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L-06-061

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

Subject: **Beaver Valley Power Station, Unit No. 1**
BV-1 Docket No. 50-334, License No. DPR-66
Clarifications / Corrections to License Amendment No. 273 Steam Generator Replacement for BVPS Unit No. 1

On February 9, 2006, NRC issued Amendment No. 273 Steam Generator Replacement for Beaver Valley Power Station (BVPS) Unit No. 1 (Reference 1). This amendment was requested by FENOC application letter L-05-069 dated April 13, 2005 (Reference 2). This amendment approved the Technical Specification changes necessary for operation of BVPS Unit No. 1 with the replacement steam generators.

Based on the FENOC review of the amendment and its associated Safety Evaluation, clarifications / corrections are provided for your consideration in Attachment 1, and a marked-up copy of the affected pages are provided in Attachment 2. FENOC believes these clarifications / corrections do not, in any way, invalidate the conclusions in the Safety Evaluation.

No new regulatory commitments are contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager – FENOC Fleet Licensing, at (330) 315-7243.

Sincerely,

James H. Lash

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Attachments:
1. Clarifications / Corrections to License Amendment No. 273 Safety Evaluation
2. Marked-up Pages from License Amendment No. 273 Denoting Clarifications / Corrections

References:
1. NRC Amendment No. 273, Beaver Valley Power Station Unit No. 1 (BVPS-1) – Issuance of Amendment Re: Steam Generator Replacement (TAC No. MC6725), dated February 9, 2006
2. FENOC Letter L-05-069, License Amendment Request 320, dated April 13, 2005.

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Attachment 1 of L-06-061

Clarifications / Corrections to License Amendment No. 273 Safety Evaluation

1. 1st page of the Safety Evaluation (SE), bottom paragraph, the title for RG 1.183 should be "**Alternative** ..." vs. "Alternate ...". This would be consistent with the title provided on page 4 of the SE for RG 1.183.

2. Page 8, Section 3.8 Control Rod Drive Mechanisms (CRDMs) (LAR Section 5.4), should be LAR Section **5.3.16**

3. Page 41, 2nd paragraph and Page 42, 3rd paragraph - discusses Reactor Coolant System radioiodine inventory is at its maximum value of 21 uci/gm permitted by the Technical Specifications (TSs). This statement is incorrect since the current BVPS-1 TSs permit only 6 uci/gm. The radiological analyses conservatively use the BVPS-2 coolant TSs at uprated conditions. The BVPS Extended Power Uprate (EPU) application includes a request to update the BVPS-1 coolant activity TSs to make it similar to BVPS-2.

4. Page 42, 2nd paragraph - states that the licensee provided dose consequences resulting from an operational response analysis (ORA) case in an August 26, 2005 response to an NRC RAI that showed that the licensing basis case was more conservative than the ORA. Need to clarify that the licensee's August 26, 2005 response was updated by Letter L-05-195, dated December 6, 2005 - some of the inputs were changed but the conclusion in the SE remains the same.

5. Page 42, 4th paragraph, 1st sentence – states that "The iodine activity from the break flow through the ruptured SG is assumed to be directly released to the environment and partitioning of iodine is not credited". This statement should be clarified. The dose model assumes that the iodine in the "flashed portion" of the break flow is released directly to the environment without partitioning; i.e., not "all" of the iodine activity in the break flow is released.

6. Page 43, first paragraph under Section 4.1.3 (LRA) - indicates that following a Locked Rotor Accident (LRA), Safety Injection (SI) is actuated. For LRA, assumptions include a reactor trip and Loss of Offsite Power (but not SI actuation). See item 7 below.

7. Page 43, last paragraph - states that the LRA "is not expected to result in a SIS" (this statement contradicts the paragraph identified in item 6 above). The SER then states that "Therefore the licensee assumes no isolation of control room." SIS will not initiate control room isolation, therefore this sentence should be deleted.

8. Page 43, 3rd paragraph - discusses that an LRA will cause 20% Failed Fuel (FF) and that a radial peaking factor of 1.75 was applied. It also states that these parameters are the "current" design bases of BVPS-1. The analyses uses the uprated conditions, which is conservative with respect to the current design basis. (The current design basis analysis assumes 18% FF and does not apply a peaking factor.)

9. Page 44, last paragraph – states that the Steam Line Break Accident (SLBA) results in a break mass flow rate of 16.79 lbm /sec with a 37% flash, and that these parameters are consistent with current design basis as reported in the BVPS-1 UFSAR. The analyses uses the uprated conditions, which is conservative with respect to the current design

basis. (The current design basis analysis assumes a mass flow rate of 16.2 lbm /sec and a flash fraction of 38%.)

10. Page 45, last paragraph – includes "CREA" among the accidents that are maintained at normal operation ventilation for the duration of the accident. This is incorrect since the analysis assumes Control Room Emergency Ventilation System operational by T=30 minutes after a CREA (Control Rod Ejection Accident).

11. Page 46, last paragraph – states that "Although the amendment request is for BVPS-1, the licensee compared X/Q values for BVPS-1 and 2 and used the more limiting control room, EAB and LPZ X/Q values in each dose assessment associated with the RSG LAR." This should be corrected since the analysis used the more limiting X/Q values for only the bounding analyses, i.e., only for the CREA, the LRA / LACP (Loss of AC Power) and the SLB (Small Line Break) outside containment (as noted correctly in Table 3 of the SE; the MSLB (Main Steam Line Break) and SGTR (Steam Generator Tube Rupture) are based on Unit specific X/Qs).

12. Page 47, last paragraph – same comment as item 11, above

Attachment 2 of L-06-061

Marked-up Pages from License Amendment No. 273 Safety Evaluation Denoting Clarifications / Corrections

(See attached affected pages)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 273 TO FACILITY OPERATING LICENSE NO. DPR-66

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

BEAVER VALLEY POWER STATION, UNIT NO. 1 (BVPS-1)

DOCKET NO. 50-334

1.0 INTRODUCTION

By application dated April 13, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051080573), as supplemented by letters dated August 26, October 28 and 31, November 18, and December 6 and 16, 2005 (ADAMS Accession Nos. ML052430345, ML053050300, ML053110142, ML053290139, ML053460239, and ML053560175), FirstEnergy Nuclear Operating Company (FENOC, the licensee), requested changes to the Technical Specifications (TSs) for BVPS-1. The supplements dated August 26, October 28 and 31, November 18, and December 6 and 16, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 21, 2005 (70 FR 35737).

The proposed changes would revise the TSs to allow replacement of the BVPS-1 steam generators (SGs) from the current Westinghouse Model 51 SGs to the new Westinghouse Model 54F SGs. These changes include revising the fuel assembly-specific departure from nucleate boiling ratios and correlations, modifying the Overtemperature ΔT and Overpower ΔT equations, revising the SG water level low-low and high-high setpoints, revising the SG secondary side level in Modes 4 and 5, revising the SG TSs to reflect the replacement SGs and remove TS requirements that are no longer applicable to the new SGs, revising the required charging pump discharge pressure for reactor coolant pump seal injection flow, raising the accumulator pressure, and adding WCAP-14565-P-A (VIPRE methodology) and WCAP-15025-P-A (WRB-2M correlation) to the list of Nuclear Regulatory Commission (NRC)-approved methodologies listed in TS 6.9.5. These WCAPs have previously been approved by the NRC. The amendment also would approve an expansion of the selective implementation of the alternate source term methodology in accordance with Regulatory Guide (RG) 1.183, "~~Alternate~~ Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and would approve use of the 1979 ANS Decay Heat + 2σ model for mass and energy releases for a main steam line break (MSLB) outside containment.

Alternative

staff found the fuel design remains acceptable, based on the results of the safety analyses addressed in Section 3.9 below.

3.7 Fuel Thermal Hydraulics Design (LAR Section 6.1)

The licensee proposed to use the rated thermal design procedure (RTDP) to perform statistical core thermal-hydraulic analyses, where applicable. Unlike the deterministic method, where the uncertainties of various plant and operating parameters are assumed simultaneously at their worst uncertainty limits in the safety analyses, the RTDP methodology statistically accounts for the system uncertainties in plant operating parameters, fabrication parameters, nuclear and thermal parameters, as well as the departure from nucleate boiling (DNB) correlation and computer codes uncertainties. The RTDP methodology establishes a design DNB ratio (DNBR) limit that statistically accounts for the effects of the key parameters on DNB. The RTDP methodology is documented in WCAP-11397-P-A (Reference 11). The DNB design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95 percent at a 95 percent confidence level for any Condition I or II event. Since the parameter uncertainties are considered in determining the RTDP design limit, the plant safety analyses are performed using input parameters at their nominal values. The DNBR margin/penalty summary for transients using RTDP is given in Table 6.1-2 of this LAR. The standard thermal design procedure (STDP) was used for those analyses where RTDP is not applicable. The DNBR margin/penalty summary for transients using STDP is given in Table 6.1-3 of this LAR. In addition, the licensee used the WRB-1, W-3, and WRB-2 DNB correlations, consistent with the analysis of record. The licensee requested adoption of the WRB-2M correlation as part of this LAR. Further discussion addressing this request is found in Section 3.9.2 of this SE. The thermal hydraulic evaluation at EPU conditions for BVPS-1 showed that sufficient DNB margin is available using the different DNB correlations at EPU conditions so that the licensing basis acceptance criteria continue to be met. The NRC staff finds the licensee's application of RTDP methodology in these analyses to be acceptable since the licensee satisfied the conditions set on the RTDP methodology for application at BVPS-1. The NRC staff finds that the use of the WRB-2M correlation is acceptable on the fuel designs stated in Section 3.6 of this SE, since the re-analyzed accidents, as stated in Section 3.9, demonstrate that the DNB safety analysis limit (SAL) was not exceeded.

5.3.16

3.8 Control Rod Drive Mechanisms (CRDMs) (LAR Section 5.4)

The licensee evaluated the rupture of a CRDM housing pursuant to the 10 CFR 50.59 screening process (Reference 12) and determined there were no TS changes required for the BVPS-1 RSG LAR (Reference 1). The analysis performed at EPU conditions was evaluated for operation with RSGs at current power level for BVPS-1. The licensee concluded that operation of BVPS-1 at its current power level with RSGs is bounded by the EPU analyses. The rod ejection analysis confirmed that the current criteria in the BVPS-1 UFSAR continue to be met. Additionally, the existing analysis for the CRDMs meet the American Society for Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) pressure requirements. The NRC staff agrees that operation at the current licensed power level with the RSGs remains bounded by the EPU analyses and that no changes are required to the CRDMs that would affect the system's design function. Therefore, the NRC staff finds that the licensee's 10 CFR 50.59 evaluation for CRDMs is acceptable with respect to the proposed RSG program.

per the BVPS-1 coolant activity Tech Spec change request submitted with the EPU application (LAR 302), which is conservative with respect to the current BVPS-1 RCS activity

The licensee stated that no fuel damage is postulated to occur because of an MSLB. The licensee stated in the BVPS-1 UFSAR that the design basis with regard to DNB is met for any steam line rupture, assuming the most reactive rod cluster control assembly is stuck in its fully withdrawn position. The NRC staff previously accepted the DNB analysis in the BVPS-1 UFSAR as a design basis and this assumption is not impacted by the RSGs or implementation of the AST. Consistent with the guidance provided in RG 1.183, the licensee assumed the released activity is the maximum reactor coolant activity specified in the BVPS-1 TSs since there is no postulated fuel damage associated with this event.

Two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the RCS radioiodine inventory is at the maximum value (21 µCi/gm) ~~permitted by the~~ TSs. The second case assumes that the event initiates a co-incident radioiodine spike. Radioiodine is released from the fuel to the RCS at a rate 500 times the normal radioiodine appearance rate for a duration of 4 hours. The iodine spiking duration of 4 hours is the current design basis in the BVPS-1 UFSAR and this value was reviewed and accepted by the NRC staff previously in Reference 23 as a design basis. The RSGs or the expanded selective implementation the AST does not impact the iodine spiking duration.

Leakage from the RCS to the SGs is assumed to be the maximum value permitted by the TSs (150 gallons per day (gpd) per SG). The maximum TS limit for all three SGs is 450 gpd. The release from the faulted SG due to primary-to-secondary leakage continues for 19 hours until the RHR system brings the primary coolant temperature down to 212 °F. The primary coolant leakage into the faulted SG is assumed to immediately flash to steam and be released to the environment without holdup or dilution. The leakage in the intact SGs mixes with the secondary coolant bulk water and is released through the MSSVs and ADVs at the assumed steaming rate. This steaming from the intact SGs is assumed to continue for 8 hours until shutdown cooling is initiated via operation of the RHR system. The licensee assumed an iodine partitioning factor of 100 in the intact SGs, and assumed no iodine partitioning in the faulted SG.

The licensee conservatively assumed manual initiation of the control room emergency ventilation system (CREVS) at 30 minutes following the MSLB event to pressurize the CR. The CR is purged at a rate of 16,200 cubic feet per minute (cfm) for a period of 30 minutes beginning at 24 hours following the MSLB event (see Section 4.2, "Control Room Habitability").

The licensee re-evaluated the radiological consequences resulting from the postulated MSLB accident for operation with the RSGs and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP, Section 15.0.1. The NRC staff's audit found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 4 and the licensee's calculated dose results are given in Table 1. The NRC staff performed an independent confirmatory dose calculation to verify the licensee's results. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the MSLB meet the applicable accident dose criteria and are therefore, acceptable.

4.1.2 SGTR

The accident considered is the complete severance of a single tube in one of the SGs, resulting in the transfer of RCS water to the ruptured SG. The primary-to-secondary break flow through the ruptured tube following an SGTR results in radioactive contamination of the secondary system. For this accident scenario, a reactor trip occurs, SI actuates, and a LOOP occurs concurrently with the reactor trip. Because the LOOP renders the main condenser unavailable, the plant is cooled down by release of steam to the environment.

The conservatism of the licensing basis thermal hydraulic analysis model, which includes a 30-minute isolation time, is supported by a supplemental BVPS-1 SGTR operational response analysis (ORA) performed by the licensee. The supplemental SGTR ORA included consideration of single active failures, the timing of operator actions in accordance with plant emergency operating procedures, and demonstrated performance during simulator exercises. The licensee stated, in an August 26, 2005 response to the NRC staff's RAI that the ORA and the radiological consequence analysis confirmed that dose estimates using the licensing basis thermal hydraulic analysis model are conservative and bound the dose estimates developed utilizing the thermal hydraulic input data based on the operational response case.
and updated in a December 6, 2005

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR and this event is described in the BVPS-1 UFSAR, Section 14.2.4, "Steam Generator Tube Rupture." Two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the RCS radioiodine inventory is at the maximum value (21 μCi/gm) ~~permitted by the~~ BVPS-1 TSs. The second case assumes the event initiates a co-incident radioiodine spike. Radioiodine is released from the fuel to the RCS at a rate 335 times the normal radioiodine appearance rate for 4 hours. As stated in Section 4.1.1 above, the iodine spiking duration of 4 hours is assumed. Primary-to-secondary leakage is assumed to be 150 gpd into the bulk water of the ruptured SG and 300 gpd total into the bulk water of the two intact SGs as permitted by the BVPS-1 TSs.
RCS activity
per the BVPS-1 coolant activity Tech Spec change request submitted with the EPU application (LAR 302) which is conservative with respect to the current

The iodine activity from the break flow through the ruptured SG is assumed to be directly released to the environment and partitioning of iodine is not credited. The radionuclides in the intact SGs bulk water are assumed to become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient. The licensee assumed that the radionuclide concentration in the SG is partitioned such that 1 percent of the radionuclides in the unaffected SGs bulk water enter the vapor space and are released to the environment. The steam release from the unaffected SGs continues for approximately 8 hours until the RHR shutdown cooling system can be used to complete the cooldown.
flashed portion of the

The licensee claimed no credit for fission product removal by the CREVS following an SGTR event and assumed the control room is maintained in normal ventilation mode. Following termination of the environmental release at 8 hours, the CR is purged at a rate of 16,200 cfm for a period of 30 minutes (see Section 4.2, "Control Room Habitability").

The radiological consequence analysis of this event was performed at an EPU reactor core power level of 2918 MWt which bounds the current licensed reactor core power level of 2689 MWt. The licensee re-evaluated the radiological consequences resulting from the postulated SGTR accident for operation with the RSGs, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP, Section 15.0.1. The NRC staff's audit found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the BVPS-1 UFSAR as the design bases. The assumptions found acceptable to the NRC staff are presented in Table 5 and the licensee's calculated dose results are given in Table 1. The NRC staff performed an independent confirmatory dose calculation to verify the licensee's results. The EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident are found to meet the applicable accident dose criteria and are therefore, acceptable.

4.1.3 LRA

The accident considered is the instantaneous seizure of an RCP rotor which causes a rapid reduction in the flow through the affected RCS loop. For the accident scenario, a reactor trip occurs, ~~SI actuates~~ and a LOOP occurs concurrently with the reactor trip. The flow imbalance creates localized temperature and pressure changes in the core. The radiological consequences are due to leakage of the radioactive reactor primary coolant to the SGs and from there to the environment. Because the LOOP renders the main condenser unavailable, the plant is cooled down by release of steam to the environment through ADVs and MSSVs. The releases to the environment are assumed to continue for 8 hours, at which time shutdown cooling is initiated by via operation of the RHR system. Appendix G of RG 1.183 identifies acceptable radiological analysis assumptions for an LRA and this event is described in the BVPS-1 UFSAR, Section 14.2.9, "Complete Loss of Forced Reactor Coolant Flow."

performed at the uprated conditions which is conservative with respect to

The licensee assumed that the RCP was inoperable and loss of primary coolant circulation may result in as much as 20 percent of the core fuel rods experiencing DNB. This will cause fuel cladding damage, and release of the damaged fuel gap activity into the RCS. No fuel melting is assumed. A radial peaking factor of 1.75 was applied to the gap activity. These parameters are the current design bases in the BVPS-1 UFSAR and they are not impacted by the SG replacement or implementation of the AST. The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage of 450 gpd for all three SGs for 8 hours. The licensee assumed that this leakage mixes with the bulk water of the SG's secondary side and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient.

The tubes in the SGs would remain covered by the bulk water. The licensee assumed that the radionuclide concentration in the SG is partitioned such that 1 percent of the radionuclides in the bulk water of the SGs enter the vapor space and is released to the environment consistent with guidance provided in RG 1.183. The activity releases associated with the release of secondary coolant through steaming and primary coolant through primary-to-secondary leakage and steaming at TS limits is insignificant compared to the activity in the gap release from the 20-percent damaged fuel.

The LRA event is not expected to result in an SIS. ~~Therefore, the licensee assumed no isolation of the control room.~~ The analyses for these events assume that the control room remains in its normal operation mode with a normal outside air intake of 500 cfm during the duration of these events (see Section 3.2, "Control Room Habitability").

The licensee re-evaluated the radiological consequences resulting from the postulated LRA using the RSGs and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in the SRP, Section 15.0.1. The radiological consequence of this event was performed at an EPU reactor core power level of 2918 MWt, which bounds the current licensed reactor core power level of 2689 MWt.

The NRC staff's audit found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the BVPS-1 UFSAR as design bases. The assumptions found acceptable to the NRC staff are presented in Table 6 and the licensee's calculated dose results are given in Table 1. The NRC staff performed an independent confirmatory dose calculation to verify the licensee's results. The EAB, LPZ, and CR doses estimated by the licensee for the LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

4.1.4 LACP

The LACP involves the loss of AC power to plant auxiliaries. Major plant loads that would be lost include the RCP, main feedwater pump, main circulating water system, and main condenser. A reactor trip will occur. With the main condenser unavailable, the plant is cooled down by release of steam to the environment via ADVs and MSSVs. The licensee stated, and the NRC staff agrees, that the LACP event is similar to the LRA, with the exception that the LRA event results in fuel cladding damage and associated release of gap activity, whereas the LACP event involves no core fuel damage. Therefore, the radiological consequences resulting from the LRA event bounds the LACP event.

4.1.5 SLBA

The SLBA event postulates the break of a 2-inch RCS letdown line in the auxiliary building outside of the containment. The letdown line is the largest piping that carries RCS fluid outside containment. A rupture of the letdown line provides a release path for the primary coolant to the environment through the auxiliary building ventilation vent. The radiological consequence analysis of this event was performed at an EPU reactor core power level of 2918 MWt, which bounds the current licensed reactor core power level of 2689 MWt.

The licensee's analysis assumed that no fuel failure results from the letdown line break which is consistent with the current licensing basis in BVPS-1 UFSAR. The radioactivity in the RCS was initially at the equilibrium iodine TS limit of 0.35 μCi/gm dose equivalent I-131 (DEI-131). The consideration of the equilibrium iodine TS limit of 0.35 μCi/gm DEI-131 is consistent with the review procedure provided in SRP, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment." Neither RG 1.183 nor SRP 15.0.1 addresses the SLB event as a DBA.

The accident was assumed to cause the iodine concentration to spike by a factor of 500 times the equilibrium iodine appearance rate. A total of 15,110 lbm of RCS fluid was assumed released through the break, based on a break mass flow rate of 16.79 lbm per second for 15 minutes. The licensee assumed 37 percent of the break flow would flash, based on a constant enthalpy process. These parameters are consistent with the current design basis in

performed to the uprated conditions which is conservative with respect to the BVPS-1 UFSAR. Neither the implementation of the AST nor RSGs impact these parameters. Additional RCS radioactivity was assumed released to the environment through SG tube leakage and secondary system steaming to cool down the plant. The iodine activity in the break flow is assumed to be airborne in proportion to the flash fraction, whereas the noble gases are assumed to be airborne and released to the environment without decontamination or holdup.

The SLBA event is not expected to result in an SI signal. Therefore, the licensee assumed no isolation of the control room. The analyses for this event assume that the control room remains in its normal operation mode with a normal outside air intake of 500 cfm for the duration of this event (see Section 4.2 of this SE, "Control Room Habitability").

The NRC staff reviewed the information provided in the licensee's submittal and supplements and the BVPS-1 UFSAR and also performed an independent calculation that confirmed the licensee's dose results. RG 1.183 does not address an SLBA outside containment. The licensee's analysis used assumptions and inputs that follow the guidance provided for similar DBAs in RG 1.183 (LRA and CREA) and the SRP, Section 15.6.2. Since there are no specific dose acceptance criteria given in the SRP, Section 15.0.1, for the letdown line break, the licensee used the most limiting dose acceptance criteria for any DBA listed in RG 1.183 (2.5 rem TEDE in the EAB and LPZ and 5 rem TEDE in the CR). It is also consistent with the dose guideline provided in the SRP, Section 15.6.2, as a "small fraction" (i.e., 10 percent) of 10 CFR Part 100.

The NRC staff's audit found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the BVPS-1 UFSAR as design bases. The assumptions found acceptable to the NRC staff are presented in Table 7 and the licensee's calculated dose results are given in Table 1. The EAB, LPZ, and CR doses estimated by the licensee for the SLBA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

4.2 Control Room Habitability

The BVPS-1 control room habitability was previously evaluated and found acceptable by the NRC staff in Reference 23 for the LOCA and CREA, which would be bounding for all DBAs. However, the control room habitability evaluation is repeated here for the MSLB, SGTR, LRA, LACP, and SLBA accidents for completeness.

The BVPS-1 and 2 control rooms are located within a common control room envelope. The joint control room is served by two ventilation intakes, one for BVPS-1, and the other for BVPS-2. These air intakes are used for both the normal as well as emergency mode operations. During normal plant operation, both ventilation intakes provide a total supply of 500 cfm of unfiltered outside makeup air. For the ~~CREA in Reference 23, and the~~ SGTR, LRA, LACP, and SLBA in the RSG LAR, the licensee assumed that the control room is maintained in normal ventilation mode without activating the CREVS during the entire duration of these accidents. For BVPS-1 emergency power is provided to the normal control room ventilation system, including all ventilation system components that are required to support control room operation in the recirculation mode. Therefore, the NRC staff finds that it is acceptable to credit the normal ventilation system for post-accident control room purging at the times specified in the accident analyses.

For the MSLB accident, the licensee has taken credit for operation of the CREVS and assumed manual initiation of the CREVS at 30 minutes following the accident. The CREVS pressurizes the control room. Once CREVS starts, the filtered intake flow rate is expected to vary between 600 and 1030 cfm. Sensitivity analyses by the licensee have shown that the lower flow rate is generally more limiting since the higher flow rate results in a greater dilution of control room atmosphere radioactivity concentrations. The licensee used 600 cfm CREVS flow rate in its radiological consequence analyses including the LOCA and CREA in Reference 23. The licensee assumed the control room unfiltered air inleakage of 300 cfm during the control room isolation (recirculation) mode (time the control room is isolated from 77 seconds to 30 minutes). For the emergency pressurized mode (time the control room is pressurized from 30 minutes to 30 days), the licensee assumed the control room unfiltered air inleakage of 30 cfm. The licensee based these leakage values on the result of tracer gas testing in the isolated recirculation and pressurized modes. An unfiltered inleakage of 10 cfm due to ingress and access was added to the mean values for the tracer gas measurements to arrive at the unfiltered inleakage values assumed in the dose calculations.

The licensee performed tracer gas measurements of the unfiltered inleakage to the control room in both the isolated (recirculation) and emergency pressurized modes in May of 2001, using the methodology described in American Society for Testing and Materials Standard E2029, "Standard Test Method for Volumetric and Mass Flow Rate Measurement Using Tracer Gas Dilution." The tracer gas test results were zero cfm (no leakage) for BVPS-1 pressurization mode and 267 cfm with 10 cfm uncertainty for the recirculation mode. The NRC staff finds unfiltered air inleakage values assumed by the licensee to be acceptable based on tracer gas testing results. The CREVS intake filters are assumed to be 99 percent efficient for particulates and 98 percent efficient for elemental and organic iodine species. The BVPS-1 control room unfiltered air inleakage values and CREVS filter efficiencies were previously accepted by the NRC staff in Reference 23.

4.3 Atmospheric Dispersion

The licensee generated new atmospheric dispersion factors (χ/Q values) using the NRC-sponsored ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") to evaluate the impact of the BVPS-1 and 2 ventilation vent and BVPS-1 MSLB point releases on the BVPS-1 and 2 control rooms. These χ/Q values represent a change from the χ/Q values used in the current BVPS-1 and 2 UFSAR, Chapter 14, accident analysis. The licensee used previously approved χ/Q values to assess the dose for a postulated release from the main steamline relief valves to the BVPS-1 and 2 air intakes and to perform dose assessments for the BVPS EAB and LPZ. Although this amendment request is for BVPS-1, the licensee compared the χ/Q values for BVPS-1 and 2 and used the more limiting control room, EAB and LPZ χ/Q values ~~in each dose assessment associated with the RG LAR.~~

for the events noted in Table 3 that were analyzed using bounding parameters and were thus applicable to both units.

4.3.1 Meteorological Data

The licensee generated new control room χ/Q values for postulated releases from the BVPS-1 and 2 ventilation vents and BVPS-1 MSLB point using site meteorological data collected from 1990–1994. The licensee previously provided these data and the NRC staff reviewed and discussed these data in the SE associated with BVPS-1 and 2, Reference 23. Based on the meteorological measurements program and meteorological database review described in the SE associated with Reference 23, the NRC staff has concluded that the 1990–1994 site meteorological database provides an acceptable basis for making atmospheric dispersion estimates for use in support of the RSG LAR.

4.3.2 Control Room Atmospheric Dispersion Factors

The licensee calculated new control room χ/Q values for one new release point, the southeast corner of the turbine building for the MSL break dose assessment, and revised control room χ/Q values for the BVPS-1 and 2 ventilation vents. These new and revised control room χ/Q values were calculated using the ARCON96 computer code and guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." The licensee executed ARCON96 using the 1990–1994 onsite hourly 10.7-meter and 45.7-meter wind data and stability class determined from the temperature difference measured between the 45.7-meter and 10.7-meter levels. All releases were modeled as point sources using the ARCON96 ground-level release mode option. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the BVPS site. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and NRC staff practice. In addition, the NRC staff performed a check of the resulting atmospheric dispersion estimates by running the ARCON96 computer code and obtained similar results.

The licensee used previously approved χ/Q values for the control room dose assessment for postulated releases from the main steamline relief valves. These χ/Q values are discussed in the SE associated with Reference 23.

For the reasons cited above, the NRC staff has concluded that the control room χ/Q values presented in Table 2 are acceptable for use in the DBA assessments described in this SE. For all release pathways, postulated releases from BVPS-1 to the BVPS-1 control room air intake were the most limiting cases.

4.3.3 EAB and LPZ Atmospheric Dispersion Factors

The licensee used existing χ/Q values that were accepted by the NRC staff in a previous licensing proceeding to evaluate the impact of the BVPS-1 and 2 postulated releases to the EAB and LPZ. Although this amendment request is for BVPS-1, the licensee compared the EAB and LPZ χ/Q values for BVPS-1 and 2 and used the more limiting χ/Q values in the dose assessment ~~discussed above~~. Based on the review described in the SE associated with Reference 23 and a review of the licensee's use of these χ/Q values in the RSG LAR, the NRC
for events identified in Table 3 that were analyzed using bounding parameters and were thus applicable to both units.