FENOC™
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077

Rod L. Penfield
Site Vice President

724-682-5234
Fax: 724-643-8069

October 20, 2019
L-19-099

10 CFR 50.90

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
<u>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"</u>

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) is requesting an amendment to both the Beaver Valley Power Station (BVPS), Unit Nos. 1 and 2 facility operating licenses. The proposed amendment would modify the Technical Specifications for the Unit No. 2 primary and secondary coolant activities, control room emergency ventilation system testing criteria, and permit a one-time change to the control room envelope unfiltered air inleakage test frequency.

The FENOC evaluation of the proposed change is enclosed. NRC staff approval of the proposed change is requested by October 30, 2020. The amendment will be implemented within 30 days of approval.

This letter contains no new regulatory commitments. If you have any questions regarding this report, 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 October 20, 2019.

Sincerely,

Rod L. Penfield

Enclosure: Evaluation of the Proposed Change

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

Subject: Application to Revise 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”

1.0 SUMMARY DESCRIPTION

The proposed change would revise Technical Specifications (TS) 3.4.16, "RCS [Reactor Coolant System] 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" for Operating Licenses DPR-66 for Beaver Valley Power Station Unit No. 1 (BVPS-1) and NPF-73 for Beaver Valley Power Station Unit No. 2 (BVPS-2).

The proposed changes to TSs 3.4.16 and 3.7.13 would be more restrictive and reduce the allowed reactor coolant system and secondary coolant specific activities for BVPS-2 only. As TSs 3.4.16 and 3.7.13 are common to both units, the proposed changes for BVPS-1 are administrative in nature.

The proposed changes to TS 5.5.7 for the control room emergency ventilation system (CREVS) are common to both units due to the shared control room envelope. The changes would relax the CREVS penetration and system bypass requirement, while the CREVS charcoal adsorber removal efficiency would be more restrictive.

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 following the test failure that occurred in October 2017.

The proposed TS changes are part of a request for NRC review and approval of updated licensing basis dose consequences for design basis accidents. The current licensing bases (CLB) at BVPS for design basis accident dose consequences use alternative source term (AST) methodology (except for the waste gas system rupture event described later) as allowed by 10 CFR 50.67, "Accident source term," and described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (Accession No. ML003716792).

Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides the existing TS Bases pages annotated to show the proposed changes and is provided for information only.

2.0 DETAILED DESCRIPTION

2.1 Current Licensing Basis and Technical Specification Requirements

2.1.1 Current Licensing Basis

FirstEnergy Nuclear Operating Company (FENOC) has implemented AST methodology into the current licensing bases at both BVPS units through various phases. Most recently, Amendment Nos. 275 (BVPS-1) and 156 (BVPS-2) fully implemented the alternative source term in support of the extended power uprate (Accession No. ML061720274). The loss of coolant accident dose consequences were then revised in

Amendment Nos. 280 (BVPS-1, Accession No. ML072680397) and 164 (BVPS-2, Accession No. ML080420549) due to containment sump screen modifications.

The proposed changes will affect the following analyses:

1. Loss of coolant accident (LOCA)
2. Control rod ejection accident (CREA)
3. Main steam line break (MSLB) outside of containment
4. Steam generator tube rupture (SGTR)
5. Reactor coolant pump (RCP) locked rotor accident (LRA)
6. Loss of alternating current power (LACP)
7. Fuel handling accident (FHA) in the fuel pool or in containment
8. Small line break (SLB) accident outside of containment

The waste gas system rupture (failure) events previously addressed in Amendment Nos. 275 (BVPS-1) and 156 (BVPS-2) are not addressed in this submittal as the proposed changes have been assessed and may be implemented by FENOC in accordance with 10 CFR 50.59, “Changes, tests and experiments.”

The eight analyses listed above are described in Chapter 14, “Safety Analysis” of the BVPS-1 Updated Final Safety Analysis Report (UFSAR) and Chapter 15, “Accident Analysis,” of the BVPS-2 UFSAR.

2.1.2 Current Technical Specification Requirements

The limiting condition for operation (LCO) for TS 3.4.16 is that RCS specific activity shall be within the limits during modes one and two, and mode three with RCS average temperature greater than or equal to 500 degrees Fahrenheit. Action A requires that the dose equivalent (DE) iodine-131 (I-131) be less than or equal to 0.35 microcuries per gram (μCi/gm), or the required action A.1 requires verification that the DE I-131 is within the acceptable region of Figure 3.4.16-1 once per four hours, and to restore DE I-131 to within the limit within 48 hours. Surveillance requirement (SR) 3.4.16.2 is only required to be performed in mode one and verifies that the reactor coolant DE I-131 specific activity is less than or equal to 0.35 μCi/gm on a frequency in accordance with the Surveillance Frequency Control Program and between two and six hours after a thermal power change of greater than or equal to 15 percent of rated thermal power within a one-hour period.

The LCO for TS 3.7.13, “Secondary Specific Activity,” is that the secondary coolant specific activity shall be less than or equal to 0.10 μCi/gm DE I-131 during modes one through four. SR 3.7.13.1 verifies the secondary coolant specific activity specified in this LCO on a frequency in accordance with the Surveillance Frequency Control Program.

TS 5.5.7, “Ventilation Filter Testing Program (VFTP),” Item b, requires, in part, that an inplace test of the CREVS charcoal adsorber shows a penetration and system bypass of less than 0.05 percent for a flowrate greater than or equal to 800 cubic feet per minute (cfm) and less than or equal to 1000 cfm when tested in accordance with the American National Standards Institute (ANSI) N510-1980.

TS 5.5.7, "Ventilation Filter Testing Program (VFTP)," Item c, requires, in part, that a laboratory test of the CREVS charcoal adsorber, when obtained as described in RG 1.52, "Design, Testing, And Maintenance Criteria For Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, or using a slotted tube sampler in accordance with ANSI N509-1980 shows, within 31 days after removal, a methyl iodide removal efficiency greater than or equal to 99 percent for both BVPS-1 and BVPS-2. The tests are required to be performed in accordance with American Society for Testing and Materials (ASTM) D3803-1989 at a temperature of 30 degrees Celsius, and inlet methyl iodide concentration of 1.75 milligrams per cubic meter (mg/m^3), a relative humidity of greater than or equal to 70 percent, and an air flow velocity of 0.68 feet per second (ft/sec) for BVPS-1 and 0.7 ft/sec for BVPS-2.

TS 5.5.14, "Control Room Envelope Habitability Program," Item c, requires, in part, that the unfilitered air inleakage past the control room envelope (CRE) boundary into the CRE be tested at the frequency specified in Section C.1 of Regulatory Guide 1.197, Revision 0, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," May 2003. Section C.1.5 of RG 1.197 states, in part, that CREs should be tested on a performance-based frequency consistent with Figure 1 of the RG. Per Figure 1 of the RG, if a CRE test fails and the test acceptance criterion is increased through recalculation of the consequences to the control room operators, a re-test is not required. However, a full test should be conducted three years later to ascertain whether the CRE's integrity has continued to degrade.

2.2 Reason for the Proposed Change

BVPS failed the most recent CRE unfiltered air inleakage test in October 2017. The test failure was documented in the FENOC Corrective Action Program, and an operability determination justified continued operation with the increased air inleakage test acceptance criterion. In accordance with Figure 1 of RG 1.197, a full CRE unfiltered air inleakage test should be conducted three years later if the test acceptance criterion is increased. This license amendment request is submitted to allow a one-time three-year extension to the test that would need to be performed by October 2020.

In addition to increasing the maximum allowable unfiltered air inleakage into the CRE, FENOC currently has low margin with three other input parameters to the BVPS design basis accident dose consequences:

1. The maximum allowable linear heat generation rate for fission product inventory in the fuel gap for both units.
2. The maximum allowable accident-induced steam generator tube leakage for BVPS-2.

3. The CREVS charcoal adsorber penetration and system bypass percentage in TS 5.5.7.b.

This license amendment would revise those input parameters identified above for future design margin.

2.3 Description of the Proposed Change

The proposed change would update the CLB for the eight analyses listed in Section 2.1 of this submittal, and make corresponding changes to the Technical Specifications. The proposed change would also provide a one-time extension to the Control Room Envelope Habitability Program unfiltered air inleakage test frequency in TS 5.5.14.

The eight analyses would have the inputs noted in Section 2.2 of this submittal updated to gain design and test margin and restrict other inputs to offset the increase in dose consequences. BVPS-2 reactor coolant and secondary coolant specific activities in TSs 3.4.16 and 3.7.13 would become more restrictive. BVPS-1 and BVPS-2 methyl iodide removal efficiency for the CREVS charcoal adsorbers in TS 5.5.7.c would become more restrictive as well.

Other changes to the dose consequence analyses include design input parameter changes to bound BVPS-1 operation, BVPS-2 operation, or a combination of both, as appropriate. The maximum allowed radial peaking factor used in core design for both BVPS-1 and BVPS-2 will be lowered from 1.75 to 1.70 as well. These changes and other minor updates to some of the design input values were incorporated as part of the re-verification effort and are listed in Attachment 3. Changes to address the Nuclear Safety Advisory Letter (NSAL) 11-5, “Westinghouse LOCA Mass and Energy Release Calculation Issues,” dated July 25, 2011 (Accession No. ML13239A479) are also included. A comparison of the major dose consequence inputs between the CLB and the new proposed inputs are provided in Section 3 below.

The proposed change to the TSs are as follows:

1. The proposed change to TS 3.4.16 would require a different DE I-131 condition in Action A for each unit.

    - Instead of the existing Condition A:

      DOSE EQUIVALENT I-131 > 0.35 μCi/gm.

      The proposed Condition A would be:

      DOSE EQUIVALENT I-131 > 0.35 μCi/gm (Unit 1), and > 0.10 μCi/gm (Unit 2).

- Required Action A.1 would change from:

  Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.

  To:

  Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1 (Unit 1), and Figure 3.4.16-2 (Unit 2).

- Words after the logical OR section of Condition C would change from:

  DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.

  To:

  DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1 (Unit 1), and Figure 3.4.16-2 (Unit 2).

- SR 3.4.16.2 would change from:

  Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.35 μCi/gm.

  To:

  Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.35 μCi/gm (Unit 1), and ≤ 0.10 μCi/gm (Unit 2).

The frequency of SR 3.4.16.2 for both units is unchanged.

Figure 3.4.16-1 would have an administrative change to add "(Unit 1)" at the end of the figure title.

A new Figure 3.4.16-2 is added for the BVPS-2 specific activity limit versus percent of rated thermal power.

2. The proposed change to TS 3.7.13 would require a different secondary coolant specific activity for each unit.

   - LCO 3.7.13 would change from:

     The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

To:

The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 1), and ≤ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2).

- SR 3.7.13.1 would change from:

  Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

  To:

  Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 1), and ≤ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2).

The frequency of SR 3.7.13.1 for both units is unchanged.

3. The proposed change to TS 5.5.7.b would change the CREVS ventilation system requirement from:

ESF Ventilation System	Penetration	Flowrate
CREVS	< 0.05%	≥ 800 cfm and ≤ 1000 cfm

To:

ESF Ventilation System	Penetration	Flowrate
CREVS	< 0.5%	≥ 800 cfm and ≤ 1000 cfm

4. The proposed change to TS 5.5.7.c would change the CREVS removal efficiency requirements from:

ESF Ventilation System	Removal Efficiency	Air Flow Velocity	RH
CREVS	99% (Unit 1)	0.68 ft/sec (Unit 1)	≥ 70% (Unit 1)
	99% (Unit 2)	0.7 ft/sec (Unit 2)	≥ 70% (Unit 2)

To:

ESF Ventilation System	Removal Efficiency	Air Flow Velocity	RH
CREVS	99.5% (Unit 1)	0.68 ft/sec (Unit 1)	≥ 70% (Unit 1)
	99.5% (Unit 2)	0.7 ft/sec (Unit 2)	≥ 70% (Unit 2)

5. The proposed change to TS 5.5.14 would add the following note to Item 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.

3.0 TECHNICAL EVALUATION

The eight BVPS current licensing basis accident dose consequence analyses listed in Section 2.1 of this submittal have been re-analyzed with the proposed changes described in Section 2.3 of this submittal. The dose acceptance criteria, the changed inputs, and the results of these analyses are described in sections 3.1 through 3.3 below. The technical evaluation for the proposed CRE unfiltered air inleakage test extension is provided in Section 3.4.

3.1 Dose Acceptance Criteria

In accordance with the CLB, the acceptance criteria for dose consequences at the exclusion area boundary (EAB) and low population zone (LPZ) are based on 10 CFR 50.67(b)(2) and also Section 4.4 and Table 6 of RG 1.183, that states, in part:

(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, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(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), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

RG 1.183 does not specifically address American Nuclear Society (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, 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 SLB accident outside of containment will be the most limiting dose criterion in Table 6 of RG 1.183, that is, 2.5 rem TEDE.

The acceptance criteria for the control room dose is from 10 CFR 50.67(b)(2):

(iii) Adequate radiation protection is provided to permit access to and 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 CLB, the habitability of the BVPS Emergency Response Facility, or ERF, (which houses the Technical Support Center (TSC)) is assessed. The assessment demonstrates compliance from a radiological perspective with paragraph IV.E.8.a(i) of Appendix E to 10 CFR Part 50 and NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability," December 17, 1982. Therefore, the dose acceptance criteria to any person working in the TSC is less than or equal to 5 rem TEDE for the duration of the accident. In accordance with the current BVPS licensing basis, the ERF is only assessed for the LOCA. This is justified as the LOCA results in the largest EAB, LPZ, and CR doses.

3.2 Proposed Input Changes

Attachment 3 provides summary tables of important parameters to the individual analyses comparing the CLB with the updates in this submittal. They are also described below.

3.2.1 Radiation Source Terms

The reactor and secondary coolant activity concentrations were updated for the proposed BVPS-2 DE-131 limits.

Current BVPS TS LCO 3.4.16 also limits the reactor coolant gross specific activity to $100/\bar{E}$ μCi/gm. This LCO remains unchanged by this application and remains applicable to both units.

In accordance with the CLB, 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 3.4.16. 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 (between iodine and non-iodine isotopes) such that both TS limits co-exist at the same time.

In accordance with the CLB, 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 International Commission on Radiological Protection (ICRP) Publication 30, “Limits for Intakes of Radionuclides by Workers,” Supplement to Part 1, 1979, as noted in the definition of DE 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:

- NUREG 76-6521, “Radiation Signature Following the Hypothesized LOCA,” Sandia Laboratories, September 1977
- DOE/TIC-11026, “Radioactive Decay Data Tables – A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments,” Kocher, David C., 1981
- Lawrence Berkeley Laboratory, University of California, Berkeley, “Table of Isotopes,” Seventh Edition
- ORIGEN2, “Isotopic Generation and Depletion Code-Matrix Exponential Method,” Oak Ridge National Laboratory, September 1983

Consistent with the CLB, the updated analysis continues to demonstrate that the failed fuel percentage associated with TS 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 LCO 3.7.13 limits will be retained for BVPS-1. The proposed LCO 3.7.13 limits for BVPS-2 will be less than or equal to 0.05 μCi/gm DE I-131. The TS iodine dose equivalent I-131 concentrations per iodine nuclide in the secondary coolant are calculated using the same methodology discussed above 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 3.2-1A and Table 3.2-1B, respectively.

The coolant concentrations based on the current 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.

Table 3.2-1A BVPS-1 Reactor and Secondary Coolant TS 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 3.2-1B Proposed BVPS-2 Reactor and Secondary Coolant TS 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	--

3.2.2 Pre-Accident and Concurrent Iodine Spike and Equilibrium Iodine Appearance Rates

In accordance with the current TS 3.4.16 limit for BVPS-1 and the proposed TS 3.4.16 limit for BVPS-2, the pre-accident iodine spike concentration limit in the reactor coolant is 21 µCi/gm DE I-131 (transient limit for full power operation) and 6 µCi/gm DE I-131, respectively (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 and the equilibrium iodine appearance rate. Consistent with the CLB, the equilibrium appearance rates are conservatively calculated based on the TS reactor coolant activities, along with the maximum design letdown rate, maximum TS allowed leakage, and an ion-exchanger iodine removal efficiency of 100 percent. 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 3.2-2A and Table 3.2-2B, respectively.

Table 3.2-2A
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 3.2-2B
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

3.2.3 Gap Fractions for Non-LOCA Events

BVPS has three design basis non-LOCA accidents that are postulated to result in fuel damage (CREA, RCP LRA for BVPS-2 only, and FHA). Table 3 of RG 1.183 provides the fractions of fission product inventory in the fuel gap for these accidents. From Note 11 of the table, referenced gap fractions are acceptable for use with currently-approved light water reactor fuel with a peak burnup up to 62,000 megawatt-days per metric ton of uranium (MWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatts per foot (kw/ft) peak rod average power for burnups exceeding 54 gigawatt-days per metric ton of uranium (GWD/MTU). As stated in Section 2.2 of this submittal, FENOC currently has low margin to exceeding the burnup limit.

Consistent with other recent industry AST-related amendments, FENOC is proposing to use the gap fractions provided in Table 3 of Draft Regulatory Guide (DG)-1199 (proposed Revision 1 of RG 1.183) "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 2009 (Accession No. ML090960464), for all BVPS Non-LOCA events other than reactivity initiated accidents, where only the fuel clad is postulated to be breached. BVPS-1 and BVPS-2 falls within, and FENOC intends to operate within, the maximum allowable power operating envelope for non-LOCA gap fractions, as shown in Figure 1 of DG-1199.

Consistent with the CLB and 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 percent
Halogens:	10 percent

The gap fractions used to assess the dose consequences of the RCP LRA and FHA are as follows:

Nuclide Group	Gap Fraction for RCP LRA and 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 3.2-3, and is consistent with the values presented for these isotopes in Table 5.3.3-1 of the FENOC letter for BVPS-1 and BVPS-2 dated June 5, 2002, “License Amendment Request Nos. 300 and 172,” (Accession No. ML021620298).

Table 3.2-3 BVPS Core Inventory of Dose Significant Isotopes in the Gap (2918 MWt)					
Noble Gases		Halogens		Alkali Metals and 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 are to be adjusted for a) failed fuel percentage, b) radial peaking factor that is being reduced from 1.75 to 1.70 with this submittal, and c) number of fuel rods damaged (if applicable), prior to assessing the associated dose consequences of gap releases following the CREA, RCP LRA, and FHA. 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.

3.2.4 Control Room Envelope Unfiltered Air Inleakage

Except as noted below with respect to the flow rates associated with unfiltered air inleakage and intake into the CRE during all modes of control room operation, the BVPS control room design and operation is the same as described in the submittals for amendments 275 (BVPS-1) and 156 (BVPS-2). 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 shared control room has 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 modes of operation.

To provide operational margin, the dose consequence analyses summarized herein assume that during normal plant operation, the joint BVPS-1 and BVPS-2 unfiltered intake plus inleakage is a maximum of 1250 cubic feet per minute (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 and egress.

The containment high-high pressure signal (the containment isolation phase B signal (CIB)) from either unit will initiate the BVPS-2 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 makes the lower intake rate of 800 cfm more limiting from a dose consequence perspective. This argument holds true because the committed effective dose equivalent (CEDE) from inhalation is far more limiting than the deep dose equivalent (DDE) from immersion which is principally from noble gases.

The CR emergency ventilation intake filter has an efficiency of 99 percent for particulates, and 98 percent 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 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 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 16,200 cfm.

In accordance with the CLB, the atmospheric dispersion factors associated with control room air inleakage are considered to be 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 and 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 and 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 and 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 3.2-4 lists key assumptions and input parameters associated with BVPS control room design and applicable to the dose consequence analyses in this submittal.

Table 3.2-4[1] Analysis Assumptions and Key Parameter Values BVPS Common Control Room	
Control Room Parameters	
Free Volume	173,000 cubic feet (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, percent	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 CREVS Considering a Loss of Offsite Power	
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.

3.2.5 BVPS-2 Maximum Allowable Accident-Induced Steam Generator Tube Leakage

BVPS-2 is licensed for the alternate repair criteria in accordance with NRC Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

The BVPS-2 main steam line break (MSLB) dose analysis is performed to establish a maximum allowable accident-induced steam generator (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 gallons per day (gpd) at standard temperature and pressure, 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 CR. The maximum allowed accident-induced SG tube leakage at BVPS-2 is increased from 2.1 gpm to 8.1 gpm.

As indicated earlier, 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 TS activity concentrations to 0.10 μCi/gm DE 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. <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 proposed BVPS-2 TS 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 TS 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 TS iodine concentrations in the coolant for BVPS-2.

b. <u>Concurrent iodine spike</u> - 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 TS 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 TSs 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 main steam safety valves and atmospheric dump valves, respectively.

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

Attachment 4 contains the iodine spiking analysis for BVPS-2 to support the reduction in reactor coolant system iodine activity below 0.35 µCi/gm DE I-131 in accordance with Section 2.b.4 of Attachment 1 to GL 95-05.

3.3 Analysis Results

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

- With the exception of the MSLB, SGTR, LRA, and FHA analyses that 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 alternate repair criteria (ARC) to establish a maximum allowable accident-induced steam generator tube leakage, against which cycle-specific 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 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 ERF (which houses the 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).

From the results listed below, the LOCA results in the largest EAB, LPZ, and CR doses. Therefore, it is the only accident considered for determining the limiting ERF (TSC) occupant dose. This is consistent with the current BVPS licensing basis.

In accordance with RG 1.183, the worst two-hour period dose at the EAB and the dose at the LPZ for the duration of the release are presented below in Table 3.3-1. 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 and LPZ attributed to direct shine from contained sources is considered negligible for all the accidents.

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 below in Table 3.3-2 for 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 and the external cloud is identified and included in the CR doses reported in Table 3.3-2.

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

The CLB accidents analyzed in this LAR have EAB, LPZ, and CR doses that remain at or below the regulatory limits.

Table 3.3-1

**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)
RCP Locked Rotor Accident(8)	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 (hr) 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 tube leakage (8.1 gpm) into the affected SG.

(6) The 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.

(8) For BVPS-1, the dose from a postulated locked rotor accident is bounded by the BVPS-2 locked rotor accident.

Table 3.3-2

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
Fuel Handling Accident(6) BVPS-1(5) BVPS-2	 4.2 1.4	5
RCP Locked Rotor Accident(6)(7)	2.9	5
Loss of AC Power(6)	(Note 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 tube 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, RCP LRA, LACP, FHA, and SLBOC.

(7) For BVPS-1, the dose from a postulated locked rotor accident is bounded by the BVPS-2 locked rotor accident.

3.4 CRE Unfiltered Air Inleakage Test Extension

TS 5.5.14, “Control Room Envelope Habitability Program,” Item c, requires, in part, that the unfiltered air inleakage past the CRE boundary into the CRE be tested at the frequency specified in Section C.1 of RG 1.197, which states, in part, that CREs should be tested on a performance-based frequency consistent with Figure 1. Per Figure 1, if a CRE test fails and the test acceptance criterion is increased through recalculation of the consequences to the control room operators, a re-test is not required. However, a full test should be conducted three years later to ascertain whether the CRE’s integrity has continued to degrade.

FENOC is proposing to extend the full test period to the normal six-year frequency instead of the three-year retest frequency. FENOC has high confidence that the CRE 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. 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.

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.

Below are the historical test results in cubic feet per minute for the CRE tracer gas tests performed at BVPS for each mode of control room ventilation system operation. These test results include the uncertainty and 10 cfm for ingress and egress as required per RG 1.197:

Mode Tests	Acceptance Criteria	Results (RG 1.197 Values)					
		2001	2008	2015	2016	2016 (Run 2)	2017
Normal	500	-	1348	-	1316	1077	659
Recirculation (Isolation)	300	287	325	366	-	-	195
Unit 1 Pressurization (Emergency)	30	45	10	89	-	-	76
Unit 2 Pressurization (Emergency)	30	10	10	98	78	86	95

As shown above, the 2017 test results for the normal mode and both emergency pressurization modes did not meet the acceptance criteria for the current licensing basis.

The proposed acceptance criteria are listed below and described in more detail in Section 3.2.4:

Mode	Current Acceptance Criteria	Proposed Acceptance Criteria
Normal	500 cfm	1250 cfm
Recirculation	300 cfm	450 cfm
Unit 1 Pressurization	30 cfm	165 cfm
Unit 2 Pressurization	30 cfm	165 cfm

Extending the interval until the next CR air inleakage tracer gas test is not expected to significantly reduce a margin of safety because the proposed amendment would provide test margin compared to the most recent test results, and is expected to bound future test results.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following NRC regulatory requirements are applicable to this license amendment request.

<u>10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants</u>
Requires licensees to establish a program for electrical equipment important to safety that is relied upon to remain functional during and following design basis events. The equipment qualification program, in part, must include consideration of the radiation environment. During normal operation, the proposed amendment would reduce the primary and secondary coolant specific activities for BVPS-2, while the activities for BVPS-1 would remain the same. The environmentally-qualified electric equipment was re-assessed for the revised accident doses and determined to remain acceptable.

<u>10 CFR 50.67, Accident Source Term</u>
Specifies requirements for holders of operating licenses who seek to revise the current accident source term used in their design basis radiological analyses. As part of the plant licensing basis, applicants for pressurized water reactor licenses are required to analyze the consequences of postulated design-basis accidents. The postulated design basis accidents with dose consequences have been re-analyzed as part of the proposed amendment, and therefore, satisfies this requirement.

<u>10 CFR 50, Appendix A. General Design Criteria for Nuclear Power Plants</u>
General Design Criterion 19 of 10 CFR Part 50, Appendix A, in part, defines requirements for a control room from which actions can be taken to maintain a nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents with adequate radiation protection to limit radiation exposure for the duration of the accident. The

postulated design basis accidents with dose consequences have been re-analyzed as part of the proposed amendment with results that meet regulatory limits, and therefore, Criterion 19 continues to be met.

4.2 No Significant Hazards Consideration Analysis

FENOC requests a change to the BVPS Technical Specifications and dose consequence design basis accident licensing bases. The proposed amendment would change 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."

The proposed changes to TSs 3.4.16 and 3.7.13 would be more restrictive and reduce the allowed reactor coolant system and secondary coolant specific activities for BVPS-2 only. As TSs 3.4.16 and 3.7.13 are common to both units, the proposed changes for BVPS-1 are administrative in nature.

The proposed changes to TS 5.5.7 for the control room emergency ventilation system are common to both units due to the shared control room envelope. They would relax the CREVS penetration and system bypass requirement, while the CREVS charcoal adsorber removal efficiency would be more restrictive.

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 following the test failure that occurred in October 2017.

FENOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment re-analyzes EAB, LPZ, and CR doses for eight design basis accidents to improve dose consequence analysis input parameters for which FENOC currently has low margin. The proposed amendment changes accident analysis inputs for calculating dose consequences at the EAB, LPZ, and CR. There are no plant modifications or operating procedure changes that would increase the probability of an accident previously evaluated. While the revised doses for the EAB and LPZ remain approximately the same as before, CR doses generally increase but remain below the allowable regulatory limits with at least 10 percent margin.

The proposed amendment would also extend the interval until the next CR air inleakage tracer gas test. Extending the test interval would not change the probability of an accident previously evaluated, and the consequences are not expected to change due to the consistent test results observed from the multiple runs between 2015 and 2017.
Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   The proposed amendment changes accident analysis inputs for calculating dose consequences at the EAB, LPZ, and CR. It would also extend the interval until the next CR air inleakage tracer gas test. There are no plant modifications or operating procedure changes.

   Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The proposed amendment re-analyzes EAB, LPZ, and CR doses for eight design basis accidents to improve dose consequence analysis input parameters for which FENOC currently has low margin. The proposed amendment changes accident analysis inputs for calculating dose consequences at the EAB, LPZ, and CR. While the revised doses for the EAB and LPZ remain approximately the same as before, CR doses generally increase yet remain below the allowable regulatory limits. The margin of safety for the radiological consequences of these accidents is provided by meeting the applicable regulatory limits. An acceptable margin of safety is inherent in these limits.

   Extending the interval until the next CR air inleakage tracer gas test is not expected to significantly reduce a margin of safety because the proposed amendment would provide test margin compared to the most recent test results, and is expected to bound future test results.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Attachment 1

Technical Specification Page Markups
(9 pages follow)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 0.35 μCi/gm~~.~~ (Unit 1), and > 0.10 μCi/gm (Unit 2).	-- **- NOTE -** LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1 (Unit 1), and Figure 3.4.16-2 (Unit 2).	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1 (Unit 1), and Figure 3.4.16-2 (Unit 2).	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	--- **- NOTE -** Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.35 μCi/gm~~.~~ (Unit 1), and ≤ 0.10 μCi/gm (Unit 2).	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	--- **- NOTE -** Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	In accordance with the Surveillance Frequency Control Program

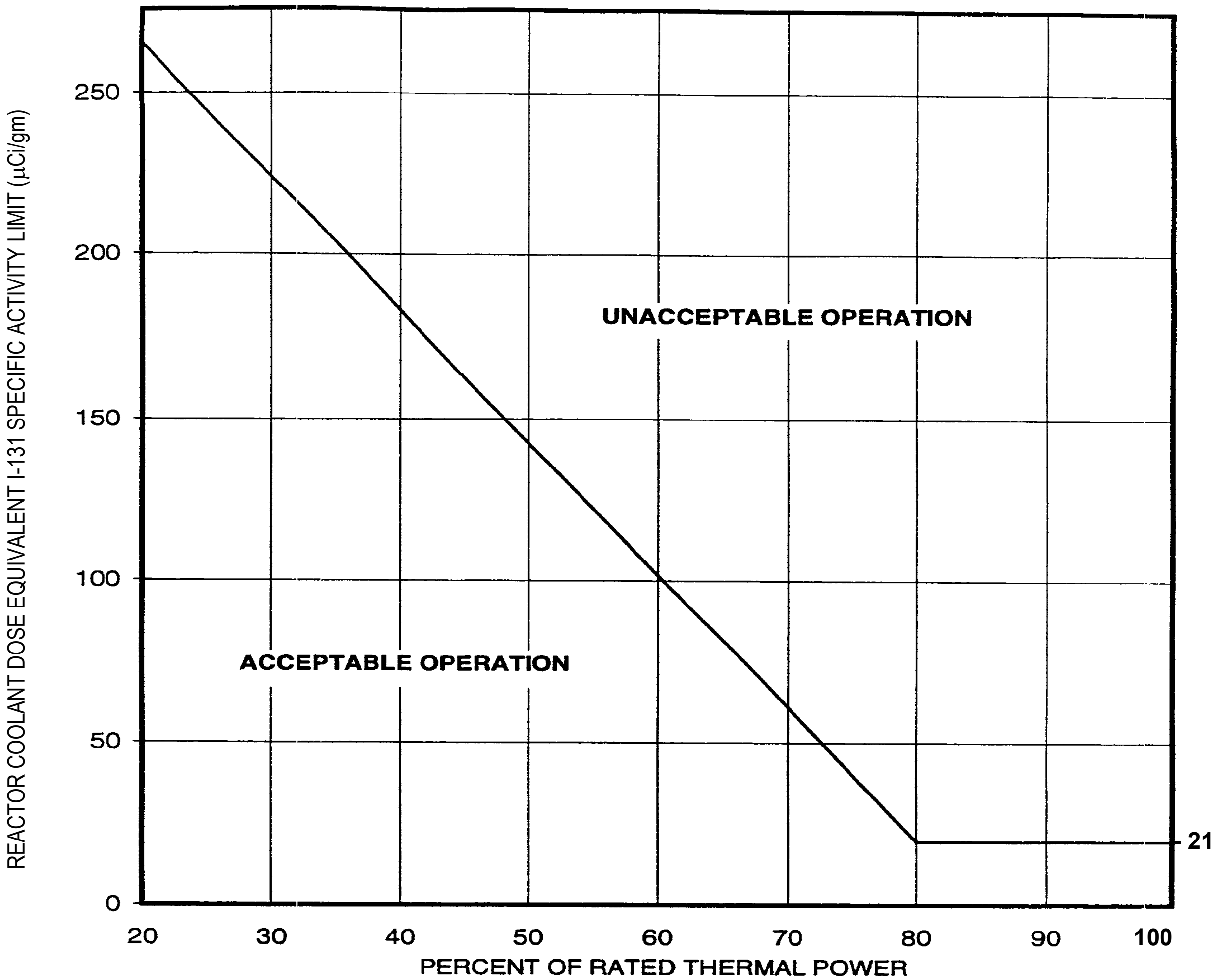

Figure 3.4.16-1 (Page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER (Unit 1)

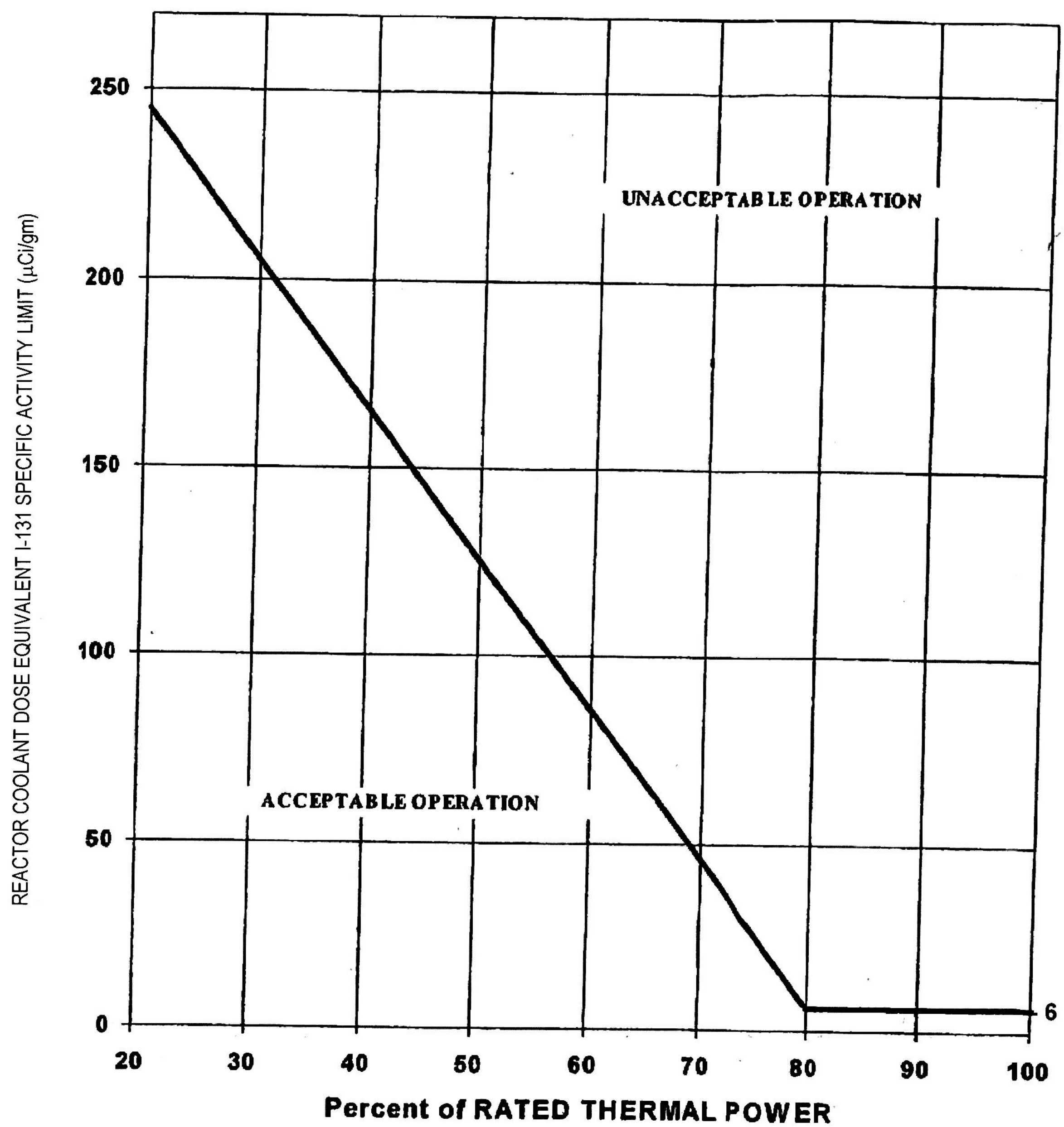

Figure 3.4.16-2 (Page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER (Unit 2)

3.7 PLANT SYSTEMS

3.7.13 Secondary Specific Activity

LCO 3.7.13 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 1), and ≤ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 1), and ≤ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2).	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

Tests described in Specification 5.5.7.c shall be performed at least once per 18 months and after the following:

- 720 hours of adsorber operation (for the Unit 1 and 2 CREVS and the Unit 1 SLCRS) or after 4 months of adsorber operation (for the Unit 2 SLCRS);
- Any structural maintenance on the charcoal adsorber bank housing;
- Significant painting, fire, or chemical release (for the Unit 1 and Unit 2 SLCRS) in any ventilation zone communicating with the system while the filtration system is operating; and
- Significant painting, fire, or chemical release (for the Unit 1 and Unit 2 CREVS) in the vicinity of control room outside air intakes while the system is operating.

Tests described in Specifications 5.5.7.d and 5.5.7.e shall be performed at least once per 18 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the required ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass specified below when tested in accordance with ANSI N510-1980 (for the Unit 1 and 2 CREVS) and the Unit 2 SLCRS and in accordance with ANSI N510-1975 (for the Unit 1 SLCRS) at the system flowrate specified below:

ESF Ventilation System	Penetration	Flowrate
SLCRS	$< 1.0\%$ (Unit 1) $< 0.05\%$ (Unit 2)	$\geq 32{,}400$ cfm and $\leq 39{,}600$ cfm (Unit 1) $\geq 51{,}300$ cfm and $\leq 62{,}700$ cfm (Unit 2)
CREVS	$< 0.05\%$	≥ 800 cfm and ≤ 1000 cfm

b. Demonstrate for each of the required ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass specified below when tested in accordance with ANSI N510-1980 (for the Unit 1 and 2 CREVS and the Unit 2 SLCRS) and ANSI N510-1975 (for the Unit 1 SLCRS) at the system flowrate specified below:

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	Flowrate
SLCRS	< 1.0% (Unit 1) < 0.05% (Unit 2)	≥ 32,400 cfm and ≤ 39,600 cfm (Unit 1) ≥ 51,300 cfm and ≤ 62,700 cfm (Unit 2)
CREVS	< 0.~~0~~5%	≥ 800 cfm and ≤ 1000 cfm

c. Demonstrate for each of the required ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, or using a slotted tube sampler in accordance with ANSI N509-1980 shows, within 31 days after removal, the methyl iodide removal efficiency greater than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, an inlet methyl iodide concentration of 1.75 mg/m^3, and an air flow velocity and relative humidity (RH) specified below:

ESF Ventilation System	Removal Efficiency	Air Flow Velocity	RH
SLCRS	90% (Unit 1) 99% (Unit 2)	0.9 ft/sec (Unit 1) 0.7 ft/sec (Unit 2)	≥ 95% (Unit 1) ≥ 70% (Unit 2)
CREVS	99.5% (Unit 1) 99.5% (Unit 2)	0.68 ft/sec (Unit 1) 0.7 ft/sec (Unit 2)	≥ 70% (Unit 1) ≥ 70% (Unit 2)

d. Demonstrate for each of the required ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified as follows:

ESF Ventilation System	Delta P	Flowrate
SLCRS	6 inches Water Gauge (Unit 1) 6.8 inches Water Gauge (Unit 2)	≥ 32,400 cfm and ≤ 39,600 cfm (Unit 1) ≥ 51,300 cfm and ≤ 62,700 cfm (Unit 2)
CREVS	6 inches Water Gauge (Unit 1) 5.6 inches Water Gauge (Unit 2)	≥ 800 cfm and ≤ 1000 cfm (Unit 1) ≥ 800 cfm and ≤ 1000 cfm (Unit 2)

5.5 Programs and Manuals

5.5.12 Containment Leakage Rate Testing Program (continued)

b) For each emergency air lock door, no detectable seal leakage when gap between door seals is pressurized to ≥ 10 psig or door seal leakage quantified to ensure emergency air lock door seal leakage rate is ≤ 0.0005 L_a when tested at ≥ 10 psig.

c) For each personnel air lock door, no detectable seal leakage when gap between door seals is pressurized to ≥ P_a or door seal leakage quantified to ensure personnel air lock door seal leakage rate is ≤ 0.0005 L_a when tested at ≥ P_a.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.13 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V,

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates, and

c. Actions to verify the remaining cells are ≥ 2.07 V when a cell or cells have been found to be < 2.13 V.

5.5.14 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

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.

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 2

Technical Specification Bases Page Markups (for information only)
(7 pages follow)

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event.

Primary to secondary LEAKAGE is a factor in the dose assessment of accidents or transients that involve secondary steam release to the atmosphere, such as a main steam line break (MSLB), a locked rotor accident (LRA), a Loss of AC Power (LACP), a Control Rod Ejection Accident (CREA) and to a lesser extent, a Steam Generator Tube Rupture (SGTR). The leakage contaminates the secondary fluid. The limit on the primary to secondary LEAKAGE ensures that the dose contribution at the site boundary from tube leakage following such accidents are limited to appropriate fractions of the 10 CFR 50.67 limit of 25 Rem TEDE as allowable by Regulatory Guide 1.183. The limit on the primary to secondary leakage also ensures that the dose contribution from tube leakage in the control room is limited to the 10 CFR 50.67 limit of 5 Rem TEDE. Among all of the analyses that release primary side activity to the environment via tube leakage, the MSLB is of particular concern because the ruptured main steam line provides a pathway to release the primary to secondary leakage directly to the environment without dilution in the secondary fluid.

For Unit 1, the safety analysis for an event resulting in steam discharge to the atmosphere conservatively assumes that primary to secondary LEAKAGE from all steam generators is 450 gallons per day (gpd) (i.e., 150 gpd per steam generator) or increases to 450 gpd as a result of accident induced conditions. Currently, the Unit 1 safety analyses do not specifically assume additional primary to secondary LEAKAGE due to accident induced conditions.

For Unit 2, due to adoption of the voltage based steam generator tube repair criteria per guidance provided by Generic Letter (GL) 95-05 (Reference 3), the safety analysis for an event resulting in steam discharge to the atmosphere conservatively assumes that primary to secondary LEAKAGE from all steam generators is 450 gallons per day (gpd) (i.e., 150 gpd per steam generator) or increases to 450 gpd as a result of accident induced conditions for all accidents other than the MSLB. Currently, the Unit 2 MSLB safety analysis is the only analysis that specifically assumes additional primary to secondary LEAKAGE due to accident induced conditions.

The Unit 2 dose consequences associated with the MSLB addresses an additional ~~2~~8.1 gpm leakage, which, per GL 95-05, is postulated to occur (via pre-existing tube defects) as a result of the rapid depressurization of the secondary side due to the MLSB, and the consequent high differential pressure across the faulted steam generator. The maximum allowed Unit 2 total accident induced leakage is ~~2~~8.4 gpm.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analyses are for two cases of reactor coolant specific activity. One case assumes specific activity at 0.35 μCi/gm DOSE EQUIVALENT I-131 (Unit 1 and Unit 2, except Unit 2 MSLB) and 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 2 MSLB only) with a concurrent large iodine spike that increases the I-131 activity appearance rate in the reactor coolant by a factor of 500 (MSLB) or 335 (SGTR) immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 21 μCi/gm DOSE EQUIVALENT I-131 (Unit 1 and Unit 2, except Unit 2 MSLB) and 6 μCi/gm DOSE EQUIVALENT I-131 (Unit 2 MSLB only) due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant is based on the equilibrium concentrations predicted while operating with 1% failed fuel, and proportionately reduced to correspond to the reduced concentrations of DOSE EQUIVALENT I-131.

The safety analyses show the radiological consequences of a~~n~~ MSLB or SGTR accident are within the Reference 1 dose guideline limits for the pre-accident iodine spike case, and well within the 10 CFR 50.67 dose guidelines for the concurrent iodine spike case. Operation with iodine specific activity levels greater than the LCO limit is permissible for up to 48 hours, if the activity levels do not exceed the limits shown in Figures 3.4.16-1 and 3.4.16-2 for Units 1 and 2, respectively. The safety ~~analysis~~ analyses ~~has~~ have pre-accident iodine spiking levels up to 21 μCi/gm DOSE EQUIVALENT I-131 (Unit 1 and Unit 2, except Unit 2 MSLB) and 6 μCi/gm DOSE EQUIVALENT I-131 (Unit 2 MSLB only).

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 (Unit 1) and Figure 3.4.16-2 (Unit 2) are acceptable because of the low probability of a MSLB or SGTR accident occurring during the established 48 hour time limit. The occurrence of a~~n~~ MSLB or SGTR accident at the permissible levels applicable from 80 to 100% power could increase the site boundary dose levels, but still be within 10 CFR 50.67 dose guideline limits.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 0.35 μCi/gm DOSE EQUIVALENT I-131 (Unit 1) and 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 2), and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the non-iodine coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the TEDE at the site boundary and in the control room during the Design Basis Accident (DBA) will be an appropriate fraction of the allowed TEDE dose. The limit on gross specific activity provides an additional indication of radionuclides (excluding iodines) that corresponds closely to the noble gas activity in the RCS and helps to ensure the effective doses during the DBA will be an appropriate fraction of the allowed dose.

BASES

LCO (continued)

The MSLB and SGTR accident analyses (Ref. 2) show that the resultant dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a~~n~~ MSLB or SGTR, lead to site boundary or control room doses that exceed the 10 CFR 50.67 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to limit the potential radiological consequences of a~~n~~ MSLB or SGTR to within the acceptable site boundary and control room dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the secondary side steam pressure is significantly reduced which in turn reduces the probability and severity of a MSLB or a SGTR.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 (Unit 1) and Figure 3.4.16-2 (Unit 2) are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

BASES

ACTIONS (continued)

B.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the secondary side steam pressure which in turn reduces the probability and severity of a MSLB or SGTR. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1 (Unit 1) and Figure 3.4.16-2 (Unit 2), the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

APPLICABLE SAFETY ANALYSES (continued)

determine the maximum MSLB induced primary to secondary leak rate that could occur without offsite doses exceeding the limits of 10 CFR 50.67 (Ref. 2) as supplemented by Regulatory Guide 1.183 (Ref. 3) and without control room doses exceeding GDC-19 (Ref. 4). An additional ~~2~~8.1 gpm leakage is assumed in the Unit 2 MSLB analysis resulting from accident conditions. Therefore, in the MSLB analysis, the steam discharge to the atmosphere includes primary to secondary LEAKAGE equivalent to the operational leakage limit of 150 gpd per SG and an additional ~~2~~8.1 gpm which results in a total assumed accident induced leakage of ~~2~~8.4 gpm.

The combined projected leak rate from all sources (i.e., voltage based repair criteria, application of F*, freespan crack, leaking plug, leakage past sleeves, etc.) for each SG must be less than the maximum allowable ~~steam line break~~MSLB leak rate limit in any one steam generator (i.e., ~~2~~8.2 gpm) in order to maintain a total assumed accident induced leakage of ≤ ~~2~~8.4 gpm as explained above. Maintaining the total assumed accident induced leakage to ≤ ~~2~~8.4 gpm limits the resulting dose to within the requirements of 10 CFR 50.67 (Ref. 2) as supplemented by Regulatory Guide 1.183 (Ref. 3) and within GDC-19 (Ref. 4) values during a postulated ~~steam line break~~MSLB event.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A Note modifies the LCO to indicate that any reference to the repair of SG tubes is only applicable to Unit 2 at this time. The Unit 1 "Steam Generator Program" (in Specification 5.5.5) has no provision for SG tube repair.

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging or repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging or repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the plugging or repair criteria but was not plugged or repaired, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.5, "Steam

B 3.7 PLANT SYSTEMS

B 3.7.13 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 150 gallons per day steam generator tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 0.35 μCi/gm DOSE EQUIVALENT I-131 (Unit 1) and 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 2) (LCO 3.4.16, "RCS Specific Activity"). The ~~steam line failure~~MSLB is assumed to result in the release of the iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE.

Operating a unit at the allowable primary and secondary coolant specific activity limits will result in exposures within the 10 CFR 50.67 (Ref. 1) total effective dose equivalent (TEDE) limits, as supplemented by Regulatory Guide 1.183 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 14 (Unit 1) and Chapter 15 (Unit 2) (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 1) and 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2). This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the 10 CFR 50.67 (Ref. 1) TEDE limits, as supplemented by Regulatory Guide 1.183 (Ref. 3).

The MSLB accident analysis assumes a total release of iodine activity in the steam generator connected to the failed steam line. In addition, a portion of the iodine activity in the remaining steam generators is also released via the steaming process due to assumption of loss of offsite

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. With the loss of offsite power, the remaining steam generators are utilized for core decay heat removal by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 (Unit 1) and ≤ 0.05 μCi/gm DOSE EQUIVALENT I-131 (Unit 2) to limit the radiological consequences of a Design Basis Accident (DBA) to within the required limits (Ref. 1 and Ref. 3).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

Attachment 3

Comparison of Important Analysis Parameters
(14 pages follow)

This attachment provides a comparison between the design input values used in the CLB dose consequence analyses supporting BVPS-1 and BVPS-2, to those utilized in the dose consequence analyses supporting this 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. The core inventory and the atmospheric dispersion factors remain unchanged by this application.

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

Table 4-1
Changes to Gap Fraction for Non-LOCA Events

Item	Proposed[(2)]	Current[(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 4-2a
BVPS-1 Reactor Coolant
TS Iodine and Noble Gas Activity Concentrations (µCi/gm)

Item	Proposed(1) 0.35 µCi/gm DE I-131	Current 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 percent fuel defects) reactor coolant concentrations.

Table 4-2b
BVPS-2 Reactor Coolant
TS Iodine and Noble Gas Activity Concentrations (μCi/gm)

Item	Proposed[1] 0.10 μCi/gm DE I-131	Current 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) The LAR is proposing to lower the TS LCO 3.4.16 limit to 0.10 μCi/gm DE I-131 for BVPS-2.

Table 4-3a BVPS-1 Reactor Coolant TS Equilibrium Iodine Appearance Rates (µCi/sec)		
Item	**Proposed[(1)]** **0.35 µCi/gm DE I-131**	**Current** **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 percent fuel defects) reactor coolant concentrations.

Table 4-3b BVPS-2 Reactor Coolant TS Equilibrium Iodine Appearance Rates (µCi/sec)		
Item	**Proposed[(1)]** **0.10 µCi/gm DE I-131**	**Current** **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) The LAR is proposing to lower the TS LCO 3.4.16 limit to 0.10 µCi/gm DE I-131 for BVPS-2.

Table 4-4a BVPS-1 Secondary Coolant TS Iodine Activity Concentrations (µCi/gm)		
Item	**Proposed[1]** **0.10 µCi/gm DE I-131**	**Current** **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 4-4b BVPS-2 Secondary Coolant TS Iodine Activity Concentrations (µCi/gm)		
Item	**Proposed[1]** **0.05 µCi/gm DE I-131**	**Current** **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) The LAR is proposing to lower the TS LCO 3.7.13 limit to 0.05 µCi/gm DE I-131 for BVPS-2.

Table 4-5a Analysis Assumptions and Key Parameter Values BVPS Common Control Room		
Control Room Parameters		
Item	**Proposed**	**Current**
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 4-5b Analysis Assumptions and Key Parameter Values BVPS Emergency Response Facility / Technical Support Center		
ERF Parameters		
Item	**Proposed**	**Current**
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[1]	90%
Recirculation Filter Efficiency (HEPA)	100% (particulate)[1]	98%
Recirculation Filter Efficiency (Charcoal)	100% (elemental/organic iodine)[1]	90%
Note: (1) While these filters were not credited for dose reduction, the filter efficiencies presented here were conservatively assumed to maximize occupant dose due to filter bank shine.		

Table 4-6 **Analysis Assumptions and Key Parameter Values** **BVPS-1 and BVPS-2 Loss of Coolant Accident**		
Item	**Proposed**	**Current**
Containment Leakage Parameters		
Sprayed Fraction	60%	63%
Spray Period	77.4 sec to 4 days	85.4 sec to 4 days
Spray Removal Constants	Updated to account for containment volume, spray flow rate, and mass and energy release updates from NSAL-11-5.	Figures E-1 and E-2 of FENOC letter L-07-017, Accession No. ML070440341
Emergency Core Cooling System (ECCS) / Refueling Water Storage Tank (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	Impact from NSAL-11-5 is bounded by the CLB.	FENOC letter L-07-017, Accession No. ML070440341
Containment Vacuum System Parameters		
RCS TS Iodine and Noble Gas Activity Concentration	Table 3.2-1A as described in Section 3.2.1	Table 4-2a

Table 4-7 **Analysis Assumptions and Key Parameter Values** **BVPS-1 and BVPS-2 Control Rod Ejection Accident**		
Item	**Proposed**	**Current**
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Peaking Factor	1.70	1.75
Steam Releases per SG 2500 sec to 8 hours	 778,000 lbs	 776,000 lbs
Termination of Secondary System Environmental Releases	30 days	8 hours

Table 4-8a Analysis Assumptions and Key Parameter Values BVPS-1 Main Steam Line Break		
Item	**Proposed**	**Current**
Minimum Reactor Coolant Mass	345,097 lbm	340,711 lbm
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Updated as described in Section 3.2	
Reactor Coolant Equilibrium Iodine Appearance Rates		
Secondary Coolant TS Iodine Activity Concentration		

Table 4-8b Analysis Assumptions and Key Parameter Values BVPS-2 Main Steam Line Break		
Item	**Proposed**	**Current**
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	Updated as described in Section 3.2	
Reactor Coolant Equilibrium Iodine Appearance Rates		
Secondary Coolant TS Iodine Activity Concentration		

Table 4-9 Analysis Assumptions and Key Parameter Values BVPS-1 and BVPS-2 Steam Generator Tube Rupture		
Item	**Proposed**	**Current**
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Updated as described in Section 3.2	
Reactor Coolant TS Equilibrium Iodine Appearance Rates		
Secondary Coolant TS Iodine Activity Concentration		

Table 4-10 Analysis Assumptions and Key Parameter Values BVPS-1 and BVPS-2 Locked Rotor Accident		
Item	**Proposed**	**Current**
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 4-11 Analysis Assumptions and Key Parameter Values BVPS-1 and BVPS-2 Loss of AC Power		
Item	**Proposed**	**Current**
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Updated as described in Section 3.2	
Secondary Coolant TS Iodine Activity Concentration		

Table 4-12 Analysis Assumptions and Key Parameter Values BVPS-1 and BVPS-2 Fuel Handling Accident		
Item	**Proposed**	**Current**
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 4-13 Analysis Assumptions and Key Parameter Values BVPS-1 and BVPS-2 Small Line Break Outside Containment		
Item	**Proposed**	**Current**
Minimum Reactor Coolant Mass	341,331 lbm	340,711 lbm
Reactor Coolant TS Iodine and Noble Gas Activity Concentration	Updated as described in Section 3.2	
Reactor Coolant Equilibrium Iodine Appearance Rates		

Table 4-14a
2-Hour Integrated Exclusion Area Boundary Dose (TEDE)

Accident	Proposed (rem)(1,3)	Current (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(8)	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

Notes:

(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-hour 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 tube leakage of 2.1 gpm (for CLB) and 8.1 gpm (LAR) into the affected SG. LAR dose is less than 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.

(8) For BVPS-1, the dose from a postulated locked rotor accident is bounded by the BVPS-2 locked rotor accident.

Table 4-14b 30-Day Integrated Low Population Zone Dose (TEDE)			
Accident	**Proposed (rem)**	**Current (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(8)	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) Not applicable.

(2) LPZ Doses are based on the duration of the release.

(3) Not applicable.

(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 and LAR).

(5) Doses are based on the maximum allowable accident-induced tube 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.

(8) For BVPS-1, the dose from a postulated locked rotor accident is bounded by the BVPS-2 locked rotor accident.

Table 4-14c
30-Day Integrated Control Room Dose (TEDE)

Accident	Control Room Operator (rem)		
	Proposed	**Current**	**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
Fuel Handling Accident(6) BVPS-1(5) BVPS-2	 4.2 1.4	 2.36 1.4	5
Locked Rotor Accident(6)(7)	2.9	2.2	5
Loss of AC Power(6)	(Note 4)	(Note 4)	5
Small Line Break Outside Containment(6)	0.7	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 and 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 tube 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.

(7) For BVPS-1, the dose from a postulated Locked Rotor Accident is bounded by the BVPS-2 locked rotor accident.

Attachment 4

Generic Letter 95-05 Iodine Spiking Analysis
(4 pages follow)

FENOC has investigated the history of iodine spiking at BVPS-2 in support of an effort to lower the DE I-131 TS limit for the RCS to 0.10 microcuries per gram. This analysis was performed pursuant to NRC GL 95-05, Section 2.b.4, for plants reducing their DE I-131 limit to justify the iodine spiking assumptions in the accident analyses.

GL 95-05 requires that the spiking analysis be performed using the methodology described in "The Iodine Spike Release Rate during a Steam Generator Tube Rupture," J.P. Adams and C.L. Atwood, Nuclear Technology, Volume 94, page 361 (1991).

BVPS-2 RCS iodine values were reviewed starting with Cycle 1 data in 1987 and concluding with Cycle 21 data in the year 2019. Cycle 2 was selected as the starting point since the iodine data in Cycle 1 lacked the data requirements listed in the Adams and Atwood methodology.

A total of 13 reactor trips were identified for the 21 operating cycles that potentially met the requirements of Adams and Atwood. The necessary data could be found for 9 of the reactor trips (excluding the data from Cycle 1).

1. Sufficient steady-state power prior to the plant to ensure an adequate buildup of iodine. The specific criterion used was a minimum of 5 days at steady state power operation, resulting in a minimum of 35 percent of the steady state I-131 concentration. In nearly all cases, the steady-state power operation lasted several weeks to several months rather than the minimum of 5 days.
2. Knowledge of the steady-state iodine concentration.
3. Availability of at least one post-trip chemistry sample taken 2 to 6 hours after the plant trip.
4. No post-trip RCS perturbation (recriticality for example) prior to RCS sample.
5. Availability of all requisite transient information (purification flow, plant trip date and time, post-trip sample date and time).

Additionally, the purification flow rate had to be constant before and after the transient, as was stated in Adams and Atwood, Section II.C.

The nine plant trips that met all of the requirements of Adams and Atwood were then subject to analysis as described in their paper. The results are listed in the table below and the values are well within the bounds of the data set listed in Adams and Atwood. There are several cycles that operated defect free. The other cycles had from one to three fuel defects present. In several instances a negative iodine spike release rate is calculated based in the fact that the post-trip iodine value was lower than the steady state value. There are 5 of 9 reactor trips that result in an iodine spike following the reactor trip. In all five of these cases, the release rate spike factor was below the value of 500 assumed in the accident analyses.

The maximum normalized release rate was calculated to be 1.95E-2 Curies per hour per megawatt electric (Ci/hr*MWe). This release rate is far below the 95 percent confidence bound of the 90^{th} percentile, given by Adams and Atwood as 1.09 Ci/hr*MWe.

Therefore, using the methodology referenced by GL 95-05, the RCS DE I-131 data fully supports lowering the RCS DE I-131 TS limit to 0.10 microcuries per gram.

The BVPS-2 RCS DE I-131 data is below most industry values because the BVPS-2 fuel failure mechanisms (primarily grid-to-rod fretting) have placed most of the fuel failures in the peripheral regions of the core. The power factors in these peripheral regions are very low due to the use of low-leakage core designs. Therefore, although BVPS-2 has had several fuel failures, they have occurred in rods where there was relatively little generation of fission products.

Evaluation of BVPS Unit 2 Iodine Release Rate Data

Sample Date and Time	Power	I-131	DE-I131	R3(2)	R3(2)/P
	(percent)	(μCi/g)	(μCi/g)	(Ci/hour)	(Ci/hour*MWe)
END OF CYCLE 1	ZERO FUEL DEFECTS In CYCLE 1				
6/22/1990 19:11	93	1.17E-04	1.29E-03		
7/2/1990 13:30	0	9.54E-05	4.93E-04	-5.7648E-02	-6.8569E-05
END OF CYCLE 2	ZERO FUEL DEFECTS In CYCLE 2				
11/21/1991 9:20	99.4	7.16E-05	6.54E-04		
11/26/1991 17:25	0	6.07E-05	3.38E-04	-2.0134E-02	-2.2407E-05
END OF CYCLE 3	THREE FUEL DEFECTS In CYCLE 3				
1/28/1993 8:40	92	5.79E-04	1.47E-03		
1/30/1993 3:36	0	1.17E-01	1.55E-01	1.6242E+01	1.9529E-02
END OF CYCLE 4	ONE FUEL DEFECT In CYCLE 4				
5/26/1994 8:05	99.7	5.78E-04	3.37E-03		
6/1/1994 19:30	0	4.49E-02	6.80E-02	6.8235E+00	7.5708E-03
END OF CYCLE 5	ZERO FUEL DEFECTS In CYCLE 5				
8/10/1995 8:15	90	1.73E-04	1.16E-03		
8/13/1995 14:25	0	3.16E-02	3.67E-02	3.7716E+00	4.6357E-03
END OF CYCLE 6	TWO FUEL DEFECTS In CYCLE 6				
1/2/1997 8:30	94	1.43E-04	6.89E-04		
1/6/1997 8:48	0	1.49E-02	2.07E-02	2.1234E+00	2.4988E-03
3/13/1997 9:04	100	2.31E-04	8.94E-04		
3/19/1997 8:45	0	2.65E-02	3.34E-02	3.4432E+00	3.8089E-03
END OF CYCLE 7	TWO FUEL DEFECTS In CYCLE 7				
END OF CYCLE 8	THREE FUEL DEFECTS In CYCLE 8				
3/15/2001 10:04	100	7.14E-05	3.37E-04		
3/18/2001 0:30	0		1.58E-04	-1.2013E-02	-1.3289E-05
END OF CYCLE 9	ZERO FUEL DEFECTS In CYCLE 9				
END OF CYCLE 10	THREE FUEL DEFECTS In CYCLE 10				
END OF CYCLE 11	ONE FUEL DEFECT In CYCLE 11				
3/30/2006 9:20	100	3.69E-05	1.54E-04		
4/2/2006 16:27	0	2.46E-05	1.14E-04	-1.0206E-03	-1.1290E-06
END OF CYCLE 12	ZERO FUEL DEFECTS In CYCLE 12				
END OF CYCLE 13	ZERO FUEL DEFECTS In CYCLE 13				
END OF CYCLE 14	ZERO FUEL DEFECTS In CYCLE 14				
END OF CYCLE 15	ZERO FUEL DEFECTS In CYCLE 15				
END OF CYCLE 16	ZERO FUEL DEFECTS In CYCLE 16				

Evaluation of BVPS-2 Iodine Release Rate Data
(continued)

END OF CYCLE 17	ZERO FUEL DEFECTS In CYCLE 17
END OF CYCLE 18	ZERO FUEL DEFECTS In CYCLE 18
END OF CYCLE 19	ZERO FUEL DEFECTS In CYCLE 19
END OF CYCLE 20	ZERO FUEL DEFECTS In CYCLE 20
CYCLE 21 ongoing	
Note: All values of R3(2)/P are < 1.09; therefore, BVPS-2 values are conservative.	