air
$1.6\text{x}10^{-6}$ (erg/Mev)	=	number of ergs per Mev
$3.7\text{x}10^{10}$ (dis/sec-Ci)	=	disintegration rate per curie
1293 (g/m^3)	=	density of air at Standard Temperature and Pressure (STP)
100	=	ergs per gram per rad
2	=	factor for converting an infinite to a semi-infinite cloud

6.2 Direct Shine Dose from External and Contained Sources

Point kernel shielding computer program SW-QADCGGP is used to calculate the deep dose equivalent (DDE) in the control room and in the ERF/TSC due to external and contained sources. The calculated DDE is added to the inhalation (CEDE) and the submersion (EDE) dose due to airborne radioactivity to develop the final TEDE. Conservative build-up factors are used and the geometry models are prepared to ensure that un-accounted streaming/scattering paths were eliminated. The dose albedo method with conservative albedo values is used to estimate the scatter dose in situations where the scattering contributions are potentially significant. ANSI/ANS 6.1.1-1977 (Reference 24) is used to convert the gamma flux to the dose equivalent rate.

7.0 RADIOLOGICAL CONSEQUENCES

As discussed in Section 1, the existing design basis accident analyses discussed in the BVPS-1 UFSAR and BVPS-2 UFSAR and listed below, are being updated as part of a long term objective of providing operational margin.

1. Loss of Coolant Accident (LOCA)
2. Control Rod Ejection Accident (CREA)
3. Main Steam Line Break (MSLB) outside Containment
4. Steam Generator Tube Rupture (SGTR)
5. Locked Rotor Accident (LRA)
6. Loss of AC Power (LACP)
7. Fuel Handling Accident (FHA) in the Fuel Pool or in Containment
8. Small Line Break Outside Containment (SLBOC)

FENOC is updating the BVPS dose consequence analyses to reflect the following proposed combination of changes to design input, methodology and operational limits:

- Increased analytical limits to control room unfiltered air inleakage and updated ventilation flow rates during all modes of Control Room Ventilation System operation; i.e., normal, isolation, and emergency.

- Similar to the recently approved AST LAR for Diablo Canyon Power Plant (DCPP), FENOC is proposing to use the gap fractions provided in Table 3 of DG-1199 for all BVPS Non-LOCA events (with the exception of the CREA) that are postulated to result in fuel damage. In accordance with the NRC SER (Reference 12) issued in response to the DCPP AST LAR, this approach is acceptable as long as the licensee can demonstrate that plant operation falls within, and intends to operate within, the maximum allowable power operating envelope for Pressurized Water Reactors shown in Figure 1 of DG-1199.

- Reduction in the reactor and secondary coolant specific activity limits for BVPS-2 allowed by the BVPS Technical Specifications. (Credit for the reduced activity limits will however be taken only for the BVPS-2 Main Steam Line Break analysis. All other BVPS-2 dose consequence analyses will continue to use the current TS activity limits that will remain applicable to BVPS-1.)

Also included in the update of the dose consequence analyses are the

- Use of design input parameter values intended to bound BVPS-1 operation, BVPS-2 operation, or a combination of both, as appropriate.

- Minor updates to some of the design input values as a result of the re-verification effort undertaken as part of the analyses update. Included in this category are design input values that were affected by the impact of Westinghouse NSAL-11-5 (Reference 13), on the LOCA Mass & Energy Release Rates.

In accordance with the current BVPS licensing basis, the updated analyses continue to reflect the following:

- With the exception of the MSLB, SGTR, and FHA dose consequence analyses which are unit specific, the remaining analyses are performed with bounding parameter values intended to encompass an event at either unit.
- The BVPS-2 MSLB dose analysis reflects the use of Alternative Repair Criteria (ARC) to establish a maximum allowable accident-induced Steam Generator tube leakage (AIL), against which cycle leakage projections are compared.
- The dose consequence analyses associated with the BVPS-1 MSLB, BVPS-2 MSLB, BVPS-1 SGTR, BVPS-2 SGTR, and BVPS-1 FHA take credit for a 30-minute control room purge after the post-accident environmental release has been terminated (or significantly reduced as in the case of the BVPS-1 SGTR). Subsequent to the air purge, the control room ventilation is returned to the Normal mode.
- No credit is taken for the filtration capability of the Supplemental Leak Collection and Release System (SLCRS).
- No credit is taken for automatic initiation of the Emergency (pressurization) mode of the Control Room Emergency Ventilation System (CREVS). Availability of CREVS, when credited, is based on manual initiation and assumed to be available 30 minutes after the accident.
- The habitability of the BVPS Emergency Response Facility (which houses the Technical Support Center (TSC)) following a LOCA is assessed without crediting the normal or emergency ventilation systems for dose reduction (via operation of the ventilation equipment in the ERF) or the ERF structure for shielding (from direct shine from the containment or the radioactive plume).

Appendix A provides a comparison of the critical input parameter values utilized in the current BVPS licensing basis dose consequence analyses versus that being proposed by this application.

7.1 Control Room Design / Operation / Transport Model

Except as noted below with respect to the flow rates associated with unfiltered air inleakage / intake into the control room during all modes of control room operation, the BVPS control room design and operation outlined in LAR Nos. 300 and 172 and LAR Nos. 302 and 173 remains unchanged by this application. Some of the critical aspects of the BVPS control room design are summarized below.

BVPS is served by a single control room that supports both units. The joint control room is serviced by two ventilation intakes, one assigned to BVPS-1 and the other to BVPS-2. These air intakes serve both units and are utilized during the normal ventilation, as well as the emergency ventilation mode of operation.

To provide operational margin, the dose consequence analyses summarized herein assume that during normal plant operation, the joint BVPS-1 & BVPS-2 unfiltered intake plus inleakage is a maximum of 1250 cfm (previously 500 cfm). This maximum normal operation unfiltered air inflow to the CR is an analytical upper bound value that is intended to include a) the CR normal operation intake flow rate (including test measurements uncertainties), and b) all unfiltered air inleakage, including a 10 cfm allowance for ingress/egress.

The containment high-high pressure signal (i.e., the Containment Isolation Phase B signal (CIB)) from either unit will initiate the BVPS-2 Control Room Emergency Ventilation System (CREVS). In the event one of the BVPS-2 trains is out of service, and the second train fails to start, operator action will be utilized to initiate the BVPS-1 CREVS.

The control room emergency filtered ventilation intake flow varies between 800 to 1000 cfm (previously assumed to vary between 600 to 1030 cfm), which includes allowance for measurement uncertainties. For reasons outlined below, the dose model uses the minimum intake flow rate of 800 cfm in the pressurized mode as it is more limiting. Although the filtered intake of radioisotopes is higher at the larger intake rate of 1000 cfm, it is small compared to the radioactivity entering the control room, in both cases, due to unfiltered air inleakage. Consequently, the depletion of airborne activity in the control room via the higher outleakage rate of 1000 cfm make the lower intake rate of 800 cfm more limiting from a dose consequence perspective. This argument holds true because the CEDE from inhalation is far more limiting than the DDE from immersion which is principally from noble gases.

The CR emergency ventilation intake filter has an efficiency of 99% for particulates, and 98% for elemental and organic iodine. Filtration of the Control Room ventilation recirculation flows during all modes of operation by particulate air filters (intended for dust removal) in the CRVS recirculation air-conditioning system is not credited.

For purposes of dose assessment, no credit is taken for automatic initiation of the CREVS following any design basis accident. For events that take credit for operation of the CREVS, the analyses assume manual initiation and that a pressurized control room is available at t = 30 minutes after the accident. For selected accidents, credit is taken for control room clean-up via a 30 minute control room purge at a minimum flow rate of 16200 cfm.

As discussed in Section 5.2, and in accordance with the current BVPS licensing basis, the atmospheric dispersion factors associated with control room air inleakage are considered the same as those utilized for the control room intake.

The unfiltered air inleakage assumed in the dose consequence analyses envelopes the results of recent tracer gas testing, includes 10 cfm for ingress/egress, and provides margin for potential deterioration between surveillance tests. The maximum unfiltered air inleakage into the CRE during the listed modes of operation are assumed to be as follows:

- CR Isolation mode – increased from a previous value of 300 cfm to 450 cfm (this represents an upper bound analytical value which includes test measurement uncertainties and a 10 cfm allowance for ingress/egress)

- CR Emergency mode – increased from a previous value of 30 cfm to 165 cfm (this represents an upper bound analytical value which includes test measurement uncertainties and a 10 cfm allowance for ingress/egress).

Since the BVPS common control room is contained in a single control room envelope, it is modeled as a single region. Isotopic concentrations in areas outside the control room envelope are assumed to be comparable to the isotopic concentrations at the control room intake locations. To support development of bounding control room doses, the control room ventilation model corresponds to an assumed "single intake" which utilizes the worst case atmospheric dispersion factor (χ/Q) from release points associated with accidents at either unit to the limiting control room intake.

Table 7.1-1 lists key assumptions and input parameters associated with BVPS control room design and applicable to the dose consequence analyses presented in this application.

7.2 Loss of Coolant Accident (LOCA)

The accidental rupture of a main coolant pipe is the event assumed to initiate a large break LOCA. Analyses of the response of the emergency core cooling system (ECCS) to ruptures of various sized reactor coolant lines are evaluated in the BVPS UFSARs. As demonstrated in these analyses, the ECCS, using emergency power, is designed to keep clad temperatures well below melting and to limit zirconium-water reactions to an insignificant level. However, as a result of the increase in clad temperature and the rapid depressurization of the core, some clad failure may occur in the hottest regions of the core. Following the clad failure, some of the core fission products would be released to the reactor coolant and subsequently to the inside of the containment building.

RG 1.183 identifies the large break LOCA as the design basis case of the spectrum of break sizes for evaluating performance of release mitigation systems including containment, and facility siting relative to radiological consequences.

In accordance with the current BVPS licensing basis, BVPS has identified four activity release paths following a LOCA: a) Containment Vacuum System Release, b) Containment Leakage, c) ESF System Leakage, and d) RWST back-leakage.

The methodology used to assess post-LOCA fission product transport and associated dose consequences at BVPS remains unchanged from that discussed in LAR Nos. 300 and 172, and as amended by LAR Nos. 334 and 205. The methodology presented in these licensing applications was approved by NRC SERs issued via References 5, 9 and 10.

Table 7.2-1 lists some of the key assumptions/parameters utilized in this application to develop the radiological consequences following a LOCA at either unit. Parameter values are bounding and encompass the event occurring at either unit.

7.2.1 Doses due to Submersion and Inhalation

Computer program PERC2 is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOCA. PERC2 utilizes an exact solution analytical computational process that addresses radionuclide progeny, time dependent releases, transport rates between regions and deposition of radionuclide concentrations in sumps, walls and filters.

7.2.1.1 Containment Vacuum System Release

In accordance with the current BVPS licensing basis, it is assumed that the containment vacuum system is operating at the initiation of the LOCA and that the release is terminated as part of containment isolation.

In accordance with RG 1.183, the entire RCS inventory, assumed to be at TS levels, is released to the containment at t = 0 hours. The analysis conservatively assumes that 100% of the volatiles are instantaneously and homogeneously mixed in containment atmosphere. As noted in Reference 8, containment pressurization due to the RCS mass and energy release (assumed to be at containment design pressure), combined with the relief line cross-sectional area, results in a 2200 scfm release of containment atmosphere (based on BVPS-1, equivalent release from BVPS-2 is 1600 scfm), to the environment over a period of 5 seconds (i.e., prior to containment isolation). Since the release is isolated within 5 seconds after the LOCA, i.e., before the onset of the gap phase release assumed to be at 30 seconds, no fuel damage releases are postulated.

Per RG 1.183, the chemical form of the iodine released from the RCS is assumed to be 97% elemental and 3% organic. As noted in Reference 8, the containment vacuum system line is routed to the Process Vent which is located on top of the BVPS-1 Cooling Tower. However, since the associated piping is non-seismic, it is conservatively assumed that the release occurs at the containment wall.

No credit is taken for processing this release via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e.; the Supplemental Leak Collection and Release System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for this release reflect the worst value between the containment wall release point and the SLCRS release point for 0 to 2 hr time period.

An assessment of the activity release via this pathway continues to demonstrate that its contribution to the site boundary and control room dose is negligible.

7.2.1.2 Containment Leakage

The inventory of fission products in the reactor core available for release via containment leakage following a LOCA is based on Table 4.1-1 which represents a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation and taking into consideration fuel enrichment and burnup.

Although the core inventory has not changed, some of the plant design input values that are used to develop fission product removal coefficients have been impacted as a result of the following:

a) The increase in the LOCA Mass & Energy Release Rates due to incorporation of Westinghouse NSAL-11-5 (Reference 13), and

b) Use of updated design input parameter values intended to conservatively bound BVPS-1 operation, BVPS-2 operation, or a combination of both, as appropriate.

Note: NSAL-11-5 is applicable to the LOCA M&E release calculations performed for Westinghouse designed PWRs utilizing methodology documented in WCAP-10325-P-A and WCAP-8264-P-A, Revision 1. It identifies six issues that can result in an increase in the LOCA M&E release to the containment. The increase in the LOCA M&E release associated with each of these issues affects the plant specific containment LOCA blowdown and post-blowdown transient conditions differently. As noted in NSAL-11-5, as a result of this methodology correction, the containment pressure, containment air temperature, and sump water temperature are typically slightly increased.

The key parameters affected by the above are spray flow rates, spray initiation and termination times, spray coverage, containment pressure, containment air temperature, and steam condensation rates in containment.

In accordance with the current BVPS licensing basis, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Containment sprays are utilized as one of the primary means of fission product cleanup following a LOCA.

BVPS design includes a containment quench spray system and a containment recirculation spray system at each of the units. Following post-LOCA containment pressurization, the quench spray system is automatically initiated by the CIB signal, and injects cooling water from the refueling water storage tank (RWST), into the containment, via the quench spray system spray headers. Based on an assumption of a LOOP coincident with the LOCA, the quench spray is assumed to be initiated at either unit, by approximately 80 seconds (maximum bounding value based on BVPS-2, *previously approximately 85 seconds*), and is available until depletion of the RWST inventory. The initiation of recirculation spray is based on RWST level; consequently, the recirculation spray initiation time is dependent on the accident scenario. For purposes of conservatism, the model is based on a scenario that maximizes the delay in start of the recirculation spray. Recirculation spray is assumed to start at 3855 seconds (bounding value based on BVPS-2, *previously 3870 seconds*). The recirculation spray system takes suction from the containment sump and provides recirculation spray inside containment via the recirculation spray headers. Credit for recirculation spray is taken up to 96 hours post-LOCA.

Mixing of the effectively sprayed volume of containment with the unsprayed volume of the containment also facilitates the cleanup. In order to quantify the effectiveness of the containment spray systems, the volume fraction of containment that is sprayed, and the mixing rate between the sprayed and unsprayed volumes are quantified.

7.2.1.2.1 Fission Product Removal

In the effectively sprayed region, fission product cleanup is actively accomplished by the quench and recirculation spray systems and passively by transport of particulates to the spray droplets and heat sink surfaces as a result of steam condensation on these surfaces. In the unsprayed region, only the passive gravitational settling phenomenon promotes particulate removal.

The analysis presented below is consistent with the current BVPS licensing basis and envelopes both BVPS-1 and BVPS-2.

Removal of Particulates by Sprays

In accordance with the current BVPS licensing basis, the particulate removal rate from the containment atmosphere by the quench and recirculation spray systems are calculated with the WECTEC proprietary Computer Program SWNAUA (Reference 30). The SWNAUA code is discussed in detail in LAR Nos. 300 and 172 (Reference 4), and its results were approved for use at BVPS via the NRC SER for Amendment Nos. 257 and 139.

As noted in Reference 4, the SWNAUA Program is a derivative of the NAUA/Mod4 Computer Program (Reference 31). The NAUA/Mod4 code does not include a model for aerosol removal by sprays. The aerosol removal model for sprays was developed and incorporated into the SWNAUA code by WECTEC as a conservative model suitable for design basis accident (DBA) calculations. As discussed in Reference 4, the model correlations that were implemented into SWNAUA tend to underestimate the spray removal coefficient. For the effectively sprayed region of the containment, WECTEC employs only the conservatively developed spray removal model and conservative condensation rates for the diffusiophoresis calculation when performing DBA calculations. While agglomeration is considered in the calculation, its impact on the resulting particulate removal rates is negligible. In summary, the aerosol removal rates calculated by SWNAUA are conservative lower bound estimates.

As noted in Reference 4, there are several aerosol mechanics phenomena that promote the depletion of aerosols from the containment atmosphere. These include both the natural phenomena of agglomeration, gravitational settling, diffusional plate-out, and diffusiophoresis; and the removal by fluid mechanical interaction with the falling droplets that enter the containment atmosphere through the spray system nozzles. For Beaver Valley, the particulate removal calculation for the effectively sprayed region only takes credit for the removal effectiveness of diffusiophoresis and sprays. Agglomeration of the aerosol is considered. If gravitational settling and diffusional plate-out were considered, the spray removal coefficients would have been slightly reduced but the total removal effectiveness by all removal mechanisms would have increased. However, gravitational settling of aerosols is credited in the unsprayed region.

The spray model utilized for the LOCA analysis, including a description of the aerosol and the aerosol injection rate, is provided in Section 5.3.6.3.1.2 of LAR Nos. 300 and 172 and was approved in the NRC SER for Amendment Nos. 257 and 139. The methodology described above was re-affirmed and approved by LAR Nos. 334 and 205 and NRC SERs for Amendment No. 280 and Amendment No. 164, respectively, and remains unchanged with this application.

The plant parameters that are utilized herein to develop the aerosol removal coefficients are bounding for both Beaver Valley Units and are listed below.

Plant Parameters for Fission Product Cleanup Calculations		
Parameter	Value	Previous Value in Reference 8
Sprayed Containment Volume	3.062×10^{10} cm^3	*3.1973×10^{10} cm^3*

Plant Parameters for Fission Product Cleanup Calculations		
Parameter	Value	Previous Value in Reference 8
Fall Height	2,403 cm	2,403 cm
Spray Flow Rate	1,821 gpm (120 to 2080 sec) 2,910 gpm (2080 to 3855 sec) 7,871 gpm (3855 to 4227 sec) 6,113 gpm (4227 to 10158 sec) 4,740 gpm (10158 to 11545 sec) 3,178 gpm (11545 to 345600 sec)	1,821 gpm (120 to 2080 sec) 2,956 gpm (2080 to 3870 sec) See below 6,108 gpm (3870 to 10938 sec) See below 3,267 gpm (10938 to 346000 sec)
Spray droplet radius	500 microns	500 microns

The containment pressure, air temperature, steam condensing rate and relative humidity transients utilized for the development of the aerosol removal coefficients are obtained from the MAAP-DBA model of the traditional design basis LOCA and are presented in Table 7.2-6. This data was recently developed and takes into consideration the impact of the updated post-LOCA M&E releases resulting from incorporation of the findings of Westinghouse NSAL-11-5 on BVPS-2. The impact of NSAL-11-5 on BVPS-1 M&E releases on the aerosol removal coefficients was evaluated subsequent to the above effort. It was determined that the aerosol removal coefficients, estimated using the data presented above, remain bounding.

Since higher containment pressure and temperature, lower steam condensing rates and lower spray flow rates result in lower aerosol removal rates, the bounding outputs of two design basis LOCA cases per unit are conservatively utilized; i.e., a) Pump Suction Double Ended Rupture (PSDER) with failure of one emergency diesel generator, and b) PSDER with CIB failure.

Removal of Particulates by Diffusiophoresis

In accordance with the current BVPS licensing basis (Reference 8), during diffusiophoresis, particulate matter is entrained in the steam as it flows to the condensation surfaces. In this calculation, steam is assumed to condense on the spray droplets, on particulate matter, and on heat sinks. The diffusiophoresis model in the SWNAUA computer code is the same as that in the NAUA/Mod4 computer code.

The bounding pressure, temperature and steam condensation rates used to develop the particulate removal coefficients due to diffusiophoresis are presented in Table 7.2-6.

The coefficient for removal of particulates from the effectively sprayed and unsprayed regions of the containment are plotted versus time in Figure 7.2-1 and Figure 7.2-2, respectively, and presented in Table 7.2-2. For the effectively sprayed region, the aerosol removal is due to sprays and diffusiophoresis. The particulate removal coefficient in the unsprayed region is due to gravitational settling only.

Removal of Elemental Iodine by Sprays and Plate-out

Consistent with the current BVPS licensing basis, the updated analyses demonstrate that during spray operation the elemental iodine removal rates due to sprays, developed using NUREG-0800, Section 6.5.2 (Reference 32) methodology, always exceed 20 hr^{-1}, which is the maximum value permitted by the referenced SRP.

In addition, and in accordance with the current BVPS licensing basis (Reference 8), during spray operation, the elemental iodine is conservatively assumed to be removed by sprays at the same rate as the aerosol particles when the aerosol removal rate is lower than 20 hr^{-1}, and at 20 hr^{-1} when the aerosol removal rate is calculated to be higher than the NRC limit.

A plate-out removal coefficient for elemental iodine has been calculated with the model provided in NUREG-0800, Section 6.5.2. Prior to spray actuation, the total containment surface area is available for wall deposition due to condensation on heat surfaces. Subsequently, due to heat up, certain portions of the heat sinks become non-condensing and can no longer be considered wetted surfaces. For purposes of analysis simplification, and consistent with the current BVPS licensing basis, after initiation of containment sprays, the wetted surface area within the sprayed volume is conservatively assumed to be limited to the carbon steel lined containment shell surface area multiplied by the spray coverage fraction of containment volume.

In summary, in the effectively sprayed region, plate-out coefficients of 4.1075 hr^{-1} and 0.5358 hr^{-1} are calculated for the period before initiation of sprays and after initiation of sprays, respectively. This allows a maximum elemental iodine removal rate in the effectively sprayed region, during the spray period, of 20.5358 hr^{-1}. No credit is taken for elemental iodine removal in the unsprayed region. Table 7.2-2 presents the elemental iodine removal coefficients as a function of time.

<u>Effectively Sprayed Containment Volume Fraction</u>

In accordance with the current BVPS licensing basis (Reference 8), the sprayed volume fraction of the containment is determined by superimposing the spray patterns onto the containment arrangement drawings. The sprayed volume is the volume of the unblocked spray patterns. The spray patterns are based on the nozzle manufacturer's laboratory tests at atmospheric conditions. The patterns have been compressed to account for the higher density atmosphere that exists during the DBA. The effectively sprayed volume is calculated by combining the highly mixed unsprayed regions with the sprayed.

For purposes of spray coverage, the quench spray is credited prior to initiation of recirculation spray. Subsequent to initiation of recirculation spray, no credit is taken for spray coverage by quench spray operation.

The effective spray coverage fraction (i.e., the effective quench or recirculation spray) is conservatively assessed to be 60 percent (previously 63 percent) of the containment free volume at either unit for the duration of time that credit is taken for sprays.

The concentration of fission products is expected to be uniform in the containment volume above the operating floor since this volume is open with very few obstructions to mixing. Consistent with the current BVPS licensing basis, the sprayed volume is taken as the free volume above the operating floor plus the volume below the operating floor that is covered by sprays.

<u>Containment Mixing</u>

Mixing in the containment following a postulated design basis LOCA results from four mechanisms: 1) momentum transfer from the fluid jet exiting the break; 2) momentum transfer from the spray droplets to the surrounding gas; 3) forced and natural convection flows within the containment atmosphere; and 4) molecular diffusion. All of these mechanisms will work together to enhance mixing within the containment to provide a homogeneous gas mixture and prevent local accumulation of fission products.

In accordance with the current BVPS licensing basis (Reference 8), the mixing rate between the effectively sprayed volume and unsprayed volume of the containment is assumed to be 2 hr^{-1}, the rate permitted by NUREG-0800, Section 6.5.2 (Reference 32).

7.2.1.2.2 Radiological Transport Model

As indicated previously, and in accordance with the current BVPS licensing basis (Reference 8), the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. In accordance with RG 1.183, two release phases are considered:

(a) gap release phase, which begins 30 seconds after the LOCA and continues for 0.5 hour
(b) early in-vessel release phase, which begins 0.5 hour after the accident and continues for 1.3 hours.

In accordance with the current BVPS licensing basis (Reference 8), the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel release phases are as follows:

Group	Gap Release Phase	Early In-Vessel Release Phase
Noble gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Group	-	0.05
Ba, Sr	-	0.02
Noble Metals	-	0.0025
Cerium Group	-	0.0005
Lanthanides	-	0.0002

The elements in each radionuclide group released to the containment following a LOCA are assumed to be as follows (note that, in accordance with the current BVPS licensing basis, the groupings were expanded from that in RG 1.183 to address isotopes in the core with similar characteristics):

Noble gases: Xe, Kr, Rn, H

Halogens:	I, Br
Alkali Metals:	Cs Rb
Tellurium Grp:	Te, Sb, Se, Sn, In, Ge, Ga, Cd, As, Ag
Ba,Sr:	Ba, Sr, Ra
Noble Metals:	Ru, Rh, Pd, Mo, Tc, Co
Cerium Grp:	Ce, Pu, Np, Th, U, Pa, Cf, Ac
Lanthanides:	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, Gd, Ho, Tb, Dy

In accordance with Regulatory Guide 1.183, since the BVPS long term sump pH is controlled to values of 7 and greater, the chemical form of the radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, all fission products released are assumed to be in particulate form.

The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LOCA.

In the effectively sprayed region the activity transport model takes credit for aerosol removal due to steam condensation, and via containment recirculation and quench sprays based on spray flowrates associated with minimum ESF. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by concentration dependent aerosol removal lambdas, and isotopic in-growth due to decay.

In the unsprayed region, the aerosol removal lambdas reflect gravitational settling. No credit is taken for elemental iodine removal in the unsprayed region.

Since the spray removal coefficients are based on calculated time dependent airborne aerosol mass, there is no restriction on the Decontamination Factor for particulate iodine. The maximum DF for elemental iodine is based on NUREG-0800, Section 6.5.2, and is limited to a DF of 200. For BVPS, this DF value is reached for elemental iodine at approximately 6.3 hours (previously approximately 6.4 hours) after the accident.

Mixing between the effectively sprayed and unsprayed regions of the containment is assumed for the duration of the accident. Though higher mixing rates are expected, the dose analysis conservatively assumes a mixing rate of 2 unsprayed volumes per hour in accordance with the default value noted in NUREG-0800, Section 6.5.2.

Current BVPS design includes chemical addition to the post-LOCA sump water via use of NaTB baskets to ensure a long term sump pH greater than or equal to 7.0 (References 26, 27, 28 and 29). Long term production of acids (HCl and HNO_3) by irradiation is included in determining the long term sump pH. Long term retention of iodine in sump liquids is strongly dependent on the sump pH. In accordance with the current BVPS licensing basis (References 8, 26 and 27), the dose analysis does not address iodine re-evolution as a sump pH of $\geq$ 7 is achieved well within

16 hours after the LOCA and is maintained for the duration of the accident. The definition of long term as it relates to sump pH and iodine re-evolution post-LOCA is based on NUREG/CR-5732 (Reference 33).

Radioactivity is assumed to leak from both the sprayed and unsprayed region to the environment at the containment Technical Specification leak rate for the first day, and half that leakage rate for the remaining duration of the accident (i.e., 29 days). No credit is taken for processing the containment leakage via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e.; the Supplemental Leak Collection and Release System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflect the worst value between the containment wall release point and the SLCRS release point at the top of containment for each time period.

7.2.1.3 ESF and RWST Back-Leakage

In accordance with RG 1.183, with the exception of noble gases, all the fission products released from the core in the gap release and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the fuel. The minimum sump volume increases to a steady state minimum value of 321,600 gallons (limiting parameter values based on BVPS-1, previously 321,155 gallons), two hours after the LOCA. In accordance with the current BVPS licensing basis (Reference 8), three sump volume values are utilized in the transport model. Up to the first half hour after the LOCA, the sump volume is about 40% of the final value. For the next one and a half hours the sump volume is about 56% of the final value. For the remainder of the accident the steady state minimum sump volume is utilized. In accordance with Regulatory Guide 1.183, with the exception of halogens, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. The subsequent environmental radioactivity release is discussed below:

- ESF leakage: In accordance with the current BVPS licensing basis (Reference 8), components carrying sump fluids and located outside containment are postulated to leak at twice the surveillance limit of 5700 cc/hr (BVPS-1 value, BVPS-2 value is 2134 cc/hr) into the Auxiliary Building. ESF leakage is expected starting at initiation of the recirculation spray which at BVPS is conservatively assumed to start at 1200 seconds (bounding for both units). Note that due to the long term nature of this release, minor variations in the start time of this release will not significantly impact the resultant doses. The peak sump water temperature after 20 minutes is 250ºF. As noted in Regulatory Guide 1.183, the fraction of total iodine in the liquid that becomes airborne should be assumed to be equal to the fraction of the leakage that flashes to vapor. The flash fraction (using Regulatory Guide 1.183 methodology) associated with this temperature is calculated to be less than 10%. Consequently, in accordance with Regulatory Guide 1.183, 10% of the halogens associated with this leakage is assumed to become airborne and are exhausted (without mixing or holdup) to the environment via the SLCRS vent located on top of Containment. In accordance with Regulatory Guide 1.183, the chemical form of the iodine released from the sump water is 97% elemental and 3% organic. No credit is taken for the SLCRS filters.

- RWST Back-leakage: In accordance with the current BVPS licensing basis (Reference 8), sump water back-leakage into the RWST (located in the Yard) is postulated to occur at twice the surveillance limit of 1 gpm and is released to the environment via the RWST vent. A significant portion of the iodine associated with the RWST back-leakage is retained within the tank due to the equilibrium iodine distribution balance between the RWST gas and liquid phases (i.e., time dependent iodine partition coefficient). The methodology used to determine the RWST release fractions was presented in LAR Nos. 334 and 205 and approved by NRC SERs for Amendment No. 280 and Amendment No. 164, respectively.

For the limiting BVPS-1 case, sump water begins to leak into the RWST at 1768 seconds (previously 1782 seconds) after the LOCA. At 3039 seconds (previously 3055 seconds), the iodine begins to flow out of the RWST and disperse to the environment. Environmental airborne iodine activity resulting from RWST leakage is assumed to be 97% elemental and 3% organic.

In accordance with the current BVPS licensing basis, in the dose model, this phenomenon is modeled using a series of effective environmental release rate lambdas from the RWST vent.

In accordance with the current BVPS licensing basis (Reference 8), for purposes of limiting the amount of re-analyses, the RWST iodine release fractions used in the dose consequence analyses supporting LAR Nos. 334 and 205 were conservatively based on twice the release fractions versus time provided in Figure 5.3.6-3 of LAR Nos. 300 and 172, with the following adjustment. The release fraction (times two) applicable at t = 5178 seconds (see Figure E-3 of Reference 8) is also applied to the time period between t = 3055 seconds and t = 5178 seconds. As noted in Reference 8, the above approach was determined to be very conservative based on confirmatory analyses performed to estimate the RWST iodine release fractions.

The RWST iodine release fractions used in the dose consequence analyses supporting Reference 8 are considered the BVPS design RWST iodine release fractions.

As demonstrated in Figure 7.2-3 and Table 7.2-3, the design RWST iodine release fractions presented in Reference 8 bound the values calculated by the confirmatory analyses performed to estimate the RWST iodine release fractions using the updated transient data supporting this application. The updated RWST iodine release fractions presented in Figure 7.2-3 and Table 7.2-3 take into consideration the impact of the use of NaTB as the buffering agent (References 26, 27, 28 and 29), and the updated post-LOCA M&E releases resulting from incorporation of the findings of Westinghouse NSAL-11-5 on the BVPS-2 containment transient. An assessment of the impact of NSAL-11-5 on BVPS-1 LOCA M&E releases on the updated RWST iodine release fractions determined that the effect was small.

In summary, the design RWST iodine release fractions used to establish dose consequences in Reference 8 remain bounding for this application.

7.2.2 Offsite Dose Assessment

The LOCA dose consequence analysis reflects bounding parameter values to encompass an event at either unit. The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

The worst 2 hour period dose at the EAB, and the dose at the LPZ for the duration of the release, is calculated based on postulated airborne radioactivity releases. This represents the post-accident dose to the public due to inhalation and submersion for each of these events. In accordance with RG 1.183, offsite breathing rates used are as follows: 0 to 8 hr (3.5E-04 m^3/sec), 8 to 24 hr (1.8E-04 m^3/sec), 1 to 30 days (2.3E-04 m^3/sec). Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible.

To find the “worst case 2 hour release window” for the EAB dose, the integrated dose versus time for each of the four pathways (i.e., containment pressure relief release path, containment leakage, ESF leakage, RWST back-leakage) is developed at increments of 0.1 hr. The 0 to 2 hr EAB Atmospheric Dispersion Factor is utilized for all cases. The analysis demonstrates that the maximum dose occurs with a two-hour release period that ends at approximately 2.5 hrs.

The bounding EAB and LPZ dose following a LOCA at either unit is presented in Section 8.

7.2.3 Control Room Occupancy Dose

7.2.3.1 Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical LOCA-specific assumptions associated with control room response and activity transport.

<u>Timing for Initiation of CREVS (if applicable)</u>:

- In accordance with the current BVPS licensing basis, due to the rapid pressure transient expected following a LOCA, the Containment Isolation Phase B (CIB) signal which initiates the CR isolation and emergency ventilation following a LOCA is assumed to actuate at t = 0 hours.
- Taking into account a LOOP, the maximum estimated delay in attaining control room isolation after receipt of a CIB signal to switch control room ventilation from normal mode to emergency mode is 77 seconds which accounts for delays due to emergency diesel generator start and load sequencing (including damper movement/re-alignment).
- No credit is taken for automatic initiation of the BVPS-2 CREVS; rather, it is assumed that operator action will be necessary to initiate the CREVS and a pressurized control room will be available within t = 30 minutes.

<u>Note</u>: In accordance with the current BVPS licensing basis (Reference 8), the analysis assumes a LOOP at t = 0 hours. The impact of a LOOP at a more unfavorable time following the accident, such as during the fuel release phase, is not addressed per NRC Information Notice 93-17 (Reference 34). The need to evaluate a design basis event assuming a simultaneous or subsequent LOOP is based on the cause/effect relationship between the two events (an example illustrated in IN 93-17 is that a LOCA results in a turbine trip and a loss of power generation to the grid, thus causing grid instability and a LOOP a few seconds later, i.e., a reactor trip could result in a LOOP). IN 93-17 concludes that plant design should reflect all credible sequences of the LOCA/LOOP, but states that a sequence of a LOCA and an unrelated LOOP (which would be the case if a LOOP was assumed to occur 1 to 2 hours after the event) is of very low probability and

is not a concern.

The 0 to 30 day dose to an operator in the control room due to airborne radioactivity releases is developed and represents the post-accident dose to the operator due to inhalation and submersion. The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The direct shine dose due to contained sources/external cloud is included in the CR doses reported for the LOCA.

Control Room Atmospheric Dispersion Factors:

The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a LOCA at either unit are provided in Table 7.2-5. Table 7.2-5 lists the various release points-receptors applicable to the LOCA and identifies the bounding χ/Q values applicable to a LOCA at either unit.

7.2.3.2 Direct Shine Dose to the Control Room from External and Contained Sources

The direct shine dose to an operator in the control room due to contained or external sources resulting from a postulated LOCA is calculated using WECTEC point kernel shielding computer program SW-QADCGGP. The post-LOCA gamma energy release rates (MeV/sec) and integrated gamma energy release (MeV-hr/sec) in the various external sources are developed with computer program PERC2.

The LOCA sources that could potentially impact the CR operator dose due to direct shine are identified below.

1. Direct shine from the containment structure (skyshine dose is insignificant due to associated soft gammas, and the 2 ft concrete roof/walls of the control room).
2. Direct shine from the airborne source located in the cable spreading room below the BVPS-2 portion of the combined control room through floor penetrations.
3. Direct shine from the airborne source located in the cable tray mezzanine area below the BVPS-1 portion of the combined control room through floor penetrations.
4. Direct shine through the CR walls and floor from the CREVS intake filters located in the adjacent room (BVPS-2 filters) and below (BVPS-1 filters), respectively.
5. Direct shine from the sump fluid that is postulated to collect in the RWST.
6. Direct shine from the contaminated cloud outside the control room pressure boundary resulting from containment leakage, ESF system leakage and RWST back-leakage.

The bounding control room operator dose following a LOCA at either unit is presented in Section 8.

7.2.4 Emergency Response Facility/Technical Support Center Dose

In accordance with the current BVPS licensing basis and except as noted below, habitability of the BVPS ERF (which houses the TSC) is assessed using methodology and accident release parameters similar to those utilized for the control room. From a radiological perspective, the assessment demonstrates compliance with paragraph IV.E.8 of Appendix E to 10 CFR Part 50 (Reference 38); regulatory guidance provided in Section 8.2.1, item f, of NUREG-0737 Supplement No. 1 (Reference 14); and the current BVPS licensing basis.

Doses due to Submersion and Inhalation

As discussed in Reference 8, BVPS is served by a single ERF/TSC that supports both units. During normal plant operation, the ERF ventilation intake flow of 3800 cfm (+/-10% for uncertainty) is processed through a HEPA filter. Unfiltered air inleakage during normal operation is estimated at 2090 cfm.

Following a LOCA, the ERF filtered air intake is manually isolated and switched to an emergency mode filtered air recirculation between t = 30 minutes to t = 60 minutes. The ERF emergency mode filtered air recirculation flowrate could be operating at its maximum flowrate of 7200 cfm (+/-10%); previously assumed to be operating at 3800 cfm (+/-10%). The ERF emergency mode processes the ventilation flow through charcoal filters and HEPA filters. The unfiltered air inleakage into the ERF during the emergency mode is estimated to be 910 cfm, which includes 10 cfm for ingress and egress.

However, for the purposes of demonstrating habitability following a LOCA, and in accordance with the current BVPS licensing basis model, no credit is taken for the ERF structure or ventilation systems when determining the inhalation or submersion dose. Because the ERF is located a sufficient distance away from the BVPS-1 and BVPS-2 containment buildings, the atmospheric dispersion characteristics of potential activity releases following a LOCA are very favorable, and therefore, no ventilation design features are required to ensure habitability. The habitability analysis of the ERF following a LOCA is performed by assuming that there is no ERF structure (i.e., the ERF is modeled as a point in the environment). Breathing rates and occupancy factors utilized are similar to those used for the control room. Table 7.2-4 lists key parameters associated with the ERF design. Table 7.2-5 lists the various release points applicable to the LOCA and identifies the bounding χ/Q values applicable to the ERF following a LOCA at either unit.

ERF Dose due to Direct Shine from External Sources

The dose contribution in the ERF due to direct shine from the external sources include shine from a) the Containment Structure (including skyshine), b) the ERF normal operation ventilation intake filter and c) the ERF emergency ventilation recirculation filters. Though not credited in the airborne dose assessment, the direct shine from the filters is included to address the potential of their usage post-accident. A bounding approach is utilized in the updated analysis presented herein with respect to estimating the direct shine dose from the ERF/TSC intake and recirculation ventilation filters; specifically, the filters are assumed to be 100% or 0% efficient, as deemed conservative, when addressing the direct shine dose. For example, to maximize the intake filter shine when calculating the direct shine dose, a 100% efficiency is assumed. However, to maximize the direct shine dose from the recirculation filter, it is assumed that the intake filters have 0% efficiency.

The external cloud shine dose is included in the submersion dose described earlier. The dose from radiation sources inside the RWST due to RWST back-leakage is negligible because the elevation of the top of the source (i.e., liquid level in the tank at t = 30 days) is below the elevation of state highway Route 168.

30 day Dose in the ERF.

The maximum estimated 30 day dose to the operator in the ERF following a LOCA at either unit is reported in Section 8.

7.3 Control Rod Ejection Accident (CREA)

This event consists of an uncontrolled withdrawal of a control rod from the reactor core. The CREA results in reactivity insertion that leads to a core power level increase, fuel damage, and a subsequent reactor trip. Following reactor trip, and based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs). Since the RHR pumps are located inside containment and are not qualified for Containment Isolation Phase B conditions which are expected to occur following a CREA, no credit is taken for initiation of shutdown cooling and environmental releases via the MSSVs/ADVs are assumed to occur for 30 days.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a CREA.

In accordance with the current BVPS licensing basis established via References 4 and 5, regulatory guidance provided in Appendix H of RG 1.183 is used to develop the dose consequences following the CREA. Table 7.3-1 lists some of the key assumptions/parameters utilized in this application to develop the radiological consequences following the CREA. Parameter values are bounding and encompass the event occurring at either unit.

Per RG 1.183, the CREA evaluation needs to address two independent release paths to the environment: first, via containment leakage of the fission products released due to the event from the primary system to containment, assuming that the containment pathway was the only one available; and second, via releases from the secondary system, outside containment, following primary-to-secondary leakage in the steam generators, assuming that the latter pathway was the only one available.

The actual doses resulting from a postulated CREA would be a composite of doses resulting from portions of the release going out via the containment building and portions via the secondary system. If regulatory compliance to dose limits can be demonstrated for each of the scenarios, the dose consequence of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios.

The BVPS CREA analysis evaluates the following two scenarios.

Scenario 1: The failed/melted fuel resulting from a postulated CREA is released into the RCS, which is released in its entirety into the containment via the ruptured control rod drive mechanism housing, mixed in the free volume of the containment, and then released at the containment Technical Specification leak rate for the first 24 hours and at half that rate for the remaining 29 days. Environmental releases occur via penetrations/cracks in the containment's steel liner.

Scenario 2: The failed/melted fuel resulting from a postulated CREA is released into the RCS, which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Environmental releases occur from the steam generators via the MSSVs/ADVs until t = 30 days.

In accordance with the current BVPS licensing basis, the CREA is postulated to result in < 10% fuel failure (resulting in the release of the associated gap activity), and < 0.25% melted fuel. To account for differences in power level across the core, a design radial peaking factor of 1.70 (previously 1.75) is applied in determining the isotope inventory of the damaged rods.

In accordance with Regulatory Guide 1.183, the gap activity is assumed to be composed of 10% of the noble gases and 10% of the iodines associated with the percentage of fuel that has failed. The above release fractions for the CREA are provided in Note 11 to Table 3 of Regulatory Guide 1.183. For this application, halogens were substituted for iodines.

In addition, and per RG 1.183, depending on the release pathway, the composition of the melted fuel is varied.

- For the containment leakage pathway, the melted fuel activity released is assumed to be composed of 100% of the core noble gas and 25% of the core halogens associated with the percentage of fuel that has melted.
- For the Secondary System Release pathway the melted fuel activity released is composed of 100% of the core noble gas and 50% of the core halogens associated with the percentage of fuel that has melted.

Guidance provided in RG 1.183 indicates that the chemical composition of the iodine in the gap/melted fuel should be assumed to be 95% CsI, 4.85% elemental and 0.15% organic. However, in accordance with the current BVPS licensing basis, because the sump pH is not credited following a CREA, it is conservatively assumed that the iodine released via the containment leakage pathway has the same composition as the iodine released via the secondary system release pathway; i.e., it is assumed that for both scenarios, 97% of all halogens available for release to the environment are elemental, while the remaining 3% is organic.

Scenario 1: Transport from the Containment

The failed/melted fuel activity released due to a CREA into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the Technical Specification leak rate of 0.10% of containment air weight per day for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen (or noble gas) concentrations airborne in the containment. No credit is taken for processing the containment leakage via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e.; the Supplemental Leak Collection and Release System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factor utilized for the containment release path reflects the worst value between the containment wall release point and the SLCRS release point for each time period.

Scenario 2: Transport from the Secondary System

The failed/melted fuel activity released due to a CREA into the RCS is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary-to-secondary steam generator (SG) tube leakage assumed to be at the maximum Technical Specification value of 150 gpd at STP from each steam generator (450 gpd total). The primary-to-secondary leakage terminates at 2500 seconds after the event when primary pressure is below secondary pressure.

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant; therefore, and per RG 1.183, the fuel/gap iodines are assumed to have a partition coefficient of 100 in the SG and released to the environment in proportion to the steaming rate and the partition coefficient. In accordance with RG 1.183, the iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The gap noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a CREA is discharged to the environment from the steam generators via the MSSVs/ADVs. Since the RHR pumps are located inside containment and are not qualified for Containment Isolation Phase B conditions which are expected to occur following a CREA, no credit is taken for initiation of shutdown cooling and environmental releases via the MSSVs/ADVs are assumed to occur for 30 days. It is conservatively assumed that the rate of steam release after t = 8 hours is the same as that applicable between t = 2 hours and t = 8 hours.

The activity associated with the release of secondary steam/liquid and primary-to-secondary leakage of normal operation reactor coolant (both at TS activity concentration levels) via the MSSVs/ADVs is insignificant compared to the failed fuel release and is therefore not included in this assessment.

Offsite Dose Assessment

The CREA dose consequence analysis reflects bounding parameter values to encompass an event at either unit. The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

AST methodology requires that the worst case dose to an individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the accident be reported as the EAB dose:

<u>Containment Release</u>: Since the noble gas and iodine activity release rates (i.e., Curies/sec) are at their highest levels at the onset of the event, the worst 2 hour window EAB dose will occur at t = 0 to 2 hours.

<u>Secondary Side Release</u>: The worst 2 hour EAB dose will occur during the initial 2 hour period because the primary-to-secondary leakage stops at 2500 seconds (0.6944 hours), and the steam release rate is also the highest during this period.

The bounding EAB and LPZ dose following a LOCA at either unit is presented in Section 8.

Accident Specific Control Room Model Assumptions

Actuation of the CIB signal is not credited in the analysis, and the control room emergency ventilation system is manually initiated at t = 30 minutes after the CREA. The remaining Control Room parameters utilized in this model are discussed in Section 7.1.

The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a CREA at either unit are provided in Table 7.3-2.

The EAB, LPZ and Control room doses following a CREA are presented in Section 8.

7.4 Main Steam Line Break Outside Containment (MSLB)

This event consists of a double-ended break of one main steam line. The analysis focuses on a MSLB *outside* the containment since a MSLB inside containment will clearly result in a lesser dose to a control room operator or to the public at offsite locations due to holdup of activity in the containment (OPEN ITEM 2). Following a MSLB outside containment, the faulted steam generator (SG) rapidly depressurizes and releases its initial secondary side liquid contents to the environment via the break. Based on an assumption of a simultaneous Loss of Offsite Power, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs/ADVs of the intact steam generators are used to cool down the reactor until initiation of shutdown cooling via the RHR system.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a MSLB at BVPS-1 and BVPS-2.

In accordance with the current BVPS licensing basis established via References 6 and 7, regulatory guidance provided in Appendix E of RG 1.183 is used to develop the dose consequences following the MSLB.

No fuel melting or clad breach is postulated for the BVPS MSLB event. Thus, and in accordance with RG 1.183, the activity released is based on the maximum coolant activity allowed by the BVPS Technical Specifications. In addition, and per RG 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike which reflects the maximum allowable iodine spike activity level per the BVPS Technical Specifications and b) an accident-initiated iodine spike (also called a concurrent iodine spike) which results in an increase in the iodine appearance rate from the fuel to the RCS by 500 times.

In addition, and in accordance with the current BVPS licensing basis, the BVPS-2 assessment supports the implementation of Alternate Repair Criteria (ARC) as defined in NRC Generic Letter (GL) 95-05 (Reference 35). In accordance with GL 95-05, the BVPS-2 MSLB dose assessment determines the maximum accident induced leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the EAB, LPZ and the control room.

Table 7.4-1a and Table 7.4-1b list some of the key assumptions/parameters utilized in this application to develop the radiological consequences following the MSLB at BVPS-1 and BVPS-2, respectively.

BVPS-1

In accordance with RG 1.183, the activity in the reactor coolant is increased due to two potential iodine spike scenarios: a pre-accident iodine spike and a concurrent iodine spike as described below.

a. Pre-accident iodine spike - the initial reactor coolant iodine activity is based on a maximum allowable pre-accident iodine spike activity level per the current BVPS Technical Specifications of 60 times the equilibrium Technical Specification iodine activity concentration of 0.35 µCi/gm DE I-131 or 21 µCi/gm of DE I-131 (transient Technical Specification limit for full power operation). The initial reactor coolant noble gas activity is consistent with the design basis relative mix and activity levels associated with the current Technical Specification iodine concentrations in the coolant for BVPS-1.

b. Concurrent iodine spike - the initial reactor coolant iodine activity is assumed to be at the BVPS-1 equilibrium Technical Specification iodine activity concentration of 0.35 µCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the reactor coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 0.35 µCi/gm DE I-131 coolant concentration allowed by the current Technical Specifications. The duration of the assumed spike is 4 hours (*see note below*). The initial reactor coolant noble gas activity is assumed to be at TS levels.

Note: In accordance with RG 1.183, the duration of the concurrent iodine spike is based on the time it takes to deplete all of the activity in the fuel gap. The current BVPS licensing basis 4-hour duration for the BVPS post-accident concurrent iodine spike is based on RG 1.183 iodine gap fractions, and a gap depletion rate based on the SGTR (i.e., 335 X the equilibrium release rate) which is smaller than the MSLB (i.e., 500 X the equilibrium release rate), and thus more limiting. In the conservative assessment documenting the basis of the 4 hour duration that was presented to the NRC as part of Reference 6, and approved by NRC via Reference 7, *no credit* was taken for the decay of the iodines in the fuel gap during the spiking period.

As part of this application, an assessment was performed of the impact of the use of fractions of fission product inventory in the gap based on DG-1199 (specifically, the effect of the increase in the gap fraction of I-132 from 0.05 to 0.23) on the spike duration. It was determined that if credit is taken for the decay of the iodines in the fuel gap during the spiking period, the iodine spike duration remains bounded by the licensing basis duration of 4 hours.

The resultant reactor coolant activity leaks into the faulted and intact SGs via SG tube leakage and is released to the environment from the break point and from the MSSVs/ADVs, respectively.

The secondary coolant iodine activity, just prior to the accident, is assumed to be at the BVPS-1 Technical Specification limit of 0.10 µCi/gm DE I-131.

BVPS Technical Specifications limit primary-to-secondary SG tube leakage to 150 gpd per steam generator for a total of 450 gpd in all 3 SGs.

Faulted Steam Generator

The release from the faulted steam generator occurs via the postulated break point of the main steam line. The faulted steam generator is conservatively assumed to dry-out instantaneously following the MSLB, releasing all of the iodines in the secondary coolant that were initially contained in the steam generator. The secondary steam initially contained in the faulted steam generator is also released; however, this contribution is not included in this analysis since the associated radioactivity is insignificant compared to the other contributions. The primary-to-secondary leakage reflects 150 gpd at STP per the BVPS Technical Specifications. In accordance with RG 1.183, all iodine and noble gas activities in the referenced tube leakage are released directly to the environment without holdup or decontamination. The primary-to-secondary leakage and associated environmental release continues until the temperature of the RCS reaches 212ºF at t = 19 hours.

Intact Steam Generator

Releases from the two remaining intact steam generators (used to cool the reactor and the primary system) occur via the plant MSSVs/ADVs.

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant; therefore, and per RG 1.183, the iodines are assumed to have a partition coefficient of 100 in the SG and released to the environment in proportion to the steaming rate and the partition coefficient. In accordance with RG 1.183, the iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The noble gases are released freely to the environment without retention in the SG.

The steam releases from the MSSVs/ADVs terminate at 8 hours after the event when shutdown cooling is initiated via the RHR system. The total leakage from the reactor coolant to the intact SGs is determined by the total allowable leakage from all 3 SGs (450 gpd) per the BVPS Technical Specifications, minus that which leaks to the faulted SG (150 gpd); i.e., 450 gpd – 150 gpd = 300 gpd.

BVPS-2

Except as noted below, the BVPS-2 dose assessment utilizes the same methodology discussed above for BVPS-1.

- In accordance with the current BVPS licensing basis, which takes into consideration Alternate Repair Criteria (ARC) for BVPS-2, the BVPS-2 MSLB dose analysis is performed to establish a maximum allowable accident-induced SG tube leakage (AIL), against which the cycle leakage projections can be compared. In accordance with GL 95-05, an accident-induced primary-to-secondary leakage is postulated to occur (via pre-existing tube defects) as a result of the rapid depressurization of the secondary side due to the

MSLB and the consequent high differential pressure across the faulted steam generator. The maximum allowable accident-induced SG tube leakage rate is the maximum primary-to-secondary leakage that could occur in the faulted SG with the offsite or control room operator doses remaining within applicable limits. The BVPS-2 MSLB analysis conservatively assigns this tube leakage to the faulted SG. Consequently, the primary-to-secondary leakage in the faulted steam generator reflects 150 gpd at STP, plus the maximum allowable accident induced tube leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the EAB/LPZ and the control room.

- As indicated earlier in Section 4.2.1, with the intent of increasing the allowable accident-induced primary-to-secondary SG tube leakage limit, FENOC is proposing to lower the BVPS-2 reactor and secondary coolant Technical Specification activity concentrations to 0.10 μCi/gm Dose Equivalent I-131 and 0.05 μCi/gm DE I-131, respectively.

Thus, in accordance with RG 1.183, the activity in the reactor coolant is increased due to the two potential iodine spike scenarios: a pre-accident iodine spike and a concurrent iodine spike as described below.

a. Pre-accident iodine spike - the initial reactor coolant iodine activity is based on a maximum allowable pre-accident iodine spike activity level per the proposed BVPS-2 Technical Specifications of 60 times the equilibrium TS iodine activity concentration of 0.10 μCi/gm DE I-131 or 6 μCi/gm of DE I-131 (transient Technical Specification limit for full power operation). The initial reactor coolant noble gas activity is consistent with the design basis relative mix and activity levels associated with the proposed Technical Specification iodine concentrations in the coolant for BVPS-2.

b. Concurrent iodine spike - the initial reactor coolant iodine activity is assumed to be at the proposed equilibrium TS iodine activity concentration of 0.10 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident, the iodine appearance rate from the fuel to the reactor coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 0.10 μCi/gm DE I-131 coolant concentration allowed by the proposed Technical Specifications for BVPS-2. The duration of the assumed spike is 4 hours. The initial reactor coolant noble gas activity is assumed to be at TS levels.

The resultant reactor coolant activity leaks into the faulted and intact SGs via SG tube leakage and is released to the environment from the break point and MSSVs/ADVs, respectively.

The secondary coolant iodine activity, just prior to the accident, is assumed to be at the proposed BVPS-2 Technical Specification limit of 0.05 μCi/gm DE I-131.

Offsite Dose Assessment

The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

AST methodology requires that the worst case dose to an individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the accident be reported as the EAB dose.

- The Source/Release for the pre-accident iodine spike case is at its maximum levels between 0 and 2 hours.
- For the concurrent iodine spike case, the worst two hour period can occur either a) during the 0 to 2 hr period due to the instantaneous release of the contents of the faulted SG (also, the noble gas release rate is the highest at this time), or b) during the t = 4 to 6 hr period when the iodine level in the SG liquid peaks.

Regardless of the starting point of the "Worst 2 hr Window," the 0 to 2 hrs X/Q is utilized.

Accident Specific Control Room Model Assumptions

The control room emergency ventilation system is manually initiated and a pressurized control room is available at t = 30 minutes after the accident. Following termination of the environmental release, at t = 24 hours, the control room is purged at a minimum flow rate of 16200 cfm for a period of 30 minutes. After purging, the ventilation system is returned to the normal mode of operation.

The remaining CR parameters utilized in these models are discussed in Section 7.1. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes are selected to determine a bounding control room dose. The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a BVPS-1 or BVPS-2 MSLB are provided in Table 7.4-2A and Table 7.4-2B, respectively.

The EAB, LPZ and Control Room doses following a MSLB are presented in Section 8.

7.5 Steam Generator Tube Rupture (SGTR)

This event assumes the instantaneous rupture of a steam generator (SG) tube with a resultant release of reactor coolant into the lower pressure secondary system. Based on an assumption of a Loss of Offsite Power (simultaneous with reactor trip), the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs/ADVs of the intact steam generators are used to cool down the reactor until initiation of shutdown cooling via the RHR system (OPEN ITEM 3). A portion of the reactor coolant break flow into the ruptured SG flashes and is released a) to the condenser before reactor trip and b) directly to the environment after reactor trip, via the MSSVs/ADVs. The remaining break flow mixes with the secondary side liquid and is released to the environment via steam releases through MSSVs/ADVs. The activity in the RCS also leaks into the intact steam generators via SG tube leakage and is released to the environment from the MSSVs/ADVs.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a SGTR at BVPS-1 and BVPS-2.

In accordance with the current BVPS licensing basis established via References 6 and 7, regulatory guidance provided in Appendix F of RG 1.183 is used to develop the dose consequences following the SGTR.

No fuel melting or clad breach is postulated for the BVPS SGTR event. Thus, and in accordance with RG 1.183, the activity released is based on the maximum coolant activity allowed by the BVPS Technical Specifications. In addition, and per RG 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike which reflects the maximum allowable iodine spike activity level per the BVPS Technical Specifications and b) a concurrent iodine spike which results in an increase in the iodine appearance rate from the fuel to the RCS by 335 times.

Table 7.5-1a and Table 7.5-1b list some of the key assumptions/parameters utilized in this application to develop the radiological consequences following the SGTR at BVPS-1 and BVPS-2, respectively.

BVPS-1

In accordance with RG 1.183, the activity in the reactor coolant is increased due to two potential iodine spike scenarios: a pre-accident iodine spike and a concurrent iodine spike as described below.

a. <u>Pre-accident iodine spike</u> - the initial reactor coolant iodine activity is based on a maximum allowable pre-accident iodine spike activity level per the BVPS-1 Technical Specifications of 60 times the equilibrium Technical Specification iodine activity concentration of 0.35 µCi/gm DE I-131 or 21 µCi/gm of DE I-131 (transient Technical Specification limit for full power operation). The initial reactor coolant noble gas activity is consistent with the design basis relative mix and activity levels associated with the BVPS-1 Technical Specification iodine concentrations in the coolant.

b. <u>Concurrent iodine spike</u> - the initial reactor coolant iodine activity is assumed to be at the BVPS-1 equilibrium TS iodine activity concentration of 0.35 µCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the reactor coolant is assumed to increase to 335 times the equilibrium appearance rate corresponding to the 0.35 µCi/gm DE I-131 coolant concentration allowed by the BVPS-1 Technical Specifications. The duration of the assumed spike is 4 hours (*see note below*). The initial reactor coolant noble gas activity is assumed to be at TS levels.

The resultant reactor coolant activity leaks into the faulted and intact SGs via SG tube leakage and is released to the environment from the break point and from the MSSVs/ADVs, respectively.

The secondary coolant iodine activity, just prior to the accident, is assumed to be at the BVPS-1 Technical Specification limit of 0.10 µCi/gm DE I-131.

BVPS Technical Specifications limit primary-to-secondary SG tube leakage to 150 gpd per steam generator for a total of 450 gpd in all 3 SGs.

The current BVPS licensing basis thermal hydraulic analysis model used to determine the dose consequences at the site boundary and control room for the BVPS-1 SGTR is a simplified transient model which was the common industry standard prior to 1980. The model uses a hand calculation method which provides conservative break flows and releases, and utilizes an assumed termination time of 30 minutes for the break flow and releases from the ruptured SG. In accordance with the current BVPS licensing basis, the analysis addresses reduced capacity of the ADVs in that environmental releases are assumed to continue between t = 8 hours to t = 24 hours.

As part of the analyses performed in support of the EPU, an assessment was performed to demonstrate that the dose estimates using the licensing basis thermal hydraulic analysis model bounded that developed based on a more realistic SGTR transient analysis. The thermal hydraulic analyses supporting the "operational response model" took into account realistic operator action times, single failure, margin for steam generator overfill, and a 10-minutes stuck-open ADV. In accordance with the current BVPS licensing basis, the analysis addresses reduced capacity of the ADVs in that environmental releases are assumed to continue between t = 8 hours to t = 24 hours.

Since the SGTR thermal-hydraulic data is not being revised as part of this assessment, the SGTR dose consequences developed as part of this application will focus only on the licensing basis model discussed above.

The SGTR results in a reactor trip and a simultaneous loss of offsite power at 225 seconds after the event. Due to the tube rupture the reactor coolant with elevated iodine concentrations (due to the pre-accident or concurrent iodine spike) flows into the faulted steam generator and the associated activity is released to the environment via secondary side steam releases. Before the reactor trip, the activity is released from the main condenser air ejector. After the reactor trip the steam release is via the MSSVs/ADVs. The reactor coolant activity assumed to leak into the intact steam generator at the maximum allowable primary-to-secondary leakage value is released also to the environment via secondary steam releases.

The initial secondary side liquid and steam activity is relatively small and its contribution to the total dose is small compared to that contributed by the rupture flow. However, the release of the secondary side liquid activity and the resultant doses are also included in this analysis.

Ruptured Steam Generator

A postulated SGTR will result in a large amount of reactor coolant being released to the ruptured steam generator via the ruptured tube with a significant portion of it flashed to the steam space. The noble gases in the entire break flow and the iodine in the flashed flow are assumed to be immediately available for release from the steam generator. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and partition factor. To maximize the calculated offsite doses it is assumed that offsite power is lost (LOOP) so that the main condensers are not available. Before the reactor trip at approximately 225 seconds, the radioactivity in the steam is released to the environment from the main condenser air ejector. The noble gases and organic iodine in the steam are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returned to all three steam generators and assumed to be available for future steaming release. After the reactor trip, the break flow continues until the primary system is fully depressurized. The steam is released from the MSSVs/ADVs. All activity releases from the ruptured steam generator cease when it is isolated at 30 minutes after the accident.

Intact Steam Generators

The activity releases from the intact steam generators is due to normal primary-to-secondary leakage and steam release from the secondary side. The primary-to-secondary leakage is assumed to be at the maximum value allowed by the Technical Specifications, i.e., 150 gpd per SG. All of the iodine activity in the referenced leakage is assumed to mix uniformly with the steam generator liquid and released in proportion to the steaming rate and the partition factor. Before the reactor trip at 225 seconds, the steam is released from the main condenser air ejector. After the reactor trip, the steam is released from the MSSVs/ADVs.

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant; therefore, and per RG 1.183, the iodines are assumed to have a partition coefficient of 100 in the SG and released to the environment in proportion to the steaming rate and the partition coefficient. In accordance with RG 1.183, the iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The noble gases are released freely to the environment without retention in the SG.

The steam releases from the MSSVs/ADVs terminate at 24 hours after the event when shutdown cooling is initiated via the RHR system.

Release of Initial SG Liquid Activity

The initial iodine inventory in the steam generator liquid at TS level (0.10 μCi/gm DE I-131) is released to the environment due to steam releases, via the condenser/air ejector before reactor trip and via MSSVs/ADVs after reactor trip. The release from the ruptured SG stops when the SG is isolated at t = 30 minutes. The release from intact steam generators continues until the RHR system is initiated at 24 hours after the accident

BVPS-2

As indicated earlier in Section 4.2, with the exception of the BVPS-2 MSLB, the BVPS-1 equilibrium TS iodine activity concentration of 0.35 μCi/gm DE I-131 for the reactor coolant and 0.10 μCi/gm DE I-131 for the secondary coolant, will be conservatively used in all BVPS-2 dose consequence analyses that are based on release of coolant at TS concentrations with or without iodine spikes.

Except as noted, the BVPS-2 dose assessment utilizes the same methodology discussed above for BVPS-1.

- The current BVPS licensing basis thermal hydraulic analysis model used to determine the dose consequences at the site boundary and control room for the BVPS-2 SGTR takes into consideration realistic operator action times, single failure, margin for steam generator overfill, and a 10-minutes stuck-open ADV.
- It is noted that the steam release from the faulted SG includes a short period release between 2 and 8 hrs when the faulted SG is manually depressurized in preparation for RHR operation.

- The release from the intact SGs continues until the RHR system is initiated at 8 hours after the accident.

Offsite Dose Assessment

The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

AST methodology requires that the worst case dose to an individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the accident be reported as the EAB dose.

The major source of radioactivity release following a SGTR, and thus dose consequences, is the flashed portion of the RCS break flow, which is released from the main condenser air ejector before reactor trip and from MSSVs/ADVs after reactor trip. For an event at either unit, the break flow release is terminated within approximately 30 to 35 minutes of the accident, when the ruptured SG is isolated. Note: To ensure the maximum 2 hour window dose is calculated for BVPS-2, the brief depressurization steam release from the ruptured steam generator is conservatively modeled as a release during the time period t = 119 to 120 minutes. Therefore, the worst 2 hour window dose occurs during t = 0 to 2 hours after the accident.

Regardless of the starting point of the "Worst 2 hr Window," the 0 to 2 hrs X/Q is utilized.

Accident Specific Control Room Model Assumptions

The control room ventilation remains in the normal mode of operation. Eight hours after the postulated accident, the control room is purged at a minimum flow rate of 16200 cfm for a period of 30 minutes. After purging, the ventilation system is returned to the normal mode of operation.

The remaining CR parameters utilized in these models are discussed in Section 7.1. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes are selected to determine a bounding control room dose. The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a BVPS-1 or BVPS-2 SGTR are provided in Table 7.5-2A and Table 7.5-2B, respectively.

The EAB, LPZ and Control Room doses following a SGTR are presented in Section 8.

7.6 Locked Rotor Accident (LRA) and Loss of AC Power (LACP)

The LRA is caused by an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage is predicted to occur as a result of this accident. There is no breach of the reactor coolant system, however, due to the pressure differential between the primary and secondary systems, and assumed SG tube leakage, elevated concentrations of fission products are assumed to be introduced from the reactor coolant into the secondary coolant system. Following reactor trip, and based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases to the environment via the MSSVs/ADVs until initiation of shutdown cooling (OPEN ITEM 3).

The Loss of Non-Emergency AC Power to the Station Auxiliaries Accident (LACP) is the result of a complete loss of either the external (offsite) grid or the onsite ac distribution system. All reactor coolant pumps are tripped simultaneously by the initiating event resulting in a flow coast-down as well as a decrease in heat removal by the secondary system. The LACP results in the main condenser becoming unavailable, a reactor trip, and reactor cooldown being achieved using steam releases via the MSSVs/ADVs until initiation of shutdown cooling.

The transport models associated with the two events are similar with the exception that the LRA results in fuel damage and associated release of gap activity, whereas the LACP has no fuel damage, and the maximum release is associated with Technical Specification concentrations. Since the reactor coolant Technical Specification activity is significantly smaller than the gap activity associated with failed fuel, it is concluded that the dose consequences of the LRA bound those of the LACP. Consequently, and in accordance with NUREG-0800, Section 15.2.6 (Reference 36) and the current BVPS licensing basis, the LACP is not specifically analyzed.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a LRA at BVPS-1 or BVPS-2.

In accordance with the current BVPS licensing basis established via References 4 and 5, regulatory guidance provided in Appendix G of RG 1.183 is used to develop the dose consequences following the LRA.

Table 7.6-1 and Table 7.6-3 list some of the key assumptions/parameters utilized in this application to assess the radiological consequences following the LRA and LACP at both BVPS-1 and BVPS-2. Parameter values are bounding and encompass the event occurring at either unit.

In accordance with the current BVPS licensing basis, a BVPS LRA results in < 20% failed fuel and a release of the associated gap activity. As discussed in Section 4.3, as part of this application, the fractions of fission product inventory in the gap assumed for the BVPS LRA dose consequence analyses are being changed from those listed in RG 1.183 to those provided in DG-1199.

The gap activity (consisting of noble gases, halogens and alkali metals) is instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary-to-secondary steam generator tube leakage assumed to be at the maximum TS value of 450 gpd at STP.

A radial peaking factor of 1.70 (previously 1.75) is applied to the activity release. The chemical form of the iodines in the gap are assumed to be 95% CsI, 4.85% elemental and 0.15% organic.

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant; therefore, and per RG 1.183, the gap iodines are assumed to have a partition coefficient of 100 in the SG and released to the environment in proportion to the steaming rate and the partition coefficient. In accordance with

RG 1.183, the iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The gap noble gases are released freely to the environment without retention in the SG whereas the particulates are assumed to be carried over in accordance with the design basis SG moisture carryover fraction.

The main condenser is assumed to be unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a LRA is discharged to the environment from the steam generators via the MSSVs/ADVs. The SG releases continue for 8 hours, at which time shutdown cooling is initiated via operation of the RHR system, and environmental releases are terminated.

The activity associated with the release of secondary steam and liquid and primary-to-secondary leakage of normal operation RCS (both at TS activity limits) via the MSSVs/ADVs is insignificant compared to the failed fuel release and is therefore not included in this assessment.

Offsite Dose Assessment

The LRA dose consequence analysis reflects bounding parameter values to encompass an event at either unit. The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

AST methodology requires that the worst case dose to an individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the accident be reported as the EAB dose.

For the LRA, the worst two hour period can occur either during the 0 to 2 hr period when the noble gas release rate is the highest, or during the 6 to 8 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at t = 8 hours).

Regardless of the starting point of the "Worst 2 hr Window," the 0 to 2 hrs X/Q is utilized.

Accident Specific Control Room Model Assumptions

The control room ventilation remains in the normal mode of operation.

The remaining CR parameters utilized in these models are discussed in Section 7.1. The most limiting atmospheric dispersion factors between the MSSVs/ADVs at each unit relative to the two CR intakes is selected to determine a bounding control room dose. The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a BVPS-1 or BVPS-2 LRA or LACP are provided in Table 7.6-2 and Table 7.6-4 respectively.

The EAB, LPZ and Control Room doses following a LRA are presented in Section 8.

7.7 Fuel Handling Accident (FHA) in the Fuel Pool Area or Containment

This event consists of the drop of a single fuel assembly in the Fuel Pool Area located in the Fuel Building or in the Reactor Cavity located inside Containment, during refueling. (Note: a FHA in the transfer canal is not specifically addressed herein since the releases from such an event will occur from either the Fuel Building or the Containment, and therefore the resultant dose consequences are bounded by one or the other of the two events.)

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a FHA in the Fuel Pool Area or Containment at BVPS-1 or BVPS-2.

In accordance with the current BVPS licensing basis established via References 4 and 5, regulatory guidance provided in Appendix B of RG 1.183 is used to develop the dose consequences following the FHA.

Table 7.7-1 lists some of the key assumptions/parameters utilized in this application to develop the radiological consequences following the FHA. Parameter values are bounding and encompass the event occurring at either unit.

BVPS-1

In accordance with the current BVPS licensing basis, the minimum decay time prior to fuel movement in the fuel pool area or containment is 100 hours after reactor shut down. Analyses of the deflections and resulting stresses on the dropped fuel results in the damage of 137 rods. All of the fuel gap activity associated with the damaged rods is assumed to be released. The equilibrium core activity of isotopes in the gap with 100 hours of decay following reactor shutdown is presented in Table 7.7-2. As discussed in Section 4.3, as part of this application, the fractions of fission product inventory in the gap assumed for the BVPS FHA dose consequence analyses are being changed from those listed in RG 1.183 to those provided in DG-1199.

A radial peaking factor of 1.70 (previously 1.75) is applied to the gap release. The activity (consisting of noble gases, halogens, and alkali metals) is released in a “puff” to the fuel pool or reactor cavity.

Per RG 1.183, the radioiodine released from the fuel gap is assumed to be 95% CsI, 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the reactor cavity (pH less than 7), the CsI will immediately disassociate, thus changing the chemical form of iodine in the water to 99.85% elemental and 0.15% organic. The minimum depth of water in the fuel pool and reactor cavity is 23 ft over the top of the damaged fuel assembly. Therefore, per RG 1.183, the pool provides an overall effective decontamination factor for elemental and organic iodines of 200. Per RG 1.183, the chemical form of the iodines above the reactor cavity is 57% elemental and 43% organic.

Noble gas and unscrubbed iodines rise to the water surface whereas all of the alkali metals released from the gap are retained in the water. Since the fuel pool area and containment are assumed to be open, and there is no credit for isolating the accident release, all of the airborne activity resulting from the FHA is exhausted out of the building in a period of 2 hours. The analysis assumes that, during refueling, the ventilation is operational above the fuel pool area.

The exhaust flow from the fuel pool area could occur via the fuel building normal operation release point, i.e., the ventilation vent. The containment could also be purged via the ventilation vent. Since the containment is “open”, releases could also occur from anywhere along the containment wall (e.g., via the equipment or personnel hatch). The exhaust flows from the containment and fuel pool area may also be directed out of the SLCRS release point. Because the location of the release is unknown, the worst case atmospheric dispersion factor is used without taking any credit for SLCRS filtration.

BVPS-2

Except as noted below, the BVPS-2 dose assessment utilizes the same methodology and input parameters discussed above for BVPS-1.

- The site boundary atmospheric dispersion factors reflect unit specific values.
- The analyses reflect the differing Control Room Ventilation System operational requirements during fuel handling at each unit.

Offsite Dose Assessment

The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

AST methodology requires that the worst case dose to an individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the accident be reported as the EAB dose. Since the event is based on a 2 hour release, the worst 2 hour period for the EAB is the 0 to 2 hour period.

Accident Specific Control Room Model Assumptions

The control room emergency ventilation system is not initiated, and the control room ventilation system remains in normal mode. Subsequent to a BVPS-1 FHA, following termination of the environmental release, the control room is purged, at t = 2 hours, at a minimum flow rate of 16200 cfm for a period of 30 minutes. For the BVPS-2 FHA, the control room ventilation system is assumed to remain in the normal mode of operation for the entire duration of the accident. The remaining CR parameters utilized in these models are discussed in Section 7.1.

The most limiting atmospheric dispersion factors for each of the potential release points (i.e., the edge of containment, the top of containment via the SLCRS release point, the ventilation vent, or the equipment hatch) relative to the two CR intakes are selected to determine a bounding control room dose. The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a BVPS-1 or BVPS-2 FHA are provided in Table 7.7-3.

The EAB, LPZ and Control Room doses following a FHA at BVPS-1 and BVPS-2 are presented in Section 8.

7.8 Small Line Break Outside Containment (SLBOC)

A SLBOC is not specifically addressed in RG 1.183. Consequently, and in accordance with the current BVPS licensing basis, the accident scenario for the SLBOC is based on guidance provided in NUREG 0800, Section 15.6.2 (Reference 37).

As outlined in Section 15.6.2, this analysis is intended to address the radiological consequences of the worst case failure outside containment of a small line connected to the reactor coolant pressure boundary (such as the letdown line of the chemical and volume control system (CVCS) or the largest instrument and sample line).

For BVPS, the licensing basis SLBOC is the break of the CVCS letdown line. The failure is assumed to occur downstream of the outboard containment isolation valve in conjunction with a single failure of one of the two containment isolation valves. In accordance with NUREG-0800, Section 15.6.2, the amount of reactor coolant released outside the containment is determined by considering the method, capability and time required to detect such a failure and the time required to isolate the failure (i.e., the time to manually close the operable isolation valve). Specifically, and per NUREG-0800, Section 15.6.2, the amount of reactor coolant released is conservatively estimated by assuming critical flow at the break location with the reactor coolant fluid enthalpy corresponding to normal reactor operating conditions. The initial fission product concentrations in the reactor coolant are assumed to be the maximum equilibrium values permitted by the BVPS Technical Specifications including an assumed concurrent iodine spike as a result of reactor shutdown or depressurization of the primary system that results in increasing the equilibrium fission product activity release rate from the fuel by a factor of 500. The fraction of the iodine assumed to become airborne and available for release to the atmosphere, without credit for plate-out, is equal to the fraction of the coolant flashing into steam in the depressurization process.

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a SLBOC at BVPS-1 or BVPS-2.

In accordance with the current BVPS licensing basis established via References 4 and 5, and as discussed above, regulatory guidance provided in NUREG-0800, Section 15.6.2, is used to develop the dose consequences following the SLBOC.

Table 7.8-1 lists some of the key assumptions/parameters utilized in this application to develop the radiological consequences following the SLBOC. Parameter values are bounding and encompass the event occurring at either unit.

In accordance with the current BVPS licensing basis, the SLBOC postulates the break of the 2-inch reactor coolant letdown line in the Auxiliary Building resulting in a maximum break flow of 16.79 lbm/sec. The break flow rate is based on subcooled critical flow, taking into consideration choked flow via two letdown orifices. The iodine activity in the break flow is assumed to become airborne in proportion to the flash fraction, whereas the noble gases are assumed to be airborne and discharged to the environment without decontamination or holdup. The flash fraction of 37% assumed in the analysis corresponds to a break fluid temperature at reactor coolant conditions (i.e., cooling of the fluid via the regenerative heat exchanger (located inside containment) is conservatively not credited).

In accordance with the current BVPS licensing basis, manual operator action from the control room within 15 minutes of accident initiation is credited to isolate the break. It is noted that if the break was large enough to cause a Safety Injection signal, the letdown line would be automatically isolated.

Since there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the reactor coolant system. In accordance with the current BVPS licensing basis and NUREG-0800, Section 15.6.2, a concurrent iodine spike is included in the source term.

The initial reactor coolant iodine activity is assumed to be 0.35 μCi/gm DE I-131 (current equilibrium BVPS Technical Specification limit for full power operation). Immediately following the accident, the iodine appearance rate from the fuel to the reactor coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 0.35 μCi/gm DE I-131 coolant concentration. In accordance with the current BVPS licensing basis, the duration of the assumed spike is 4 hours. The iodine released from the RCS is assumed to be 97% elemental and 3% organic.

The activity in the auxiliary building is released to the environment via the ventilation vent. The most limiting atmospheric dispersion factors between the ventilation vent release point at each unit relative to the two CR intakes is selected to determine a bounding control room dose. No credit is taken for auxiliary building holdup or filtration.

Offsite Dose Assessment

The SLBOC dose consequence analysis reflects bounding parameter values to encompass an event at either unit. The EAB and LPZ atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are presented in Table 5.1-1.

AST methodology requires that the worst case dose to an individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the accident be reported as the EAB dose. Since the event is based on a 15 minute release, the worst 2 hour period for the EAB is the 0 to 2 hour period.

Accident Specific Control Room Model Assumptions

The control room ventilation system remains in the normal mode of operation.

The remaining CR parameters utilized in these models are discussed in Section 7.1. The most limiting atmospheric dispersion factors between the ventilation vent at each unit relative to the two CR intakes is selected to determine a bounding control room dose. The bounding atmospheric dispersion factors applicable to the radioactivity release points/control room receptors for a BVPS-1 or BVPS-2 SLBOC are provided in Table 7.8-2.

The EAB, LPZ and Control Room doses following a SLBOC are presented in Section 8.

Table 7.1-1[1] Analysis Assumptions & Key Parameter Values BVPS Common Control Room	
Control Room Parameters	
Free Volume	173,000 ft^3
Normal Operation Unfiltered Air Intake (includes unfiltered air inleakage and a 10 cfm allowance for ingress/egress)	< 1250 cfm
Isolation Mode Unfiltered Air Inleakage (includes 10 cfm for ingress/egress)	< 450 cfm
Emergency Mode Filtered Air Intake	800 to 1000 cfm
Emergency Mode Filtered Recirculation	Not Credited
Emergency Mode Intake Filter Efficiency	99% (aerosols) 98% (elemental/organic iodine)
Emergency Mode Recirculation Filter Efficiency	N/A
Emergency Mode Unfiltered Air Inleakage (includes 10 cfm for ingress/egress)	< 165 cfm
Occupancy Factors	0 to 24 hours (1.0) 1 to 4 days (0.6) 4 to 30 days (0.4)
Operator Breathing Rate	0 to 30 days (3.5E-04 m^3/sec)
Delay in Initiation of Control Room Emergency Ventilation System considering a LOOP	
Automatic (on receipt of CIB signal)	
Control Room ventilation is automatically isolated (includes emergency diesel generator startup and load sequencing)	t = 77 seconds
Control Room ventilation is automatically placed in Emergency mode (automatic function is not credited in the analysis)	t = 137 seconds
Manual	
Control Room ventilation is manually placed in Emergency mode (after being automatically isolated at t = 77 seconds)	t = 30 minutes

Note:(1) Bounding parameter values are used to encompass an event at either unit.

Table 7.2-1[1] **Analysis Assumptions & Key Parameter Values** **BVPS-1 and BVPS-2 Loss of Coolant Accident**	
Containment Leakage	
Power Level	2918 MWT
Minimum Free Volume	1.75E+6 ft^3
Sprayed Fraction	60%
Spray Period	77.4 sec to 96 hours
Mixing Rate	2 Unsprayed vol/hr
Containment Leakrate (0 to 24 hrs)	0.1% vol fractions per day
Containment Leakrate (1 to 30 days)	0.05% vol fractions per day
Maximum DF for Elemental Iodine	200
Sump/Recirculation Spray pH	≥ 7.0
Fuel Activity Release Fractions	Per RG 1.183
Fuel Release Timing (Gap Release)	Onset: 30 sec Duration: 0.5 hr
Fuel Release Timing (Early In-Vessel Release)	Onset: 0.5 hr Duration: 1.3 hr
Chemical Form of Iodine Released	4.85% elemental 95% particulate 0.15% organic
Spray Removal Constants	Figure 7.2-1 & Figure 7.2-2, and Table 7.2-2
Equilibrium Core Inventory	Table 4.1-1
Environmental Release Point	Worst case between Containment Wall & SLCRS Vent (Containment Top)
ECCS/RWST Leakage Parameters	
Minimum Sump Volume (5 to 30 min) (0.5 to 2 hr) (2 hr to 30 day)	19,253 ft^3 (1.1379E6 lbm) 24,909 ft^3 (1.5133E6 lbm) 43,824 ft^3 (2.6837E6 lbm)

Table 7.2-1(1) (Continued) Analysis Assumptions & Key Parameter Values BVPS-1 and BVPS-2 Loss of Coolant Accident	
ESF Leakage	
ESF Leakrate	11400 cc/hr (2 × expected)
Leakage Period	1200 sec to 30 days
Iodine Release Fraction	0.1
Chemical Form of Iodine Released	97% elemental; 3% organic
Environmental Release Point	SLCRS Vent (Containment Top) as an unfiltered PUFF release
RWST (back-leakage)	
Sump water back-flow to RWST	2 gpm (includes factor of 2 margin)
Onset of back-leakage	1768 seconds
Onset of RWST activity venting	3039 seconds
End of release period	30 days
Iodine Release Fraction via RWST	Figure 7.2-3 & Table 7.2-3
Environmental Release Point	RWST Vent
Containment Vacuum System	
Reactor Coolant Tech Spec Activity	Table 4.2-1A (bounds Table 4.2-1B)
Chemical Form of Iodine Released	97% elemental; 3% organic
Containment Vacuum System Release	2200 scfm (5 second release)
Environmental Release Point	Worst case between Containment Wall & SLCRS Vent (Containment Top)
CREVS Initiation Signal Timing	
Initiation time (signal)	Assumed to be at t = 0 (CIB)
Control Room Isolation time	t = 77 seconds (automatic)
Emergency mode initiation time	t = 30 minutes (manual initiation)

Note:(1) Bounding parameter values are used to encompass an event at either unit.

Table 7.2-2
Aerosol and Elemental Iodine Removal Rates within Sprayed and Unsprayed Regions of Containment

From (sec)	To (sec)	From (hour)	To (hour)	Sprayed Region λ_P (1/hr) (particulate)	Sprayed Region λ_E (1/hr) (elemental)	Unsprayed Region λ_P (1/hr) (particulate)	Unsprayed Region λ_E (1/hr) (elemental)
0	30	0.00000	0.00833	0	0	0	0
30	77	0.00833	0.02139	9.5823	4.1075	0.0030	4.1075
77	263	0.02139	0.07300	5.1476	5.1476	0.0032	0
263	507	0.07300	0.14096	3.4057	3.4057	0.0034	0
507	722	0.14096	0.20056	2.9256	2.9256	0.0037	0
722	967	0.20056	0.26860	2.5650	2.5650	0.0041	0
967	1434	0.26860	0.39841	2.1868	2.1868	0.0048	0
1434	1800	0.39841	0.50000	2.0967	2.0967	0.0057	0
1800	1830	0.50000	0.50833	2.0482	2.0482	0.0061	0
1830	1851	0.50833	0.51416	2.7954	2.7954	0.0090	0
1851	1963	0.51416	0.54541	6.7158	6.7158	0.0213	0
1963	2233	0.54541	0.62038	11.6469	11.6469	0.0387	0
2233	2570	0.62038	0.71383	14.5583	14.5583	0.0500	0
2570	2975	0.71383	0.82628	14.6596	14.6596	0.0545	0
2975	3536	0.82628	0.98233	14.6855	14.6855	0.0578	0
3536	3896	0.98233	1.08230	16.5427	16.5427	0.0611	0
3896	4817	1.08230	1.33802	29.0424	20.5358	0.0650	0
4817	5244	1.33802	1.45671	30.1924	20.5358	0.0688	0
5244	5599	1.45671	1.55529	30.2929	20.5358	0.0706	0
5599	6460	1.55529	1.79435	30.4578	20.5358	0.0727	0
6460	6510	1.79435	1.80833	30.3747	20.5358	0.0741	0
6510	6529	1.80833	1.81353	27.3476	20.5358	0.0743	0
6529	6604	1.81353	1.83451	18.7190	18.7190	0.0749	0
6604	6803	1.83451	1.88966	11.6128	11.6128	0.0764	0
6803	7177	1.88966	1.99361	8.0644	8.0644	0.0792	0
7177	7200	1.88966	2.00000	6.3482	6.3482	0.0793	0
7200	7487	2.00000	2.07982	5.7581	5.7581	0.0822	0
7487	7870	2.07982	2.18606	4.8614	4.8614	0.0843	0
7870	8434	2.18606	2.34274	4.0857	4.0857	0.0864	0
8434	8992	2.34274	2.49782	3.4963	3.4963	0.0882	0
8992	9579	2.49782	2.66091	3.0940	3.0940	0.0894	0
9579	10832	2.66091	3.00886	2.3312	2.3312	0.0901	0

Table 7.2-2 (Continued)
Aerosol and Elemental Iodine Removal Rates within Sprayed and Unsprayed Regions of Containment

From (sec)	To (sec)	From (hour)	To (hour)	Sprayed Region		Unsprayed Region	
				λ_P (1/hr) (particulate)	λ_E (1/hr) (elemental)	λ_P (1/hr) (particulate)	λ_E (1/hr) (elemental)
10832	12518	3.00886	3.47714	1.5493	1.5493	0.0901	0
12518	18000	3.47714	5.00000	1.3892	1.3892	0.0874	0
18000	23040	5.00000	6.40000	1.2533	1.2533	0.0831	0
23040	28800	6.40000	8.00000	1.1568	1.1568	0.0789	0
28800	36000	8.00000	10.00000	1.0765	1.0765	0.0746	0
36000	84285	10.00000	23.41250	0.8901	0.8901	0.0498	0
84285	202202	23.41250	56.16722	0.6923	0.6923	0	0
202202	345600	56.16722	96.00000	0.4	0	0	0

Table 7.2-3
Design and Bounding RWST Iodine Release Rates

Period		RWST Vent Iodine Release Rates	
From (hour)	To (hour)	Design[1] (day^{-1})	Calculated[2] [4] (day^{-1})
0	0.7011389	0	0
0.7011389	0.83444	0	1.592E-02
0.83444	0.84861	0	7.491E-03
0.84861[3]	1.06777	1.0E-02	7.491E-03
1.06777	1.66667	1.0E-02	2.093E-03
1.66667	1.75	8.0E-03	2.093E-03
1.75	1.7844	6.0E-03	2.093E-03
1.7844	2	6.0E-03	5.773E-04
2	3	4.0E-03	6.500E-05
3	5	2.0E-03	6.500E-05
5	8	1.1E-03	9.302E-06
8	9	1.1E-03	2.807E-06
9	11	1.1E-03	2.807E-06
11	24	2.4E-04	2.807E-06
24	48	2.4E-04	1.548E-06
48	72	1.1E-04	8.210E-07
72	96	3.0E-05	8.210E-07
96	120	1.0E-05	5.524E-07
120	144	6.0E-06	4.841E-07
144	168	2.0E-06	4.386E-07
168	192	1.0E-06	3.985E-07
192	216	8.0E-07	3.667E-07
216	264	7.0E-07	3.289E-07
264	312	6.0E-07	2.901E-07
312	384	5.0E-07	2.534E-07
384	480	4.0E-07	2.115E-07
480	576	3.0E-07	1.844E-07
576	672	2.4E-07	1.574E-07
672	720	2.0E-07	1.574E-07

Notes:
[1] Design RWST Iodine Release Rate values used in LAR Nos. 334 and 205 and herein to establish dose consequences.
[2] BVPS-1 & BVPS-2 bounding values that reflect inclusion of NaTB baskets as the sump water buffering agent and the impact of NSAL-11-5 on the BVPS-2 LOCA transient.
[3] Time 0.84861 hour corresponds to 3055 seconds, the start time for the environmental release in Reference 8. This time is updated to 0.84417 hr (3039 seconds).
[4] An assessment of the impact of NSAL-11-5 on BVPS-1 LOCA M&E releases on the updated RWST iodine release fractions presented above determined that the effect (primarily due to increased sump water temperatures) would be small, and any minor increase would remain bounded by the design values used in the dose consequence analysis.

Table 7.2-4 Analysis Assumptions & Key Parameter Values BVPS Emergency Response Facility	
ERF Parameters	
Minimum Free Volume	478,610 ft^3
Normal Operation Filtered Air Intake	3800 cfm +/-10%
Intake Filter Efficiency (HEPA)	100% or 0% (particulate), whichever is more conservative
Normal Operation Unfiltered Air Inleakage	2090 cfm
Emergency Mode Filtered Air Recirculation	7200 cfm +/-10%
Recirculation Filter Efficiency	100% (particulate) 100% (elemental/organic iodine)
Emergency Mode Filtered Air Intake	0 cfm
Emergency Mode Unfiltered Air Inleakage	910 cfm (includes 10 cfm for ingress/egress)
Occupancy Factors	0 to 24 hours (1.0) 1 to 4 days (0.6) 4 to 30 days (0.4)
Occupant Breathing Rate	0 to 30 days (3.5E-04 m^3/sec)
Delay in Initiation of ERF Emergency Ventilation	
Manual	
ERF in emergency recirculation mode	t = 30 minutes

Table 7.2-5
Control Room/ERF Limiting Atmospheric Dispersion Factors (sec/m^3)
BVPS-1 and BVPS-2 Loss of Coolant Accident

Release Location / Receptor	0 to 2 hr	2 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
<u>CR Bounding Dispersion Factors</u>					
Containment leakage	8.16E-04	5.78E-04	2.53E-04	2.00E-04	1.78E-04
ESF leakage	8.16E-04	5.78E-04	2.27E-04	1.71E-04	1.47E-04
RWST leakage	7.34E-04	6.17E-04	2.54E-04	1.96E-04	1.57E-04
<u>ERF Bounding Dispersion Factors</u>					
Containment leakage	7.22E-05	6.43E-05	2.96E-05	2.48E-05	2.15E-05
ESF leakage	7.22E-05	6.43E-05	2.96E-05	2.48E-05	2.15E-05
RWST leakage	9.42E-05	8.37E-05	3.81E-05	2.97E-05	2.58E-05

<u>Notes</u>:

1. CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake.
2. ERF X/Q values – Bounding values based on BVPS-2 release points to the closest edge of the ERF.
3. Containment leakage can either be released directly to the environment or into adjacent areas that are ventilated by the SLCRS. Since no credit is taken for SLCRS filtration, dispersion is the only means of lowering the leakage concentrations. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflect the worst value between the containment wall release point and the SLCRS release point for each time period.
4. ESF leakage is released via the SLCRS release point.

Table 7.2-6
Containment Thermodynamic Data
BVPS-1 & BVPS-2 Loss of Coolant Accident

Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
0.0	14.2	0.0	108.0	0.0	0.0	0.0	0.500
0.001	14.2	0.001	108.1	40.1	279591	5.0	0.500
40.1	52.3	40.1	258.5	80.1	120655	10.0	1.000
80.2	51.2	80.2	256.0	120.2	101211	345600.0	1.000
120.6	49.6	120.6	253.2	160.2	84363		
160.2	49.0	160.2	251.3	200.2	74004		
200.2	48.3	200.2	250.0	240.2	66345		
240.2	47.8	240.2	249.4	280.7	61909		
280.7	47.6	280.7	248.9	321.0	58435		
321.0	47.4	321.0	248.5	361.0	55660		
361.0	47.4	361.0	248.4	401.0	53361		
401.0	47.3	401.0	248.3	441.0	51477		
441.0	47.3	441.0	248.3	481.0	49834		
481.0	47.4	481.0	248.3	522.1	48459		
522.1	47.5	522.1	248.4	562.1	47186		
562.1	47.6	562.1	248.6	602.1	47106		
602.1	47.7	602.1	248.9	642.1	46143		
642.1	47.9	642.1	249.2	682.1	45339		
682.1	48.1	682.1	249.5	722.1	44671		
722.1	48.2	722.1	249.8	762.1	44046		
762.1	48.4	762.1	250.1	802.1	43450		
802.1	48.6	802.1	250.5	842.1	40987		

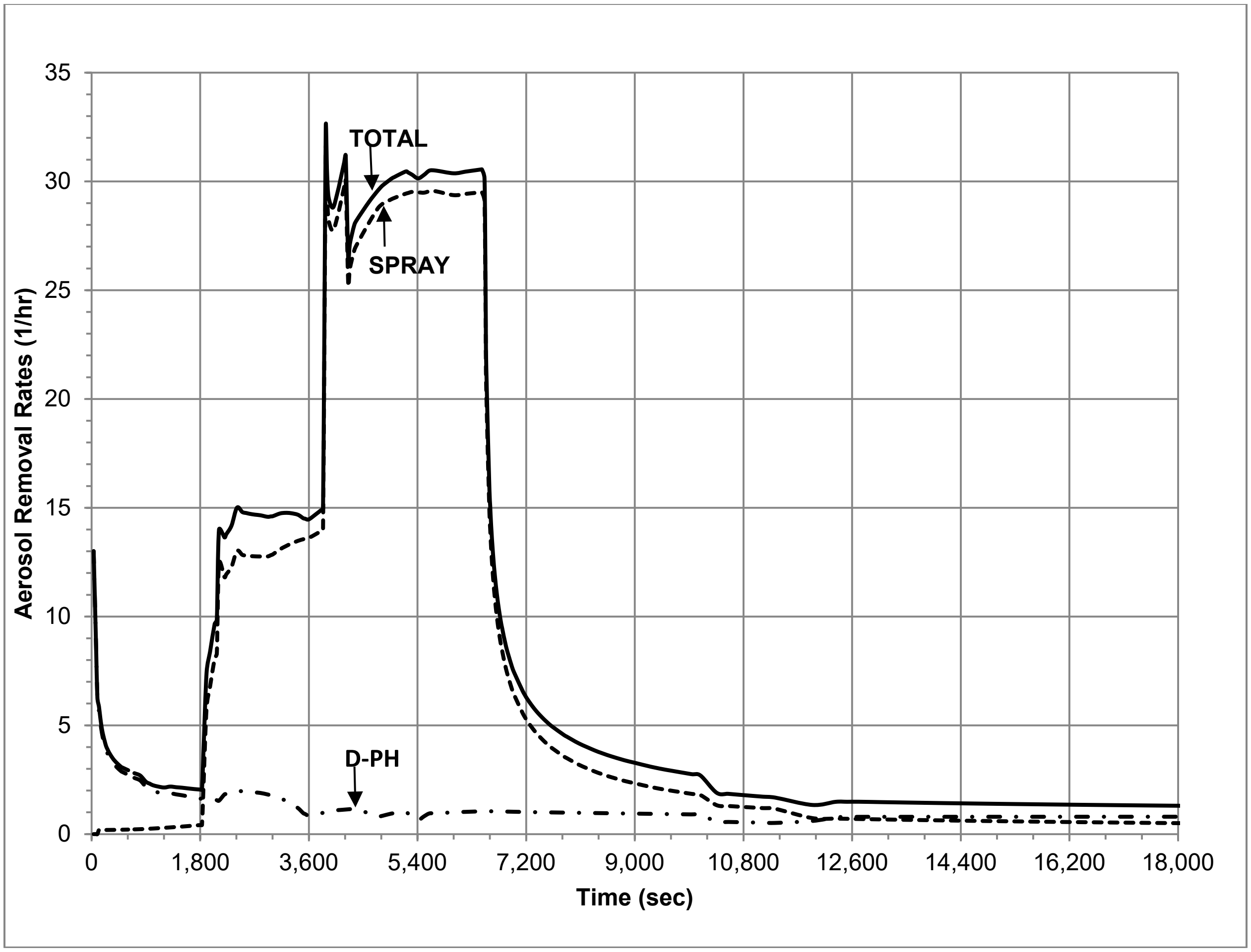
Table 7.2-6 (continued) Containment Thermodynamic Data BVPS-1 & BVPS-2 Loss of Coolant Accident							
Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
842.1	48.9	842.1	250.9	882.1	39002		
882.1	48.6	882.1	250.4	922.1	36625		
922.1	48.3	922.1	249.8	962.1	35512		
962.1	48.0	962.1	249.2	1002.1	34577		
1002.1	47.7	1002.1	248.6	1042.1	33760		
1042.1	47.4	1042.1	248.1	1082.1	33032		
1082.1	47.2	1082.1	247.7	1122.1	32377		
1122.1	47.0	1122.1	247.3	1162.1	31885		
1162.1	46.8	1162.1	246.9	1202.1	31344		
1202.1	46.6	1202.1	246.6	1242.1	32339		
1242.1	46.5	1242.1	246.3	1282.1	31822		
1282.1	46.3	1282.1	246.0	1322.1	31350		
1322.1	46.2	1322.1	245.7	1362.1	30936		
1362.1	46.1	1362.1	245.4	1402.1	30555		
1402.1	45.9	1402.1	245.2	1442.1	30201		
1442.1	45.8	1442.1	244.9	1482.1	29871		
1482.1	45.7	1482.1	244.7	1522.1	29564		
1522.1	45.6	1522.1	244.5	1562.1	29183		
1562.1	45.4	1562.1	244.1	1602.1	28593		
1602.1	45.2	1602.1	243.6	1642.1	28095		
1642.1	45.0	1642.1	243.2	1682.1	27661		

Table 7.2-6 (continued)
Containment Thermodynamic Data
BVPS-1 & BVPS-2 Loss of Coolant Accident

Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
1682.1	44.8	1682.1	242.8	1722.1	27262		
1722.1	44.6	1722.1	242.4	1800.1	26465		
1762.1	44.4	1762.1	242.0	2004.0	25498		
1804.0	44.2	1804.0	241.6	2151.1	23937		
1844.0	44.0	1844.0	241.2	2164.0	28256		
1884.0	43.8	1884.0	240.8	2844.0	25766		
1924.0	43.6	1924.0	240.5	3066.0	23153		
1964.0	43.5	1964.0	240.1	3248.4	19907		
2004.0	43.3	2004.0	239.8	3448.4	14942		
2044.0	43.1	2044.0	239.4	3548.4	10768		
2078.0	43.0	2078.0	239.1	4049.5	13086		
2093.0	43.0	2080.0	239.1	4359.5	13219		
2140.0	42.8	2081.0	239.3	4759.5	8477		
2170.0	42.1	2100.0	239.1	4959.5	10098		
2212.0	41.0	2152.4	238.2	5059.5	10485		
2252.0	40.1	2200.0	234.3	5159.5	10929		
2350.0	37.9	2252.0	231.3	5259.5	11132		
2404.0	36.3	2964.0	216.9	5363.8	8500		
2564.0	35.5	3605.0	209.0	5463.8	9584		
2964.0	33.8	3877.5	204.6	5563.8	10871		
3084.0	33.3	3957.0	196.5	5663.8	10344		
3404.0	32.1	4749.5	180.1	5763.8	10871		

Table 7.2-6 (continued) Containment Thermodynamic Data BVPS-1 & BVPS-2 Loss of Coolant Accident							
Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
3605.0	31.4	4849.5	183.7	5863.8	11165		
3957.0	29.0	5749.5	191.2	6526.9	11867		
4749.5	24.9	6749.5	191.4	7329.6	11391		
4849.5	25.3	7263.9	192.6	10144.7	10108		
5949.5	27.2	18074.7	188.9	10274.1	6472		
6749.5	27.2	35998.0	180.5	11662.9	5552		
7263.9	27.4	72198.0	170.9	12362.9	8920		
18074.7	26.8	144020.0	162.6	20062.9	8936		
35998.0	24.9	259020.0	155.3	30080.0	8293		
72198.0	23.1	346020.0	150.4	49997.6	7255		
144020.0	21.7			59997.6	6910		
259020.0	20.8			69997.6	6689		
346020.0	20.2			79997.6	6368		
				90000.0	6020		
				100000.0	5643		
				200020.0	5044		
				259020.0	4685		
				300020.0	4455		
				346028.5	4189		

Figure 7.2-1
Aerosol Removal Rates Within Sprayed Region (Design Basis LOCA Case)



Notes:

- Total - Represents total aerosol removal rates
- Spray - Represents aerosol removal rates due to sprays
- D-PH – Represents aerosol removal rates due to diffusiophoresis

Figure 7.2-2
Aerosol Removal Rates Within Unsprayed Region (Design Basis LOCA Case)

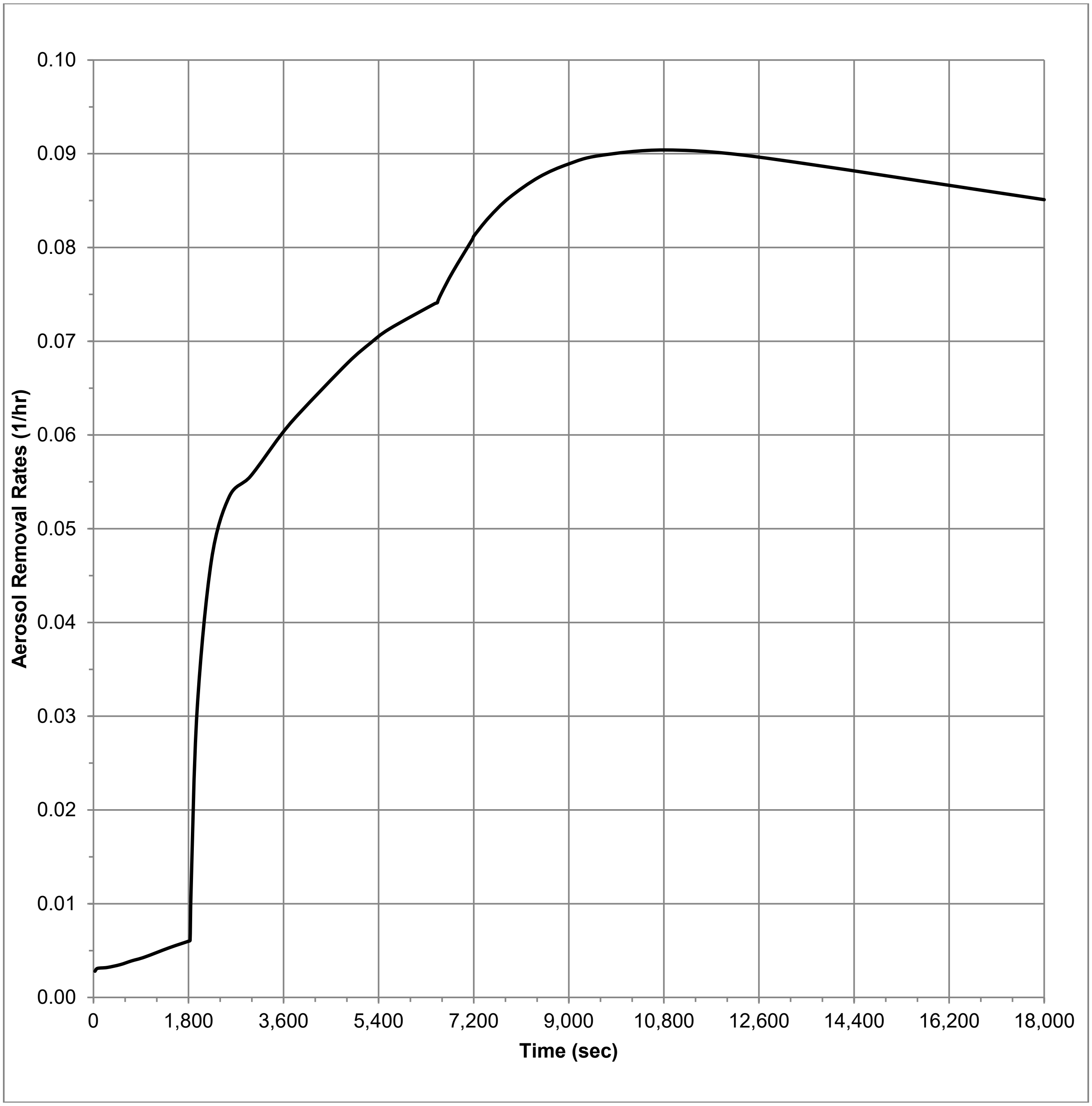
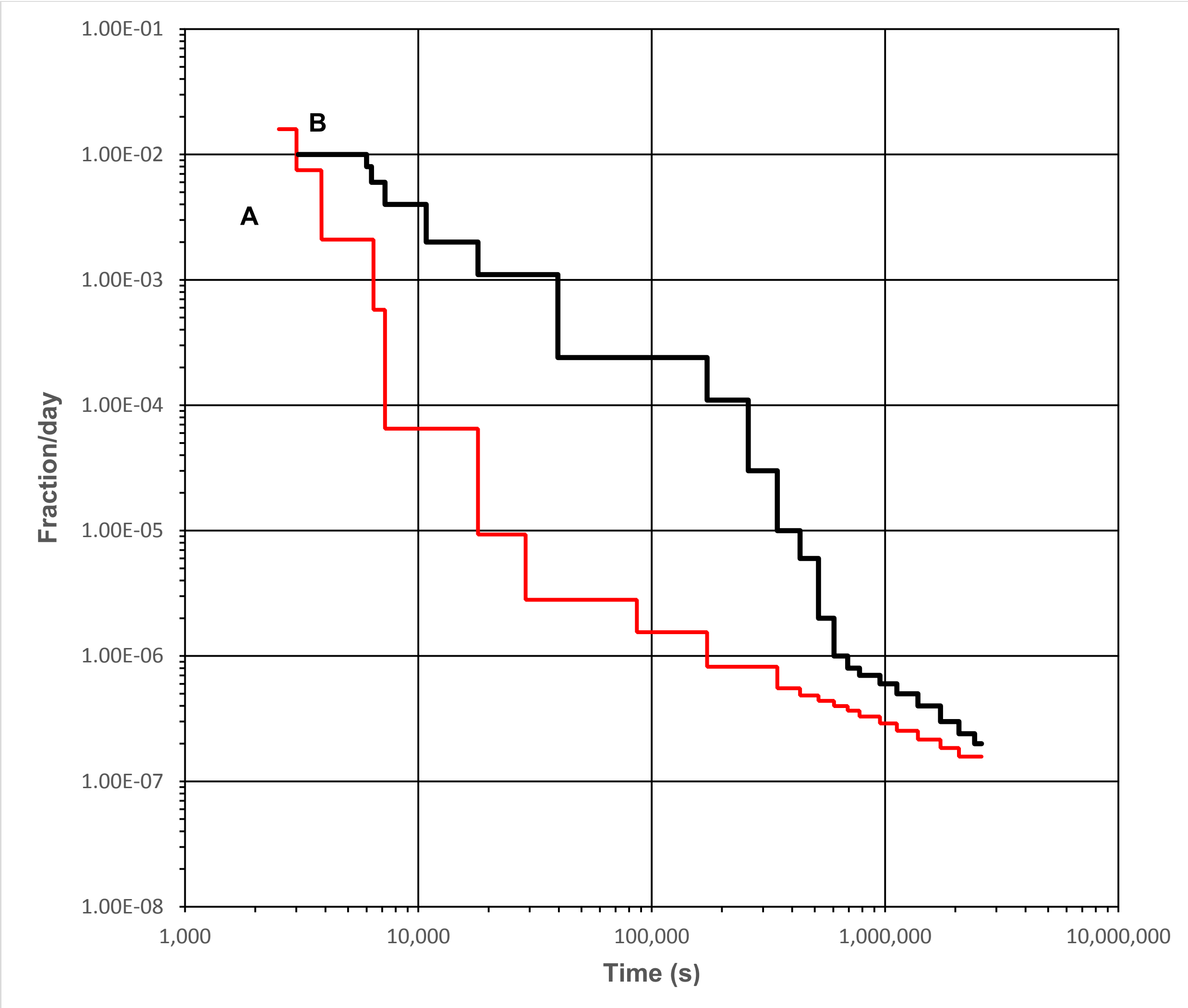


Figure 7.2-3
RWST Iodine Release Fractions



Legend:

A: BVPS-1 & BVPS-2 bounding values that reflect inclusion of NaTB baskets as the sump water buffering agent and impact of NSAL-11-5 on the BVPS-2 LOCA transient. A follow-up assessment of the impact of NSAL-11-5 on BVPS-1 LOCA M&E releases on the updated RWST iodine release fractions determined that the effect (primarily due to increased sump water temperatures) would be small and remains bounded by the design RWST Iodine Release rates.

B: Design RWST Iodine Release rates - used in LAR Nos. 334 and 205 and herein to establish dose consequences.

Table 7.3-1(1)
Analysis Assumptions & Key Parameter Values
BVPS-1 and BVPS-2 Control Rod Ejection Accident

Containment Pathway Parameters	
Power Level	2918 MWth
Minimum Free Volume	1.75E+6 ft^3
Containment Leakrate (0 to 24 hr)	0.10% vol fractions per day
Containment Leakrate (1 to 30 day)	0.05% vol fractions per day
Failed Fuel Percentage	< 10%
Percentage of Core Inventory in Fuel Gap	10% (noble gases & halogens)
Melted Fuel Percentage	< 0.25%
Percentage of Core Inventory in Melted Fuel released to Containment Atmosphere	100% Noble Gas; 25% Halogens
Chemical Form of Iodine in Failed/Melted Fuel	4.85% elemental; 95% CsI 0.15% organic
Radial Peaking Factor	1.70
Core Activity Release Timing	PUFF
Form of Iodine in the Containment Atmosphere	97% elemental; 3% organic
Equilibrium Core Inventory	Table 4.1-1
Termination of Containment Release	30 days
Environmental Release Point	Containment Wall/SLCRS Vent (Containment Top)
Secondary Side Pathway Parameters	
Minimum Reactor Coolant Mass	341,331 lbm
Primary-to-Secondary Leakage	150 gpd per SG at STP, 450 gpd total
Termination of Primary-to-Secondary Leakage	2500 secs
Fraction of Failed/Melted Fuel	Same as Containment Pathway

Note:
(1) Bounding parameter values are used to encompass an event at either unit.

Table 7.3-1 (Continued) Analysis Assumptions & Key Parameter Values BVPS-1 and BVPS-2 Control Rod Ejection Accident		
Percentage of Core Inventory in Melted Fuel released to Reactor Coolant		100% Noble Gas; 50% Halogens
Iodine Species released to Environment		97% elemental; 3% organic
Iodine Partition Coefficient		100 (all tubes submerged)
Fraction of Noble Gas Released		1.0 (released without holdup)
Minimum Post-Accident SG Liquid Mass		99,217 lbm per SG
Steam Releases per SG	0 to 150 secs	900 lbs/sec
	150 to 300 secs	300 lbs/sec
	300 to 2500 secs	150 lbs/sec
	2500 secs to 8 hrs	778,000 lbs
	8 hrs to 30 days	Rate of release conservatively assumed to be same as that applicable to the previous time period.
Termination of Release from SGs		30 days
Environmental Release Point		MSSVs/ADVs
CREVS Initiation Signal Timing		
Emergency mode initiation time		t = 30 minutes (manual initiation)

Table 7.3-2 Control Room Limiting Atmospheric Dispersion Factors (sec/m^3) Control Rod Ejection Accident					
Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
CR Bounding Dispersion Factors					
Containment leakage	8.16E-04	5.78E-04	2.53E-04	2.00E-04	1.78E-04
MSSVs/ADVs	1.24E-03	9.94E-04	4.08E-04	3.03E-04	2.51E-04

Notes:

1. CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake.
2. Containment leakage can either be released directly to the environment, or into adjacent areas that are ventilated by the SLCRS. Since no credit is taken for SLCRS filtration, dispersion is the only means of lowering the leakage concentrations. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflect the worst value between the containment wall release point and the SLCRS release point for each time period.

Table 7.4-1A **Analysis Assumptions & Key Parameter Values** **BVPS-1 Main Steam Line Break**	
Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	345,097 lbm
Leakrate into Faulted Steam Generator	150 gpd at STP
Amount of Accident Induced Leakage (AIL) into Faulted SG	N/A
Maximum time to cool RCS to 212F	19 hrs
Leakrate into Intact Steam Generators	300 gpd total from 2 SGs at STP
Failed/Melted Fuel Percentage	0%
Reactor Coolant Tech Spec Iodine & NG Activity Concentration	Table 4.2-1A (0.35 µCi/gm DE I-131)
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2A (0.35 µCi/gm DE I-131)
Pre-Accident Iodine Spike Activity	Table 4.2-2A (21 µCi/gm DE I-131)
Concurrent Iodine Spike Appearance Rate	500 times equilibrium appearance rates
Duration of Concurrent Iodine Spike	4 hours
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Secondary Coolant Tech Spec Iodine Activity Concentration	Table 4.2-1A (0.10 µCi/gm DE I-131)
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (released without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (released without holdup)
Fraction of Noble Gas Released from Faulted SG	1.0 (released without holdup)
Minimum & Initial Post-Accident Intact SG Liquid Mass	101,799 lbm per SG
Maximum Liquid in Faulted SG	163,150 lbm
Steam Releases from Intact SG	345,000 lbm (0 to 2 hr) 734,000 lbm (2 to 8 hr)
Dryout of Faulted SG	Instantaneous
Termination of release from Faulted SG	19 hours
Termination of release from Intact SG	8 hours
Environmental Release Point: Faulted SG	Break Point
Environmental Release Point: Intact SG	MSSVs/ADVs
CREVS Initiation Signal Timing	
Emergency mode initiation time	t = 30 minutes (manual initiation)
Control Room Purge (Time/Rate)	24 hours after DBA 30 minutes at 16200 cfm (minimum)
Note: Faulted SG contents released as a PUFF release	

Table 7.4-1B
Analysis Assumptions & Key Parameter Values
BVPS-2 Main Steam Line Break

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass (min)	341,331 lbm
Leakrate into Faulted Steam Generator	150 gpd at STP
Amount of Accident Induced Leakage (AIL) into Faulted SG.	8.1 gpm at STP
Maximum time to cool RCS to 212F	21 hrs
Leakrate to Intact Steam Generators	300 gpd total from 2 SGs at STP
Failed/Melted Fuel Percentage	0%
Reactor Coolant Tech Spec Iodine & NG Activity Concentration	Table 4.2-1B (0.10 µCi/gm DE I-131)
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2B (0.10 µCi/gm DE I-131)
Pre-Accident Iodine Spike Activity	Table 4.2-2B (6 µCi/gm DE I-131)
Concurrent Iodine Spike Appearance Rate	500 times equilibrium appearance rates
Duration of Concurrent Iodine Spike	4 hours
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Secondary Coolant Tech Spec Iodine Activity Concentration	Table 4.2-1B (0.05 µCi/gm DE I-131)
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (released without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (released without holdup)
Fraction of Noble Gas Released from Faulted SG	1.0 (released without holdup)
Initial & Minimum Post-Accident Intact SG Liquid Mass	105,076 lbm per SG
Maximum Liquid in Faulted SG	162,800 lbm
Steam Releases from Intact SG	350,000 lbm (0 to 2 hr) 730,000 lbm (2 to 8 hr)
Dryout of Faulted SG	Instantaneous
Termination of release from Faulted SG	21 hours
Termination of release from Intact SG	8 hours
Environmental Release Point: Faulted SG	Break Point
Environmental Release Point: Intact SG	MSSVs/ADVs
CREVS Initiation Signal Timing	
Emergency mode initiation time	t = 30 minutes (manual initiation)
Control Room Purge (Time/Rate)	24 hours after DBA 30 minutes at 16200 cfm (minimum)

Note:
Faulted SG contents released as a PUFF release

Table 7.4-2A Control Room Limiting Atmospheric Dispersion Factors (sec/m^3) BVPS-1 Main Steam Line Break					
Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
CR Bounding Dispersion Factors					
BVPS-1 MS Line break location	1.05E-02	7.72E-03	3.01E-03	------	------
BVPS-1 MSSVs/ADVs	1.24E-03	9.94E-04	------	------	------

Note:
CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake

Table 7.4-2B Control Room Limiting Atmospheric Dispersion Factors (sec/m^3) BVPS-2 Main Steam Line Break					
Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
CR Bounding Dispersion Factors					
BVPS-2 MS Line break location	1.03E-03	7.84E-04	3.57E-04	------	------
BVPS-2 MSSVs/ADVs	5.01E-04	3.58E-04	------	------	------

Note:
CR X/Q values – Bounding values based on BVPS-2 release points and the BVPS-2 CR Intake

Table 7.5-1A Analysis Assumptions & Key Parameter Values BVPS-1 Steam Generator Tube Rupture	
Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	373,100 lbm
Break Flow to Ruptured Steam Generator	21,900 lbm (0 to 225 sec) 128,000 lbm (225 to 1800 sec)
Time of Reactor Trip	225 sec
Termination of Release from Ruptured SG	1800 seconds
Fraction of Break Flow that Flashes	0.2227 (0 to 225 sec) 0.1645 (225 to 1800 sec)
Leakage Rate to Intact Steam Generators	150 gpd at STP for each SG
Failed/Melted Fuel Percentage	0%
Reactor Coolant Tech Spec Iodine & NG Activity Concentration	Table 4.2-1A (0.35 μCi/gm DE I-131)
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2A (0.35 μCi/gm DE I-131)
Pre-Accident Iodine Spike Activity	Table 4.2-2A (21 μCi/gm DE I-131)
Concurrent Iodine Spike Appearance Rate	335 times equilibrium appearance rates
Duration of Concurrent Iodine Spike	4 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	91,000 lbm
Ruptured SG Liquid Mass (min)	91,000 lbm
Initial SG Liquid Mass per Steam Generators	96,000 lbm
Secondary Coolant Tech Spec Iodine Activity Concentration	Table 4.2-1A (0.10 μCi/gm DE I-131)
Form of All Iodine Released to the Environment via Steam Generators	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion)	1.0 (released without holdup)
Fraction of Noble Gas Released from any SG	1.0 (released without holdup)
Partition Factor in Condenser	100 elemental iodine 1 organic iodine / Noble Gases
Steam Flowrate to Condenser before Reactor Trip	1207.407 lbm/sec per SG (0 to 225 sec)
Ruptured SG Steam Releases via MSSVs/ADVs	68,900 lbm (225 to 1800 sec)
Intact SG Steam Releases via MSSVs/ADVs	417,100 lbm (225 to 7200 sec) 979,500 lbm (2 to 8 hr) 658,400 lbm (8 to 16 hr) 546,700 lbm (16 to 24 hr)
Termination of Release from Intact SGs	24 hours
Environmental Release Points	Condenser Air Ejector (0 to 225 sec) MSSVs/ADVs (225 sec to 24 hr)
CREVS Initiation Signal Timing	
Control Room ventilation is maintained in Normal mode	
CR Purge Initiation (Manual) Time and Rate	8 hours after DBA 30 minutes at 16200 cfm (minimum)

Table 7.5-1B **Analysis Assumptions & Key Parameter Values** **BVPS-2 Steam Generator Tube Rupture**	
Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	368,000 lbm
Break Flow to Ruptured Steam Generator	9200 lbm (0 to 116 sec) 197,400 lbm (116 to 4076 sec)
Time of Reactor Trip	116 sec
Termination of Break Flow to Ruptured SG	4076 sec
Amount of Break Flow that Flashes	1730.2 lbm (0 to 116 sec) 6814.5 lbm (116 to 1932.5 sec)
Leakage Rate to Intact Steam Generators	150 gpd at STP for each SG
Failed/Melted Fuel Percentage	0%
Reactor Coolant Tech Spec Iodine & NG Activity Concentration	Table 4.2-1A (0.35 µCi/gm DE I-131)
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2A (0.35 µCi/gm DE I-131)
Pre-Accident Iodine Spike Activity	Table 4.2-2A (21 µCi/gm DE I-131)
Concurrent Iodine Spike Appearance Rate	335 times equilibrium appearance rates
Duration of Concurrent Iodine Spike	4 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	95,150 lbm
Ruptured SG Liquid Mass (min)	95,150 lbm
Initial Mass in Steam Generators	95,150 lbm
Secondary Coolant Tech Spec Iodine Activity Concentration	Table 4.2-1A (0.10 µCi/gm DE I-131)
Form of All Iodine Released to the Environment via Steam Generators	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion))	1.0 (released without holdup)
Fraction of Noble Gas Released from any SG	1.0 (released without holdup)
Partition Factor in Condenser	100 elemental iodine 1 organic Iodine/Noble Gases
Steam Flowrate to Condenser before Reactor Trip	142,300 lbm (Ruptured SG) 281,900 lbm (Intact SGs)
Ruptured SG Steam Releases via MSSVs/ADVs	67,300 lbm (116 to 4076 sec) 0 lbm (4076 to 7200 sec) 46,800 lbm (7200 to 28800 sec)
Intact SG Steam Releases via MSSVs/ADVs	163,500 lbm (116 to 4076 sec) 216,800 lbm (4076 to 7200 sec) 798,500 lbm (7200 to 28800 sec)
Termination of Release from SGs	8 hours
Environmental Release Points	Condenser Air Ejector (0 to 116 sec) MSSVs/ADVs (116 sec to 8 hr)
CREVS Initiation Signal Timing	
Control Room ventilation is maintained in Normal mode	
CR Purge Initiation (Manual) Time and Rate	8 hours after DBA 30 minutes at 16200 cfm (minimum)

Table 7.5-2A Control Room Limiting Atmospheric Dispersion Factors (sec/m^3) BVPS-1 Steam Generator Tube Rupture					
Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
<u>CR Bounding Dispersion Factors</u>					
BVPS-1 Condenser Air Ejector (TB SE Corner)	1.05E-02	------	------	------	------
BVPS-1 MSSVs/ADVs	1.24E-03	9.94E-04	4.08E-04	------	------

<u>Note</u>:
CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake

Table 7.5-2B Control Room Limiting Atmospheric Dispersion Factors (sec/m^3) BVPS-2 Steam Generator Tube Rupture					
Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
<u>CR Bounding Dispersion Factors</u>					
BVPS-2 Condenser Air Ejector (TB NW Corner)	1.03E-03	------	------	------	------
BVPS-2 MSSVs/ADVs	5.01E-04	3.58E-04	------	------	------

<u>Note</u>:
CR X/Q values – Bounding values based on BVPS-2 release points and the BVPS-2 CR Intake

Table 7.6-1 Analysis Assumptions & Key Parameter Values BVPS-1 and BVPS-2 Locked Rotor Accident(1)	
Core Power Level Minimum Reactor Coolant Mass Primary-to-Secondary SG tube leakage Failed Fuel Percentage Melted Fuel Percentage Radial Peaking Factor Fraction of Core Inventory in Fuel Gap Core Activity of Isotopes in Gap Iodine Chemical Form in Gap	2918 MWt 341,331 lbm 450 gpd at STP < 20% 0% 1.70 I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%) Table 4.3-1 4.85% elemental 95% CsI 0.15% organic
Secondary Side Parameters Minimum Post-Accident SG Liquid Mass Iodine Species released to Environment Iodine Partition Coefficient in SGs Particulate Carry-Over Fraction in SGs Steam Releases from SGs Termination of releases from SGs Fraction of Noble Gas released to Environment Environmental Release Point	 101,799 lbm per SG 97% elemental; 3% organic 100 (all tubes submerged) 0.0025 348,000 lbm (0 to 2 hr) 778,000 lbm (2 to 8 hr) 8 hours 1.0 (released without holdup) MSSVs/ADVs
CREVS Initiation Signal Timing Control Room ventilation is maintained in Normal mode	
Note: (1) Bounding parameter values are used to encompass an event at either unit.	

Table 7.6-2
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)
BVPS-1 & BVPS-2 Locked Rotor Accident

Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
CR Bounding Dispersion Factors					
Secondary System leakage	1.24E-03	9.94E-04	------	------	------

Note:
CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake

Table 7.6-3 Analysis Assumptions & Key Parameter Values BVPS-1 and BVPS-2 Loss of AC Power(1)	
Core Power Level Minimum Reactor Coolant Mass Primary-to-Secondary SG tube leakage Failed Fuel Percentage Melted Fuel Percentage Reactor Coolant Tech Spec Iodine & NG Activity Concentration	2918 MWt 341,331 lbm 450 gpd at STP 0% 0% Table 4.2-1A (0.35 µCi/gm DE I-131)
Secondary Side Parameters Minimum Post-Accident SG Liquid Mass Iodine Species released to Environment Secondary Coolant Tech Spec Iodine Activity Concentration Iodine Partition Coefficient in SGs Fraction of Noble Gas Released from SGs Steam Releases from SGs Termination of releases from SGs Environmental Release Point	 101,799 lbm per SG 97% elemental; 3% organic Table 4.2-1A (0.10 µCi/gm DE I-131) 100 (all tubes submerged) 1.0 (released without holdup) 348,000 lbm (0 to 2 hr) 778,000 lbm (2 to 8 hr) 8 hours MSSVs/ADVs
CREVS Initiation Signal Timing Control Room ventilation is maintained in Normal mode	
Note: (1) Bounding parameter values are used to encompass an event at either unit.	

Table 7.6-4
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)
BVPS-1 & BVPS-2 Loss of AC Power

Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
<u>CR Bounding Dispersion Factors</u>					
Secondary System leakage	1.24E-03	9.94E-04	------	------	------

<u>Note</u>:
CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake.

Table 7.7-1
Analysis Assumptions & Key Parameter Values
BVPS-1 and BVPS-2 Fuel Handling Accident in Fuel Pool Area or Containment(1)

Core Power Level	2918 MWt
Number of Rods in Fuel Assemblies	264
Total Number of Fuel Assemblies	157
Number of Damaged Rods	137
Decay Time Prior to Fuel Movement	100 hours
Radial Peaking Factor	1.70
Fraction of Core Inventory in Fuel Gap	I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%)
Core Activity of Isotopes in Gap with 100 hrs decay	Table 7.7-2
Iodine Form of gap release before scrubbing	99.85% elemental 0.15% organic
Min depth of water in Fuel Pool or Reactor Cavity	23 ft
Scrubbing Decontamination Factors	Iodine (200) Noble Gas (1) Particulates (∞)
Rate of Release from Fuel	PUFF
Environmental Release Rate (unfiltered) within a 2 hour period	All airborne activity
<u>Environmental Release Points</u>	
Accident in Fuel Pool Area	More Restrictive of Ventilation Vent or SLCRS Vent (Containment Top)
Accident in Containment	More Restrictive of Equipment Hatch, Ventilation Vent, Containment Wall or SLCRS Vent (Containment Top)
CREVS Initiation Signal Timing	
BVPS-1 and BVPS-2 Control Room ventilation is maintained in Normal mode	
BVPS-1 Control Room purge initiation (Manual) Time and Rate	2 hours after DBA 30 minutes at 16200 cfm (minimum)
Note: (1) Bounding parameter values are used to encompass an event at either unit.	

Table 7.7-2
Core Activity of Isotopes in Gap with 100 hrs decay

Nuclide	Composite Core Activity (Ci)
KR-85	8.27E+05
KR-85M	3.77E+00
XE-127	9.50E+00
XE-129M	4.49E+03
XE-131M	1.00E+06
XE-133	1.11E+08
XE-133M	2.07E+06
XE-135	2.13E+05
XE-135M	6.51E+02
BR- 82	4.25E+04
I-129	2.86E+00
I-130	7.64E+03
I-131	5.62E+07
I-132	4.74E+07
I-133	5.86E+06
I-135	3.98E+03

Table 7.7-3
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)
Fuel Handling Accident

Release Location / Receptor	0 to 2 hr	2 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
CR Bounding Dispersion Factors (BVPS-1)					
BVPS-1 Ventilation Vent	4.75E-03	3.66E-03	1.43E-03	1.02E-03	8.84E-04
CR Bounding Dispersion Factors (BVPS-2)					
BVPS-2 Ventilation Vent	9.39E-04	6.69E-04	3.08E-04	2.23E-04	1.54E-04

Notes:

(a) *BVPS-1 FHA: the bounding atmospheric dispersion factors for the FHA in the Fuel Pool Area or Containment is from the BVPS-1 ventilation vent to the BVPS-1 CR intake.*

(b) *BVPS-2 FHA: the bounding atmospheric dispersion factors for the FHA in the Fuel Pool Area or Containment is from the BVPS-2 ventilation vent to the BVPS-2 CR intake.*

Table 7.8-1 Analysis Assumptions & Key Parameter Values BVPS-1 and BVPS-2 Small Line Break Outside Containment[(1)]	
Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	341,331 lbm
CVCS letdown line break - mass flow rate	16.79 lbm/s
Break Flow Flash Fraction	37%
Time to isolate break	15 minutes
Failed Fuel Percentage	0%
Melted Fuel Percentage	0%
Reactor Coolant Tech Spec Iodine & NG Activity Concentration	Table 4.2-1A (0.35 µCi/gm DE I-131)
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2A (0.35 µCi/gm DE I-131)
Concurrent Iodine Spike Appearance Rate	500 times equilibrium appearance rates
Duration of Concurrent Iodine Spike	4 hours
Iodine Species released to Environment	97% elemental; 3% organic
SLCRS Filter Efficiency	0%
Environmental Release Point	Ventilation Vent
CREVS Initiation Signal Timing Control Room ventilation is maintained in Normal mode	
Note: (1) Bounding parameter values are used to encompass an event at either unit.	

Table 7.8-2
Small Line Break Outside Containment
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	**0 to 2 hr**	**2 to 8 hr**	**8 to 24 hr**	**24 to 96 hr**	**96 to 720 hr**
CR Bounding Dispersion Factors					
BVPS-1 Ventilation Vent	4.75E-03				

Note:
CR X/Q values – Bounding values based on BVPS-1 release points and the BVPS-1 CR Intake.

8.0 SUMMARY OF RESULTS: CONTROL ROOM/SITE BOUNDARY DOSES

The BVPS licensing basis accidents listed below have been analyzed for dose consequences at the site boundary and control room.

1. Loss of Coolant Accident (LOCA)
2. Control Rod Ejection Accident (CREA)
3. Main Steam Line Break (MSLB) outside Containment
4. Steam Generator Tube Rupture (SGTR)
5. Locked Rotor Accident (LRA)
6. Loss of AC Power (LACP)
7. Fuel Handling Accident (FHA) in the Fuel Pool or in Containment
8. Small Line Break Outside Containment (SLBOC)

From the results in Section 8.0, the LOCA is the limiting accident for the Exclusion Area Boundary, Low Population Zone, and Control Room doses. Therefore, it is the only accident considered for determining the limiting TSC occupant dose. This is consistent with the current BVPS licensing basis.

In accordance with RG 1.183, the “worst 2 hour period” dose at the EAB and the dose at the LPZ “for the duration of the release” are presented in Table 8.1-1. These dose values represent the post-accident dose to the public due to inhalation and submersion for each of these events. Due to distance and plant shielding, the dose contribution at the EAB/LPZ attributed to direct shine from contained sources is considered negligible for all the accidents. The associated regulatory limit as discussed in Section 2.4 is also presented.

Per regulatory guidance, the CR dose is integrated over 30 days. The calculated doses address the fact that for events with a duration less than 30 days, the CR dose needs to include the remnant radioactivity within the CR envelope after the event has been terminated. The 30 day integrated dose to the control room operator, due to inhalation and submersion, is presented in Table 8.1-2 for all of the referenced design basis accidents. No credit is taken for use of personal protective equipment or prophylactic drugs, such as potassium iodide pills.

The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The dose contribution due to direct shine from post-LOCA contained sources/external cloud is identified and included in the CR doses reported for the LOCA in Table 8.1-2.

In accordance with the current BVPS licensing basis, the ERF/TSC design has been evaluated for the LOCA. The 30 day integrated dose to the occupant in the ERF (bounds the TSC) due to inhalation, submersion, and direct shine from the post-LOCA contained sources/external cloud is 4.02 rem TEDE (note: the dose contribution of direct shine to this total is ~0.78 rem TEDE) without crediting the ventilation or filtration systems for dose reduction via operation of the ventilation equipment in the ERF, and assuming bounding values for filter efficiency when estimating the direct shine dose from the ERF normal operation intake filters, and the ERF emergency recirculation filters.

Table 8.1-1
Beaver Valley Power Station BVPS-1 and BVPS-2
Exclusion Area Boundary and Low Population Doses (TEDE)

Accident	EAB Dose (rem)(1,3)	LPZ Dose (rem)(2)	Regulatory Limit (rem)
Loss of Coolant Accident	16.62	2.9	25
Control Rod Ejection Accident(4)	3	1.4	6.3
Main Steam Line Break (BVPS-1)(7)	0.11 0.14	0.02 0.04	25 (PIS) 2.5 (CIS)
Main Steam Line Break (BVPS-2)(5)(7)	0.4 2.5	0.1 0.7	25 (PIS) 2.5 (CIS)
Steam Generator Tube Rupture (BVPS-1)(7)	2.3 0.88	0.14 0.06	25 (PIS) 2.5 (CIS)
Steam Generator Tube Rupture (BVPS-2)(7)	1.3 0.64	0.08 0.05	25 (PIS) 2.5 (CIS)
Locked Rotor Accident	2.3	0.35	2.5
Loss of AC Power	(Note 6)	(Note 6)	2.5
Fuel Handling Accident BVPS-1 BVPS-2	 2.1 2.5	 0.12 0.12	6.3
Small Line Break Outside Containment	0.22	0.011	2.5

Notes:
(1) EAB Doses are based on the worst 2 hour period following the onset of the event.
(2) LPZ Doses are based on the duration of the release.
(3) Except as noted, the maximum 2 hr dose for the EAB is based on the 0 to 2 hr period:
- LOCA: 0.5 to 2.5 hr
- MSLB (CIS): 4 to 6 hr (BVPS-2 only)
- LRA: 6 to 8 hr

(4) Doses are based on the containment release scenario. The dose consequences based on the secondary side release scenario are 1 Rem (EAB) and 0.1 Rem (LPZ).
(5) Doses are based on the maximum allowable Accident Induced Leakage (8.1 gpm) into the affected SG. LAR dose is ≤ Regulatory Limit.
(6) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.
(7) PIS: Pre-accident iodine spike; CIS: Concurrent iodine spike.

Table 8.1-2
30 Day Integrated Control Room Doses (TEDE)

Accident	Control Room Operator	
	Dose (rem)	Reg. Limit (rem)
Loss of Coolant Accident(1)	4.5 (0.61)	5
Control Rod Ejection Accident(2)	3.1	5
Main Steam Line Break (BVPS-1)(5)	1.7	5
Main Steam Line Break (BVPS-2)(3)(5)	1.5	5
Steam Generator Tube Rupture (BVPS-1)(5)(6)	2.6	5
Steam Generator Tube Rupture (BVPS-2)(5)(6)	0.4	5
Locked Rotor Accident(6)	2.9	5
Loss of AC Power(6)	(Note 4)	5
Fuel Handling Accident(6) BVPS-1(5) BVPS-2	 4.2 1.4	5
Small Line Break Outside Containment(6)	0.7	5

Notes:
(1) Portion shown in parentheses for the LOCA represents that portion of the TEDE dose that is the contribution of direct shine from contained sources/external cloud.
(2) Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario is 0.2 Rem.
(3) Dose is based on the maximum allowable Accident Induced Leakage (8.1 gpm) into the affected SG.
(4) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.
(5) The CR is purged for 30 minutes at a flow rate of 16200 cfm (minimum) following termination of the environmental releases (or significant reduction in releases as in the case of the BVPS-1 SGTR), and by:
- MSLB: Purge within 24 hrs
- SGTR: Purge within 8 hrs
- FHA (BVPS-1): Purge at 2 hrs

(6) The following accidents do not take credit for CREVS operation: SGTR, LRA, LACP, FHA, and SLBOC.

9.0 CONCLUSIONS

It is concluded that the proposed combination of changes to methodology and operational limits incorporated in this assessment to support the operational flexibility sought by FENOC at BVPS are acceptable in that the dose consequences at the site boundary and control room following all BVPS design basis accidents remain within the regulatory requirements of 10 CFR 50.67.

In summary, based on the proposed changes outlined in Section 2 and the conclusions of this assessment, FENOC is requesting NRC approval to:

1. Increase the allowable unfiltered air inleakage into the BVPS CRE during the listed modes of operation of the CR ventilation system as noted below:

    - CR Ventilation Isolation mode – Increase the allowable unfiltered air inleakage from the current maximum value of 300 cfm to 450 cfm (updated value represents an upper bound analytical value which includes test measurement uncertainties and a 10 cfm allowance for ingress/egress)

    - CR Ventilation Emergency mode – Increase the allowable unfiltered air inleakage from the current maximum value of 30 cfm to 165 cfm (the updated value represents an upper bound analytical value which includes test measurement uncertainties and a 10 cfm allowance for ingress/egress).

2. Eliminate the commitment to Note 11 of Regulatory Guide 1.183, Revision 0, with respect to limiting the linear heat generation rate to < 6.3 kw/ft peak rod average power for burnups exceeding 54,000 MWD/MTU.

3. Increase the maximum allowed accident-induced steam generator tube leakage at BVPS-2 from 2.1 gpm to 8.1 gpm.

10.0 REFERENCES

1. Code of Federal Regulations 10 CFR 50.67, "Accident Source Term".

2. NUREG-0800, Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Revision 0.

3. NRC Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", 7/2000.

4. Beaver Valley Power Station Units 1 and 2 License Amendment Request (LAR) Nos. 300 and 172, letter L-02-069, Subject: Operation with Atmospheric Containments, 6/5/2002 [ADAMS Accession ML021620298].

5. NRC Safety Evaluation Report issued for Amendment No. 257 (BVPS-1) and No. 139 (BVPS-2), "Selective Implementation of Alternate Source Term and Control Room Habitability Technical Specification Changes," 9/10/2003 [ADAMS Accession ML032530204].

6. Beaver Valley Power Station Units 1 and 2 License Amendment Request (LAR) Nos. 302 and 173, letter L-04-125, Subject: Extended Power Uprate, 10/4/2004 [ADAMS Accession ML042920300].

7. NRC Safety Evaluation Report issued for Amendment No. 275 (BVPS-1) and No. 156 (BVPS-2), Subject: Extended Power Uprate, 7/19/2006 [ADAMS Accession ML061720248].

8. Beaver Valley Power Station Units 1 and 2 License Amendment Request (LAR) Nos. 334 and 205, letter L-07-017, Subject: Recirculation Spray System Pump Start Signal, Modular Accident Analysis Program – Design Basis Accident, and Aerosol Removal Coefficients Calculation Methodology, 2/9/2007 [ADAMS Accession ML070440341].

9. NRC Safety Evaluation Report issued for Amendment No. 280 (BVPS-1), "Changes to the Recirculation Spray System Pump Start Signal due to the Containment Sump Screen Modification," 10/5/2007 [ADAMS Accession ML072680397].

10. NRC Safety Evaluation Report issued for Amendment No. 164 (BVPS-2), "Changes to the Recirculation Spray System Pump Start Signal Due to the Containment Sump Screen Modification," 3/11/ 2008 [ADAMS Accession ML080420549].

11. NRC Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," 10/2009.

12. NRC Safety Evaluation Report issued for Amendment No. 230 (DCPP-1) and No. 232 (DCPP-2), "Diablo Canyon Power Plant Units 1 & 2 – Issuance of Amendments Re: Revise Licensing Bases to Adopt Alternate Source Terms," 4/27/2017 [ADAMS Accession ML17012A246].

13. Westinghouse Nuclear Safety Advisory Letter NSAL-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues", 7/25/2011.

14. NUREG-0737 Supplement No. 1, Clarification of TMI Action Plan Requirements, 1/1983.

15. NRC Safety Evaluation Report issued for Amendment No. 273 (BVPS-1), "Steam Generator (SG) Replacement," 2/9/2006 [ADAMS Accession ML060240146].

16. International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers", Supplement to Part 1 (pages 192 to 212), 1979.

17. NUREG 76-6521, "Radiation Signature Following the Hypothesized LOCA," Sandia Laboratories (SAND76-0740), 9/1977.

18. DOE/TIC-11026, "Radioactive Decay Data Tables - A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments," Kocher, David C., 1981.

19. Lawrence Berkeley Laboratory, University of California, Berkeley, "Table of Isotopes," Seventh Edition.

20. Industry Computer Code ORIGEN2, "Isotopic Generation and Depletion Code-Matrix Exponential Method," developed by Oak Ridge National Laboratory, 9/1983.

21. EPA-520/1-88-020, Federal Guidance Report No. 11, 9/1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion".

22. ANSI/ANS 6.1.1-1991, "Neutron and Gamma-Ray Fluence-to-Dose Factors".

23. TID-24190, Air Resources Laboratories, "Meteorology and Atomic Energy", 7/1968.

24. ANSI/ANS 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors".

25. Beaver Valley Power Station Units 1 and 2, Beaver Valley Power Station Units 1 and 2 License Amendment Request (LAR) Nos. 300 and 172, letter L-03-135, "Withdrawal of the Containment Conversion Portion of License Amendment Request Nos. 300 and 172," 9/5/2003 [ADAMS Accession ML032541026].

26. Beaver Valley Power Station Units 1 and 2 License Amendment Request No. 10-021, letter L-11-141, "Replacement of BVPS-1 Spray Additive System by Containment Sump pH Control System," 5/27/2011 [ADAMS Accession ML111510646].

27. Beaver Valley Power Station Units 1 and 2 License Amendment Request No. 08-006, letter L-08-236, "Replacement of BVPS-2 Spray Additive System by Containment Sump pH Control System," 9/24/2008 [ADAMS Accession ML082730716].

28. NRC Safety Evaluation Report issued for Amendment No. 289 (BVPS-1), "Spray Additive System by Containment Sump pH Control," 3/14/2012 [ADAMS Accession ML120530591].

29. NRC Safety Evaluation Report issued for Amendment No. 168 (BVPS-2), “Spray Additive System by Containment Sump pH Control,” 4/16/2009 [ADAMS Accession ML090780352].

30. Lischer, D.J., User Manual, Aerosol Behavior in a Condensing Atmosphere (SWNAUA), 6/1993 (Proprietary).

31. Bunz, H., Koyro, M., Schöck, W., 1982, NAUA/Mod4 - A Code for Calculating Aerosol Behaviour in LWR Core Melt Accidents, Code Description and Users Manual, KfK.

32. NUREG-0800, Standard Review Plan, Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 2.

33. NUREG/CR-5732, “Iodine Chemical Forms in LWR Severe Accidents – Final Report,” 4/1992.

34. NRC Information Notice 93-17, Revision 1, “Safety Systems Response to Loss of Coolant and Loss of Offsite Power,” 3/25/1994 (original issue 3/8/1993).

35. NRC Generic Letter 95-05, “Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,” 8/3/1995.

36. NUREG-0800, Standard Review Plan, Section 15.2.6, “Loss of Non-Emergency AC Power to Station Auxiliaries”, Revision 1.

37. NUREG-0800, Standard Review Plan, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment", Revision 2.

38. Appendix E to 10 CFR Part 50, Emergency Planning and Preparedness for Production and Utilization Facilities.

39. Millstone, Unit 2, “Supplemental Information - Selective Implementation of the Alternative Source Term Fuel Handling Accident Analyses”, 5/7/2004 [ADAMS Accession ML041320350].

40. Beaver Valley Power Station Units 1 and 2 License Amendment Request (LAR) Nos. 317 and 190, letter L-04-073, Subject: Operation with an Atmospheric Containment Design,” 6/2/2004 [ADAMS Accession ML041560461].

41. NRC Safety Evaluation Report issued for Amendment No. 271 (BVPS-1) and No. 153 (BVPS-2), “Containment Conversion from Subatmospheric to Atmospheric Operating Conditions,” 2/6/2006 [ADAMS Accession ML060100325].

42. Beaver Valley Power Station Unit 1 License Amendment Request (LAR) No. 320, letter L-05-069, Subject: Replacement Steam Generators, 4/13/2005 [ADAMS Accession ML051080573].

APPENDIX A: CHANGES TO KEY DESIGN INPUT VALUES (ARRANGED BY ACCIDENT): CLB VS THIS LICENSE AMENDMENT REQUEST (LAR)

As noted in Section 1.0, as part of a long term objective, and with the intent of relaxing certain operational limits that have significant effect on plant operations, FENOC is updating the site boundary and control room dose consequence analyses associated with the following BVPS design basis accidents that represent its licensing basis.

1. Loss of Coolant Accident (LOCA)
2. Control Rod Ejection Accident (CREA)
3. Main Steam Line Break (MSLB) outside Containment
4. Steam Generator Tube Rupture (SGTR)
5. Locked Rotor Accident (LRA)
6. Loss of AC Power (LACP)
7. Fuel Handling Accident (FHA) in the Fuel Pool or in Containment
8. Small Line Break Outside Containment (SLBOC)

Appendix A provides a comparison between the design input values used in the current BVPS licensing basis (CLB) dose consequence analyses supporting BVPS-1 and BVPS-2, to those utilized in the dose consequence analyses supporting this License Amendment Request (LAR). The information is arranged by accident and in tabular format. Only those design parameters that have undergone a change are listed.

The control room parameter values, which are generally applicable to all accidents, are summarized separately. As noted in the main document, the core inventory and the atmospheric dispersion factors remain unchanged by this application.

Table No.	Subject
Table A-1	Non-LOCA Gap Fractions
Table A-2a	BVPS-1 Reactor Coolant Tech Spec Iodine & Noble Gas Activity Concentrations
Table A-2b	BVPS-2 Reactor Coolant Tech Spec Iodine & Noble Gas Activity Concentrations
Table A-3a	BVPS-1 Reactor Coolant Tech Spec Equilibrium Iodine Appearance Rates
Table A-3b	BVPS-2 Reactor Coolant Tech Spec Equilibrium Iodine Appearance Rates
Table A-4a	BVPS-1 Secondary Coolant Tech Spec Iodine Activity Concentrations
Table A-4b	BVPS-2 Secondary Coolant Tech Spec Iodine Activity Concentrations
Table A-5a	Control Room (CR)
Table A-5b	Emergency Response Facility (ERF) / Technical Support Center (TSC)
Table A-6	BVPS-1 & BVPS-2 Loss-of-Coolant Accident (LOCA)
Table A-7	BVPS-1 & BVPS-2 Control Rod Ejection Accident (CREA)
Table A-8a	BVPS-1 Main Steam Line Break (MSLB)
Table A-8b	BVPS-2 Main Steam Line Break (MSLB)
Table A-9a	BVPS-1 Steam Generator Tube Rupture (SGTR)
Table A-9b	BVPS-2 Steam Generator Tube Rupture (SGTR)
Table A-10	BVPS-1 & BVPS-2 Locked Rotor Accident (LRA)
Table A-11	BVPS-1 & BVPS-2 Loss of AC Power (LACP)
Table A-12	BVPS-1 & BVPS-2 Fuel Handling Accident (FHA)
Table A-13	BVPS-1 & BVPS-2 Small Line Break Outside Containment (SLBOC)
Table A-14a	Exclusion Area Boundary (EAB) Dose
Table A-14b	Low Population Zone (LPZ) Dose
Table A-14c	Control Room Operator Dose

Table A-1
Changes to Gap Fraction for Non-LOCA Events

Item	LAR(2)	CLB(1)
I-131	0.08	0.08
I-132	0.23	0.05
KR-85	0.35	0.10
Other Noble Gases	0.04	0.05
Other Halogens	0.05	0.05
Alkali Metals	0.46	0.12

Notes:
(1) CLB values are based on RG 1.183 gap fractions
(2) LAR values are based on proposed change to DG-1199 gap fractions

Table A-2a BVPS-1 Reactor Coolant Technical Specification Iodine and Noble Gas Activity Concentrations (µCi/gm)		
Item	**LAR[(1)] 0.35 µCi/gm DE I-131**	**CLB 0.35 µCi/gm DE I-131**
I-131	2.73E-01	2.74E-01
I-132	1.13E-01	1.08E-01
I-133	4.17E-01	4.10E-01
I-134	6.47E-02	6.00E-02
I-135	2.46E-01	2.36E-01
Kr-83m	4.09E-02	3.89E-02
Kr-85m	1.48E-01	1.35E-01
Kr-85	1.30E+01	1.18E+01
Kr-87	9.68E-02	9.00E-02
Kr-88	2.74E-01	2.52E-01
Xe-131m	5.54E-01	4.84E-01
Xe-133m	4.59E-01	3.99E-01
Xe-133	3.34E+01	2.95E+01
Xe-135m	9.87E-02	9.09E-02
Xe-135	1.02E+00	9.16E-01

Note:
(1) BVPS-1 will continue to use the current BVPS TS LCO 3.4.16 limit of 0.35 µCi/gm DE I-131. The minor changes in the isotopic concentrations used for the LAR reflect updated design input parameter values used to develop the updated design (1% fuel defects) reactor coolant concentrations.

Table A-2b BVPS-2 Reactor Coolant Technical Specification Iodine and Noble Gas Activity Concentrations (µCi/gm)		
Item	**LAR[1] 0.10 µCi/gm DE I-131**	**CLB 0.35 µCi/gm DE I-131**
I-131	7.79E-02	2.74E-01
I-132	3.22E-02	1.08E-01
I-133	1.19E-01	4.10E-01
I-134	1.85E-02	6.00E-02
I-135	7.04E-02	2.36E-01
Kr-83m	1.17E-02	3.89E-02
Kr-85m	4.22E-02	1.35E-01
Kr-85	3.71E+00	1.18E+01
Kr-87	2.77E-02	9.00E-02
Kr-88	7.85E-02	2.52E-01
Xe-131m	1.58E-01	4.84E-01
Xe-133m	1.31E-01	3.99E-01
Xe-133	9.55E+00	2.95E+01
Xe-135m	2.82E-02	9.09E-02
Xe-135	2.93E-01	9.16E-01

Note:
(1) FENOC is proposing to lower the TS LCO 3.4.16 limit to 0.10 µCi/gm DE I-131 for BVPS-2.

Table A-3a
BVPS-1 Reactor Coolant
Technical Specification Equilibrium Iodine Appearance Rates (μCi/sec)

Item	LAR(1) 0.35 μCi/gm DE I-131	CLB 0.35 μCi/gm DE I-131
I-131	2.27E+03	2.53E+03
I-132	2.83E+03	2.66E+03
I-133	4.17E+03	4.42E+03
I-134	3.39E+03	3.00E+03
I-135	3.44E+03	3.41E+03

Note:
(1) BVPS-1 will continue to use the current BVPS TS LCO 3.4.16 limit of 0.35 μCi/gm DE I-131. The minor changes in the iodine appearance rates used for the LAR reflect updated design input parameter values used to develop the updated design (1% fuel defects) reactor coolant concentrations.

Table A-3b
BVPS-2 Reactor Coolant
Technical Specification Equilibrium Iodine Appearance Rates (μCi/sec)

Item	LAR(1) 0.10 μCi/gm DE I-131	CLB 0.35 μCi/gm DE I-131
I-131	6.48E+02	2.53E+03
I-132	8.06E+02	2.66E+03
I-133	1.19E+03	4.42E+03
I-134	9.71E+02	3.00E+03
I-135	9.86E+02	3.41E+03

Note:
(1) FENOC is proposing to lower the TS LCO 3.4.16 limit to 0.10 μCi/gm DE I-131 for BVPS-2.

Table A-4a BVPS-1 Secondary Coolant Technical Specification Iodine Activity Concentrations (μCi/gm)		
Item	**LAR(1) 0.10 μCi/gm DE I-131**	**CLB 0.10 μCi/gm DE I-131**
I-131	8.34E-02	8.33E-02
I-132	1.39E-02	1.40E-02
I-133	9.32E-02	9.39E-02
I-134	1.90E-03	1.95E-03
I-135	3.34E-02	3.39E-02

Note:

(1) BVPS-1 will continue to use the current BVPS TS LCO 3.7.13 limit of 0.10 μCi/gm DE I-131. The minor changes in the iodine appearance rates used for the LAR reflect updated design input parameter values used to develop the updated design secondary coolant concentrations

Table A-4b BVPS-2 Secondary Coolant Technical Specification Iodine Activity Concentrations (μCi/gm)		
Item	**LAR(1) 0.05 μCi/gm DE I-131**	**CLB 0.10 μCi/gm DE I-131**
I-131	4.17E-02	8.33E-02
I-132	6.93E-03	1.40E-02
I-133	4.66E-02	9.39E-02
I-134	9.52E-04	1.95E-03
I-135	1.67E-02	3.39E-02

Note:

(1) FENOC is proposing to lower the TS LCO 3.7.13 limit to 0.05 μCi/gm DE I-131 for BVPS-2.

Table A-5a Analysis Assumptions & Key Parameter Values BVPS Common Control Room		
Control Room Parameters		
Item	**LAR**	**CLB**
Normal Operation Unfiltered Air Intake	< 1250 cfm	500 cfm
Isolation Mode Unfiltered Air Inleakage (includes 10 cfm for ingress/egress)	< 450 cfm	300 cfm
Emergency Mode Filtered Air Intake	800 to 1000 cfm	600 to 1030 cfm
Emergency Mode Unfiltered Air Inleakage (includes 10 cfm for ingress/egress)	< 165 cfm	30 cfm

Table A-5b Analysis Assumptions & Key Parameter Values BVPS Emergency Response Facility / Technical Support Center		
ERF Parameters		
Item	**LAR**	**CLB**
Minimum Free Volume	462,129 ft^3	478,610 ft^3
ERF Maximum Filtered Recirculation Air Flow rate	7200 cfm +/-10%	3800 cfm +/-10%
Intake Filter Efficiency (HEPA)	100% or 0% (particulate), whichever is more conservative	90%
Recirculation Filter Efficiency (HEPA)	100% or 0% (particulate) whichever is more conservative	98%
Recirculation Filter Efficiency (Charcoal)	100% or 0% (elemental/organic iodine) whichever is more conservative	90%

Table A-6 Analysis Assumptions & Key Parameter Values BVPS-1 & BVPS-2 Loss of Coolant Accident		
Item	**LAR**	**CLB**
Containment Leakage Parameters		
Sprayed Fraction	60%	63%
Spray Period	77.4 sec to 96 hours	85.4 sec to 4 days
Spray Removal Constants	Figure 7.2-1, Figure 7.2-2 and Table 7.2-2	Reference 8, Figure E-1 & Figure E-2
ECCS/RWST Leakage Parameters		
Minimum Sump Volume (5 to 30 min) (0.5 to 2 hr) (2 hr to 30 day)	 19,253 ft^3 (1.1379E6 lbm) 24,909 ft^3 (1.5133E6 lbm) 43,824 ft^3 (2.6837E6 lbm)	 19,111 ft^3 (1.13E6 lbm) 25,333 ft^3 (1.51E6 lbm) 43,577 ft^3 (2.68E6 lbm)
RWST - Onset of back-leakage	1768 seconds	1782 seconds
RWST - Onset of RWST activity venting	3039 seconds	3055 seconds
Iodine Release Fraction via RWST	Figure 7.2-3 & Table 7.2-3	Reference 8, Figure E-3 values X 2
Containment Vacuum System Parameters		
RCS TS Iodine and Noble Gas Activity Concentration	Table 4.2-1A	Table A-2a

Table A-7 Analysis Assumptions & Key Parameter Values BVPS-1 & BVPS-2 Control Rod Ejection Accident		
Item	**LAR**	**CLB**
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Peaking Factor	1.70	1.75
Steam Releases per SG 2500 sec to 8 hrs	 778,000 lbs	 776,000 lbs
Termination of Secondary System Environmental Releases	30 days	8 hours

Table A-8a Analysis Assumptions & Key Parameter Values BVPS-1 Main Steam Line Break		
Item	**LAR**	**CLB**
Minimum Reactor Coolant Mass	345,097 lbm	340,711 lbm
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Table 4.2-1A	Table A-2a
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2A	Table A-3a
Secondary Coolant TS Iodine Activity Concentration	Table 4.2-1A	Table A-4a

Table A-8b Analysis Assumptions & Key Parameter Values BVPS-2 Main Steam Line Break		
Item	**LAR**	**CLB**
Minimum Reactor Coolant Mass	341,331 lbm	341,332 lbm
Amount of Accident Induced Leakage (AIL) into Faulted SG	8.1	2.1
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Table 4.2-1B	Table A-2b
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2B	Table A-3b
Secondary Coolant TS Iodine Activity Concentration	Table 4.2-1B	Table A-4b

Table A-9a
Analysis Assumptions & Key Parameter Values
BVPS-1 Steam Generator Tube Rupture

Item	LAR	CLB
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Table 4.2-1A	Table A-2a
Reactor Coolant TS Equilibrium Iodine Appearance Rates	Table 4.2-2A	Table A-3a
Secondary Coolant TS Iodine Activity Concentration	Table 4.2-1A	Table A-4a

Table A-9b
Analysis Assumptions & Key Parameter Values
BVPS-2 Steam Generator Tube Rupture

Item	LAR	CLB
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Table 4.2-1A	Table A-2a
Reactor Coolant TS Equilibrium Iodine Appearance Rates	Table 4.2-2A	Table A-3a
Secondary Coolant TS Iodine Activity Concentration	Table 4.2-1A	Table A-4a

Table A-10
Analysis Assumptions & Key Parameter Values
BVPS-1 & BVPS-2 Locked Rotor Accident

Item	LAR	CLB
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Fraction of Core Inventory in Fuel Gap	Based on DG-1199 Table 3	Based on RG 1.183 Table 3
Radial Peaking Factor	1.70	1.75

Table A-11 Analysis Assumptions & Key Parameter Values BVPS-1 & BVPS-2 Loss of AC Power		
Item	**LAR**	**CLB**
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Table 4.2-1A	Table A-2a
Secondary Coolant TS Iodine Activity Concentration	Table 4.2-1A	Table A-4a

Table A-12 Analysis Assumptions & Key Parameter Values BVPS-1 & BVPS-2 Fuel Handling Accident		
Item	**LAR**	**CLB**
Radial Peaking Factor	1.70	1.75
Fraction of Core Inventory in Fuel Gap	Based on DG-1199 Table 3	Based on RG 1.183 Table 3

Table A-13 Analysis Assumptions & Key Parameter Values BVPS-1 & BVPS-2 Small Line Break Outside Containment		
Item	**LAR**	**CLB**
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Table 4.2-1A	Table A-2a
Reactor Coolant Equilibrium Iodine Appearance Rates	Table 4.2-2A	Table A-3a

Table A-14a
2 Hour Integrated Exclusion Area Boundary Dose (TEDE)

Accident	LAR (rem)(1,3)	CLB (rem)	Regulatory Limit (rem)
Loss of Coolant Accident	16.62	16.1	25
Control Rod Ejection Accident(4)	3	3.1	6.3
Main Steam Line Break (BVPS-1)(7)	0.11 0.14	0.08 0.11	25 (PIS) 2.5 (CIS)
Main Steam Line Break (BVPS-2)(5)(7)	0.4 2.5	0.4 2.5	25 (PIS) 2.5 (CIS)
Steam Generator Tube Rupture (BVPS-1)(7)	2.3 0.88	2.27 0.93	25 (PIS) 2.5 (CIS)
Steam Generator Tube Rupture (BVPS-2)(7)	1.3 0.64	1.3 0.68	25 (PIS) 2.5 (CIS)
Locked Rotor Accident	2.3	2	2.5
Loss of AC Power	(Note 6)	(Note 6)	2.5
Fuel Handling Accident BVPS-1 BVPS-2	 2.1 2.5	 2.02 2.43	6.3
Small Line Break Outside Containment	0.22	0.23	2.5

<u>Notes:</u>
(1) EAB Doses are based on the worst 2 hour period following the onset of the event.
(2) N/A
(3) Except as noted, the maximum 2 hr dose for the EAB is based on the 0 to 2 hr period:
- LOCA: 0.5 to 2.5 hr
- MSLB (CIS): 4 to 6 hr (BVPS-2 only)
- LRA: 6 to 8 hr

(4) Doses are based on the containment release scenario. The dose consequences based on the secondary side release scenario is 1 Rem (CLB and LAR).
(5) Doses are based on the maximum allowable Accident Induced Leakage of 2.1 gpm (for CLB) and 8.1 gpm (LAR) into the affected SG. LAR dose is <u>< Regulatory Limit.</u>
(6) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.
(7) PIS: Pre-accident iodine spike; CIS: Concurrent iodine spike.

Table A-14b
30 Day Integrated Low Population Zone Dose (TEDE)

Accident	LAR (rem)	CLB (rem)[2]	Regulatory Limit (rem)
Loss of Coolant Accident	2.9	2.9	25
Control Rod Ejection Accident[4]	1.4	1.5	6.3
Main Steam Line Break (BVPS-1)[7]	0.02 0.04	0.01 0.04	25 (PIS) 2.5 (CIS)
Main Steam Line Break (BVPS-2)[5][7]	0.1 0.7	0.1 0.7	25 (PIS) 2.5 (CIS)
Steam Generator Tube Rupture (BVPS-1)[7]	0.14 0.06	0.14 0.06	25 (PIS) 2.5 (CIS)
Steam Generator Tube Rupture (BVPS-2)[7]	0.08 0.05	0.07 0.05	25 (PIS) 2.5 (CIS)
Locked Rotor Accident	0.35	0.33	2.5
Loss of AC Power	(Note 6)	(Note 6)	2.5
Fuel Handling Accident BVPS-1 BVPS-2	 0.12 0.12	 0.12 0.12	6.3
Small Line Break Outside Containment	0.011	0.012	2.5

Notes:
(1) N/A.
(2) LPZ Doses are based on the duration of the release.
(3) N/A
(4) Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario is 0.1 Rem (CLB & LAR).
(5) Doses are based on the maximum allowable Accident Induced Leakage of 2.1 gpm (for CLB) and 8.1 gpm (LAR) into the affected SG.
(6) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.
(7) PIS: Pre-accident iodine spike; CIS: Concurrent iodine spike.

Table A-14c
30 Day Integrated Control Room Dose (TEDE)

Accident	Control Room Operator (rem)		
	LAR	CLB	Reg. Limit
Loss of Coolant Accident[(1)]	4.5 (0.61)	2.16 (0.61)	5
Control Rod Ejection Accident[(2)]	3.1	1.3	5
Main Steam Line Break (BVPS-1)[(5)]	1.7	0.66	5
Main Steam Line Break (BVPS-2)[(3)(5)]	1.5	0.6	5
Steam Generator Tube Rupture (BVPS-1)[(5)(6)]	2.6	1.96	5
Steam Generator Tube Rupture (BVPS-2)[(5)(6)]	0.4	0.32	5
Locked Rotor Accident[(6)]	2.9	2.2	5
Loss of AC Power[(6)]	(Note 4)	(Note 4)	5
Fuel Handling Accident[(6)] BVPS-1[(5)] BVPS-2	 4.2 1.4	 2.36 1.4	5
Small Line Break Outside Containment[(6)]	0.7	0.7	5

<u>Notes</u>:

(1) Portion shown in parentheses for the LOCA represents that portion of the TEDE dose that is the contribution of direct shine from contained sources/external cloud.

(2) Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario is 0.06 rem (CLB) and 0.2 Rem (LAR).

(3) Dose is based on the maximum allowable Accident Induced Leakage of 2.1 gpm (for CLB) and 8.1 gpm (LAR) into the affected SG.

(4) Dose from a postulated Loss of AC Power is bounded by the Locked Rotor Accident.

(5) The CR is purged for 30 minutes at a flow rate of 16200 cfm (minimum) following termination of the environmental releases (or significant reduction in releases as in the case of the BVPS-1 SGTR), and by:
- MSLB: Purge within 24 hrs
- SGTR: Purge within 8 hrs
- FHA (BVPS-1): Purge at 2 hrs

(6) The following accidents do not take credit for CREVS operation: SGTR, LRA, LACP, FHA, and SLBOC.

Owner Acceptance Review Comment Resolution Form

BV Document ID	2710.230-000-004	Unit	BV1/2	Rev.	0	Add.	N/A
Vendor	WECTEC	Document No.	101173-RADR-002-00	Rev.	0B	Add.	N/A
Document Title	Technical Report - Impact on Dose Consequences - Proposed Changes to Promote Operational Flexibility – Beaver Valley Power Station						

FENOC Reviewer(s): __D.T.Bloom/M.G.Unfried/M.S.Ressler______________ / __3/28/2019__
Date

ODO Reviewer (print/sign): __Sreela Ferguson__ ______ / ___3/29/2019_______
Date

Item	*Comment*	*Resolution*
1	Section 1.0: Has NRC Regulatory Issue Summary 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests", been reviewed to ensure there are no shortcomings in this technical report?	Comment is a question. No change to text The updated dose consequence analyses reflect the guidance provided in RIS 2001-19. <u>Note:</u> Item 3e of RIS 2001-19 indicates that parameters based on results of surveillance testing such as testing of *charcoal filters* should address *degradation* that may occur between periodic tests. Incorporation of the above guidance is achieved by following the guidance of GL 99-02 in the dose consequence analyses with respect to the value used for filter efficiency. The GL 99-02 methodology incorporates a factor of 2 margin between the filter efficiency test acceptance criteria in the Tech Specs, and that used in the dose consequence analysis.
2	Section 2.0: Per NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", licensees who wish to take credit for reduced reactor coolant system iodine activities (below 0.35 microcuries per gram dose equivalent I-131) in the radiological dose calculation should provide a justification supporting the request that evaluates the release rate data described in Reference 6 (J. P. Adams and	Incorporated as noted. *This outstanding FENOC evaluation has been identified as Open Item 1 and is to be completed prior to submitting the license amendment request.*

	C. L. Atwood, "The Iodine Spike Release Rate During a Steam Generator Tube Rupture", Nuclear Technology, Vol. 94, p. 361 (1991). This evaluation should be listed as a FENOC action to complete.	
3	Section 2.0: It is noted that the proposed reduction in FQ from 1.75 to 1.70 will not be a challenge because the radial peaking factor is currently designed to 1.62 per the Core Operating Limits Reports in BV1 Licensing Requirements Manual Rev 102 and BV2 LRM Rev 94. Please update Section 2.3, item 2 to include verification that the radial peaking factor is less than or equal to the lowered value of 1.70 as part of the update of the BVPS core design process.	Incorporated as noted. This action has been included as a confirmatory action in Section 2.3 following receipt of NRC approval of the license amendment request.
4	Table 4.3-1: BV1/2 calculation UR(B)-484 lists Strontium as being in the core inventory, but this table does not include Sr in the gap. Sr, which is chemically similar to Barium and has substantial core activity, is not included in the gap activity.	No change to text: Table 4.3-1 presents the *isotopic gap activity* in the equilibrium core immediately after shutdown. For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are prescribed by Table 3 of NRC Regulatory Guide 1.183. These include noble gases, halogens, and alkali metals (does not include Ba and Sr). The only reason Ba-137m is listed in Table 4.3-1 is because it is in secular equilibrium with Cs-137; i.e., Cs-137 cannot exist without Ba-137m.
5	Section 7.3: Table 6 of NRC Regulatory Guide 1.183 assumes the Analysis Release Duration for a PWR Rod Ejection Accident is "until cold shutdown is established for secondary pathway"; however, for safe shutdown, BV1 is licensed as a Hot Standby plant and BV2 is licensed as an Approach to Cold Shutdown plant, so the Residual Heat Release Systems may not be available following a Control Rod Ejection Accident where Containment Isolation Phase B is initiated.	Incorporated. Generic text in the first paragraph of Section 7.3 describing the CREA has been updated to reflect the site specific assumption with respect to RHR availability as noted below: "Since the RHR pumps are located inside containment and are not qualified for Containment Isolation Phase B conditions which are expected to occur following a CREA, no credit is taken for initiation of shutdown cooling and environmental releases via the MSSVs/ADVs are assumed to occur for 30 days."
6	Section 7.4: For safe shutdown, BV1 is licensed as a Hot Standby plant and BV2 is licensed as an Approach to Cold Shutdown plant. The Residual Heat Removal Systems, which are used to achieve cold shutdown, are safety related; however, the inside containment RHR pump motors and RHR valve motor operators are not environmentally qualified for a Loss of Coolant Accident (specifically spray).	Incorporated as noted. *Confirmation of the assumption that a MSLB outside containment would result in dose consequences that are higher than those of a MSLB inside containment has been identified as Open Item 2 and is to be completed prior to submitting the license amendment request.*

	Historically, the RHR system has been credited for shutdown cooling in the BVPS dose consequence analyses of record. Section 7.4 addresses a Main Steam Line Break outside containment. A statement is made that the dose consequences of a MSLB inside containment are not analyzed since they would clearly be bounded by the dose consequences of a MSLB outside containment. Based on engineering judgment, this statement is true if RHR is available for shutdown cooling. However, because the RHR Systems may not be available to achieve cold shutdown following an accident that impacts the inside containment environment, FENOC requires confirmation of this assumption that a MSLB outside containment remains bounding.	
7	Section 7.5: BVPS-1 Steam Generator Tube Rupture purge is at 8 hrs, while releases go for 24 hrs; this is acceptable although it may not be optimal. (BVPS-2 SGTR purge is at 8 hrs and releases are terminated at 8 hrs.)	No change to text. This is part of the current BVPS licensing basis. Note: Pg 36, Addendum 4 of URB-219, R2 (current analysis of record for the BVPS1 SGTR), indicates that the dose will increase if the Control Room purge (currently assumed to occur at t=8 hours) was assumed to occur at t=24 hours.
8	Section 7.6: Table 6 of NRC Regulatory Guide 1.183 assumes the Analysis Release Duration for a PWR Locked Rotor Accident is “until cold shutdown is established”; for safe shutdown, BV1 is licensed as a Hot Standby plant and BV2 is licensed as an Approach to Cold Shutdown plant, so the Residual Heat Release Systems may not be available following a LRA that results in fuel damage and consequential high radiation fields with up to 20% failed fuel (contained in the Reactor Coolant System) for a LRA; however, the dose field may result in a significant cumulative dose, especially after RCS flow is initiated through the RHR pumps.	Incorporated as noted. *Assessment of the post-LRA radiation environment at the relevant RHR components to confirm the acceptability of the assumption that shutdown cooling via initiation of the RHR System is available for a LRA has been identified as Open Item 3 and is to be completed prior to submitting the license amendment request.* *Note that the radiation environment associated with increased coolant radioactivity due to a LRA (with 20% failed fuel) is expected to bound that associated with the SGTR (Iodine Spike). This assessment will be documented as part of the listed Open Item.*
9	Table 7.3-1: Specify the release rate from 8 hours to 30 days.	Incorporated to note that the rate of release during the 8 to 30 day period is conservatively assumed to be the same as that applicable to the previous time period.
10	Editorial changes were recommended.	Changes were incorporated, as agreed upon.

11	Section 10.0: ANSI/ANS 6.1.1-1991 is reference 22; later, ANSI/ANS 6.1.1-1977 is listed as reference 24; why not use 1991 (Ref 22) in place of 1977 (Ref 24)?	No change to text. Summarized below are some definitions relevant to this response: • Deep Dose Equivalent: The external whole-body exposure dose equivalent at a tissue depth of 1 cm (1000 mg/cm2) • Whole body Dose: Average value of the dose equivalent over the head, trunk, upper arms and upper thighs as a result of irradiation of the whole body and considered as uniform • Effective Dose Equivalent: The sum of the products of the dose equivalent to the organ or tissue (DE) and the weighting factors (WT) applicable to each of the body organs or tissues that are irradiated Section 4.1.1 of RG 1.183 notes that the total effective dose equivalent (TEDE) is the sum of the committed effective dose equivalent (CEDE) from inhalation, and the deep dose equivalent (DDE) from external exposure (i.e., the dose contribution from external "contained" sources such as piping, filters, the containment, etc; including external sources such as the radioactive cloud in which the operator is immersed, etc.). Section 4.1.4 of RG 1.183 allows the use of “Effective Dose Equivalent” (in lieu of Deep Dose Equivalent) when estimating the dose contribution from an external source that is irradiating “uniformly” on a body (e.g., an operator located in a control room, immersed in a finite radioactive cloud). ANSI/ANS 6.1.1-1991 provides the “effective” whole body dose equivalent (i.e., the sum of weighted organ dose equivalents) gamma ray fluence to dose conversion factors. However, based on Section 4.1.1, the dose contribution from external “contained” sources, such as the containment, filters, etc. (specifically, sources that have a directional relationship with the receptor), continue to be based on the maximum “deep dose equivalent”, whole body gamma ray fluence to dose conversion factors which are not included in the 1991 version, but are provided in the earlier version of this standard, i.e., ANSI/ANS 6.1.1-1977. Note: The whole body DCFs in the 1977 version of the standard bound the values presented in the 1991 version of the standard.