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December 30, 2005
L-05-204

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

Subject: Beaver Valley Power Station, Unit Nos. 1 and 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Additional Information Regarding Dose Consequence Analysis in Support of License Amendment Request Nos. 302 and 173

On October 4, 2004, FirstEnergy Nuclear Operating Company (FENOC) submitted License Amendment Request (LAR) Nos. 302 and 173 by letter L-04-125 (Reference 1). This submittal requested an Extended Power Uprate (EPU) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 and is known as the EPU LAR.

On April 13, 2005, FENOC submitted LAR No. 320 for BVPS Unit No. 1 by letter L-05-069 (Reference 2). This submittal requested the Technical Specification changes necessary for operation of BVPS Unit No. 1 with the replacement steam generators and is known as the RSG LAR.

The information provided in this submittal was discussed during the NRC dose assessment audit conducted at BVPS on November 29-30, 2005, and is provided in support of the EPU LAR. Enclosure 1 contains the revised thermal-hydraulic analysis which provided input for the BVPS Unit No. 2 Steam Generator Tube Rupture (SGTR) radiological dose assessment. The revised analysis provides additional operator action time for isolation of a failed open steam generator atmospheric dump valve (ADV) on the ruptured steam generator.

Enclosure 2 contains updated information for the BVPS Unit No. 1 Main Steam Line Break (MSLB) radiological assessment. This updated information is intended to ensure consistency with that provided in the RSG LAR (Reference 2). Updated radiological assessments are also being provided for the BVPS Unit No. 2 MSLB and the BVPS Unit No. 2 SGTR events. In addition, Enclosure 2 reflects the removal of details regarding the Waste Gas System Rupture (WGSR) radiological assessment, which has been revised under 10 CFR 50.59 to demonstrate compliance with current licensing basis dose consequence criteria.

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The additional information provided by this transmittal has no impact on the proposed Technical Specification changes nor does it negatively impact the no significant hazards consideration, transmitted by References 1 or 2.

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.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 30, 2005.

Sincerely,

James H. Lash

Enclosures:

1. Updated Thermal-Hydraulic Analysis for Radiological Dose Assessment of the BVPS-2 Steam Generator Tube Rupture (SGTR) Event for EPU
2. Updated Radiological Assessments for the BVPS-1 & 2 Main Steam Line Break (MSLB) and the BVPS-2 Steam Generator Tube Rupture (SGTR) for the EPU LAR

References:

1. FENOC Letter L-04-125, License Amendment Requests 302 and 173, dated October 4, 2004.
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)

Enclosure 1 of L-05-204

Updated Thermal-Hydraulic Analysis for Radiological Dose Assessment of the BVPS-2 Steam Generator Tube Rupture (SGTR) Event for EPU

Section 5.4.3 "BVPS-2 Thermal and Hydraulic Analysis for Offsite Radiological Consequences" of the Extended Power Uprate (EPU) License Amendment Request (LAR) Licensing Report provides the thermal-hydraulic analysis for the Steam Generator Tube Rupture (SGTR) event. The SGTR evaluation has been updated and a summary of the changes to Section 5.4.3 of the Licensing Report is provided below:

- The BVPS-2 SGTR analysis for EPU conditions included cases to provide thermal-hydraulic tube rupture data for use in radiological dose consequence analysis. The sequence of events for offsite dose analysis for BVPS-2 is shown in Table 5.4.3-1 of the Licensing Report. The sequence of events table shows that termination of the event (i.e., primary-to-secondary break flow terminated) occurs at 3932 seconds versus 3160 seconds. The termination time used in the radiological dose consequence analysis was increased to allow additional operator action time for the operator to isolate the failed-open steam generator atmospheric dump valve (ADV) on the ruptured steam generator. Figures 5.4.3-1 through 5.4.3-12 of the Licensing Report show the response of the various parameters for the BVPS-2 SGTR offsite radiation dose analysis.

- The changes in the mass releases and total mass flow for BVPS-2 at the EPU conditions are due to the revised sequence of events for operator action times and are shown in Table 5.4.3-2 of the Licensing Report.

Revised Section 5.4.3 "BVPS-2 Thermal and Hydraulic Analysis for Offsite Radiological Consequences" of the EPU Licensing Report, including Figures 5.4.3-1 through 5.4.3-12, is attached. Text changes are provided in bold and highlighted.

5.4.3 BVPS-2 Thermal and Hydraulic Analysis for Offsite Radiological Consequences

5.4.3.1 Introduction

The Steam Generator Tube Rupture (SGTR) analyses are performed for BVPS-2 using the analysis methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in Safety Evaluation Reports (SERs) dated December 17, 1985 and March 30, 1987. The methodology was developed for use with the LOFTTR2 program, an updated version of the LOFTTR1 program. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in References 1 and 2. This is the same methodology employed in the most recent analyses performed by Westinghouse for BVPS-2, documented in WCAP-12737 (Reference 3).

In Section 5.4.2, it was determined that the SGTR would not result in water relief. Since this has been confirmed, a SGTR thermal and hydraulic analysis is performed to provide input to the radiological consequences analysis for BVPS-2. The thermal and hydraulic analysis will consider the limiting single failure which maximizes the primary-to-secondary break flow, flashed break flow, and steam releases to the environment.

The mass releases are calculated with the LOFTTR2 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases from and feedwater flows to the intact and ruptured steam generators are determined from a mass and energy balance using the calculated reactor coolant system (RCS) and steam generator conditions at the time of leakage termination. The mass release information is used to calculate the radiation doses at the site boundary and low population zone and to the operators in the control room. The thermal and hydraulic analysis is performed using the LOFTTR2 program and the methodology developed in References 1 and 2, and using the plant specific parameters for BVPS-2. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

5.4.3.2 Input Parameters and Assumptions

The thermal-hydraulic analysis, which determines the offsite dose mass releases, models the plant operating at the higher end of the T_{avg} window, since a higher operating temperature results in increased steaming from the ruptured steam generator and a higher fraction of the break flow flashing to steam inside the ruptured steam generator. The analysis assumes that the plant is operating with the feedwater temperature at the higher end of the temperature window, since this is determined to result in slightly lower secondary mass. No tube plugging is assumed in the analysis as this maximizes heat transfer to the ruptured steam generator. A high heat transfer rate during the transient maximizes the amount of mass released from the steam generator due to steaming.

Design Basis Accident

The design basis accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold

side of the steam generator results in higher primary-to-secondary leakage than a break on the hot side of the steam generator, as determined by Reference 1. However, as indicated subsequently, the break flow flashing fraction is conservatively calculated assuming that all of the break flow comes from the hot leg side of the steam generator. The combination of these conservative assumptions regarding the break location results in a very conservative calculation of the offsite radiation doses. It is also assumed that loss-of-offsite power occurs at the time of reactor trip, and the highest worth control assembly is assumed to be stuck in its fully withdrawn position at reactor trip. Due to the assumed loss-of-offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator atmospheric steam dump valves (ASDVs). After reactor trip and loss-of-offsite power, the RCPs begin to coast down.

Based on the information in Reference 2, the most limiting single failure with respect to offsite doses is a failed open ASDV on the steam generator with the ruptured tube. Failure of this ASDV will cause an uncontrolled depressurization of the steam generator, which will increase primary-to-secondary leakage and the mass release to the atmosphere. Pressure in the ruptured steam generator will remain below that in the primary system until the failed ASDV can be isolated, and recovery actions completed.

Conservative Assumptions

The integrated primary-to-secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere until break flow termination were calculated with the LOFTTR2 program. This information is used in calculating the offsite radiation doses. This section includes a discussion of the methods and assumptions used to analyze the SGTR event and to calculate the mass releases, the sequence of events during the recovery operations, and the calculated results.

Most of the conservative conditions and assumptions used for the margin to overfill analysis are also conservative for the offsite dose analysis, and thus most of the same assumptions are used for both analyses. The major differences in the assumptions that are used for the LOFTTR2 analysis for offsite doses are discussed below.

1. Reactor Trip and Turbine Runback

   An earlier reactor trip is conservative for the offsite dose analysis. Due to the assumed loss-of-offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator ASDVs. Thus an earlier trip time leads to more steam released to the atmosphere from the ruptured and intact steam generators. The time of reactor trip is calculated by modeling the BVPS-2 protection system and this time was used in the analysis. Turbine runback is not modeled since a high power and low secondary mass results in high steam releases.

2. Steam Generator Secondary Mass

   A lower initial mass in the ruptured steam generator results in a conservative prediction of offsite doses. The initial steam generator total fluid mass is assumed to be 10% below the nominal full-power fluid mass.

3. Auxiliary Feedwater (AFW) System Operation

For this analysis, the minimum AFW flow rate of 305 gpm to the ruptured steam generator is assumed to be initiated 60 seconds after reactor trip. A minimum AFW flow rate maximizes steam releases to the atmosphere.

4. Flashing Fraction

When calculating the fraction of break flow that flashes to steam, 100% of the break flow is assumed to come from the hot leg side of the break. Since the tube rupture flow actually consists of flow from the hot leg and cold leg sides of the steam generator, the temperature of the combined flow will be less than the hot leg temperature and the flashing fraction will be correspondingly lower. Thus the assumption is conservative for a SGTR analysis.

Operator Action Times

The major operator actions required for the recovery from a SGTR are discussed in Section 5.4.2.2, and the operator action times used for the margin to overfill analysis are presented in Table 5.4.2-1. The operator action times assumed for the margin to overfill analysis are also used for the offsite dose analysis. However, for the offsite doses analysis, the ASDV on the ruptured steam generator is assumed to fail open at the time the ruptured steam generator is isolated. Before proceeding with the recovery operations, it is assumed that the failed-open ASDV on the ruptured steam generator is isolated by locally closing the associated block valve. BVPS-2 has determined that an operator can locally close the block valve for the ASDV on the ruptured steam generator within 10.0 minutes after the failure. Thus, it is assumed that the ruptured steam generator ASDV is isolated at 10.0 minutes after the valve is assumed to fail open. The operator action time to close the block valve for the ASDV on the ruptured steam generator is the same as that modeled in the Reference 3 analysis. After the ruptured steam generator ASDV is isolated, the additional delay time of 2 minutes (Table 5.4.2-1) is assumed for the operator action time to initiate the RCS cooldown.

Mass Releases

The mass releases are determined for use in evaluating the offsite and control room radiological consequences of the SGTR using the methodology of Reference 2. The steam releases from the ruptured and intact steam generators, the feedwater flow to the ruptured and intact steam generators, and primary-to-secondary break flow into the ruptured steam generator are determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0 to 2 hours are used to calculate the radiation doses at the site boundary for a 2 hour exposure, and the releases for 0 to 8 hours are used to calculate the radiation doses at the low population zone and to the operators in the control room for the duration of the accident.

In the LOFTTR2 analyses, the SGTR recovery actions in the E-3 guideline are simulated until the termination of primary-to-secondary leakage. After the primary-to-secondary leakage is terminated, the operators will continue the SGTR recovery actions to prepare the plant for cooldown to cold shutdown conditions. When these recovery actions are completed, the plant should be cooled and depressurized to cold shutdown conditions. In accordance with the methodology in Reference 2 it is assumed that the

cooldown is performed using the BVPS-2 Emergency Operating Procedures (EOPs) ES-3.3, Post-SGTR Cooldown Using Steam Dump, since this method results in a conservative evaluation of the long-term releases for the offsite dose analysis compared to the other cooldown methods in the EOPs. This procedure for depressurizing the ruptured steam generator is assumed even though the LOFTTR2 analysis performed to calculate releases up until break flow termination has assumed ASDV isolation.

The high level actions for the post-SGTR cooldown method using steam dump in the BVPS-2 EOP ES-3.3 are discussed below.

1. Prepare for Cooldown to Cold Shutdown.

   The initial steps to prepare for cooldown to cold shutdown will be continued if they have not already been completed. A few additional steps are also performed prior to initiating cooldown. These include isolating the cold leg SI accumulators to prevent unnecessary injection, energizing pressurizer heaters as necessary to saturate the pressurizer water and to provide for better pressure control, and assuring shutdown margin in the event of a potential boron dilution due to in-leakage from the ruptured steam generator.

2. Cooldown RCS to Residual Heat Removal (RHR) System Temperature.

   The RCS is cooled by steaming and feeding the intact steam generators similar to a normal cooldown. Since all immediate safety concerns have been resolved, the cooldown rate should be maintained less than the maximum allowable rate of 100°F/hr. The preferred means for cooling the RCS is steam dump to the condenser, since this minimizes the radiological releases and conserves feedwater supply. The ASDVs on the intact steam generators can also be used if steam dump to the condenser is unavailable. When the RHR system operating temperature is reached, the cooldown is stopped until RCS pressure can also be decreased. This prevents the pressure/temperature limits from being exceeded.

3. Depressurize RCS to RHR System Pressure.

   When the cooldown to RHR system temperature is completed, the pressure in the ruptured steam generator is decreased by releasing steam from the ruptured steam generator. Steam release to the condenser is preferred, since this minimizes radiological releases, but steam can be released to the atmosphere using the ASDV on the ruptured steam generator if the condenser is not available. As the ruptured steam generator pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured steam generator in order to prevent in-leakage of secondary side water or additional primary-to-secondary leakage. Although normal pressurizer spray is the preferred means of RCS pressure control, auxiliary spray or pressurizer PORV can be used to control RCS pressure if pressurizer spray is not available.

4. Cooldown to Cold Shutdown.

   When RCS temperature and pressure have been reduced to the RHR system in-service values, RHR system cooling is initiated to complete the cooldown to cold shutdown. When cold shutdown conditions are achieved, the pressurizer can be cooled to terminate the event.

5.4.3.3 Description of Analyses and Evaluations

The LOFTTR2 thermal and hydraulic analysis results for the BVPS-2 offsite dose evaluation are described below, considering operation at the NSSS power of 2910 MWt. The sequence of events for the analysis is presented in Table 5.4.3-1. The transient results for this case are similar to the transient results from the margin to overfill analysis until the ruptured steam generator is isolated. The transient behavior is different after this time, as it is assumed that the ruptured steam generator ASDV fails open after isolation.

Following the tube rupture, the RCS pressure decreases as shown in Figure 5.4.3-1 due to the primary-to-secondary leakage. In response to this depressurization, the reactor trips on Overtemperature Delta-T at about 116 seconds. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. The steam dump system is inoperable due to the assumed loss-of-offsite power, which results in the secondary pressure rising to the steam generator ASDV setpoint as shown in Figure 5.4.3-2. The RCS pressure and pressurizer level also decrease more rapidly following reactor trip as shown in Figures 5.4.3-1 and 5.4.3-3. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at approximately 146 seconds.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator.

   Recovery actions begin by throttling the auxiliary feedwater flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously in Section 5.4.2, isolation of the AFW flow to the ruptured steam generator is assumed to be completed when the narrow range level reaches 27.5% on the ruptured steam generator or at 5.5 minutes after reactor trip of the SGTR, whichever is longer. Complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 27.5% on the ruptured steam generator or at 15 minutes after reactor trip, whichever is longer. For the BVPS-2 analysis, the time to reach 27.5% is approximately 11.25 minutes after reactor trip. AFW isolation is assumed at that time, and the ruptured steam generator is assumed to be isolated at approximately 15 minutes after reactor trip.

   The ruptured steam generator ASDV is also assumed to fail open at this time. The failure causes the steam generator to rapidly depressurize, which results in an increase in primary-to-secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary-to-secondary, which results in RCS pressure and temperature decreasing more rapidly than in the margin to overfill analysis. The ruptured steam generator depressurization causes a cooldown in the intact steam generators loops. It is assumed that the time required for the operator to identify that the ruptured steam generator ASDV is open and to locally close the associated block valve is 10.0 minutes. At 620 seconds the depressurization of the ruptured steam generator is terminated and the ruptured steam generator pressure begins to increase as shown in Figure 5.4.3-2.

2. Cooldown the RCS to Establish Subcooling Margin.

After the block valve for the ruptured steam generator ASDV is closed, there is a 2-minute operator action time imposed prior to initiation of cooldown. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the ASDVs on the intact steam generators. Since offsite power is assumed to be lost at reactor trip for this analysis, the cooldown is performed by dumping steam via the ASDVs on the intact steam generators. The cooldown begins at 1740 seconds and is complete at 2908 seconds.

The reduction in the intact steam generators' pressure required to accomplish the cooldown is shown in Figure 5.4.3-2 and the effect of the cooldown on the RCS temperatures is shown in Figures 5.4.3-4 and 5.4.3-5. The RCS pressure and pressurizer level also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 5.4.3-1 and 5.4.3-3. The break flow flashing fraction is calculated throughout the transient based on the difference between the enthalpy of the break flow and the saturation enthalpy at the ruptured steam generator pressure. Break flow is calculated to stop flashing at approximately 1968.5 seconds as shown in Figure 5.4.3-7.

3. Depressurize to Restore Inventory.

After the RCS cooldown is completed, a 4-minute operator action time is included prior to the RCS depressurization. The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The RCS depressurization is initiated at 3152 seconds and continues until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 4% for pressurizer level uncertainty, or pressurizer level is greater than 75%, or RCS subcooling is less than the 21°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated at 3236 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 4%. The RCS depressurization reduces the break flow as shown in Figure 5.4.3-6 and increases SI flow to refill the pressurizer, as shown in Figure 5.4.3-3.

4. Terminate SI to Stop Primary-to-Secondary Leakage.

The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory so that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary-to-secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 21°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 4% allowance for uncertainty.

After depressurization is completed, an operator action time of 3 minutes is assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions are performed at

3416 seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 5.4.3-1.

The intact steam generator ASDVs are automatically opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the ASDVs are opened, the increased energy transfer from primary-to-secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The ruptured steam generator pressure increases to the ASDV setpoint and steam release is reinitiated. Steam generator pressure is maintained at the steam generator ASDV setpoint rather than the safety valve setpoint for modeling efficiency. This modeling is conservative since it delays break flow termination by requiring the RCS pressure to drop further, maximizes the break flow rate by maintaining a larger primary-to-secondary pressure differential, and results in more steam release from the ruptured steam generator. The primary-to-secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

Calculation of Mass Releases

The operator actions for the SGTR recovery up to the termination of primary-to-secondary leakage are simulated in the LOFTTR2 analyses. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary-to-secondary leakage into the ruptured steam generator are determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, it is assumed that the RCS and intact steam generators conditions are maintained stable for a 20 minute period until the cooldown to cold shutdown is initiated. The ASDVs are assumed to be used to cool down the RCS to the RHR system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hr. The RCS and the intact steam generators temperatures at 2 hours are then determined using the RCS and intact steam generators parameters at the time of leakage termination and the RCS cooldown rate. The steam releases and the feedwater flows for the intact steam generators for the period from leakage termination until 2 hours are determined from a mass and energy balance using the calculated RCS and intact steam generators conditions at the time of leakage termination and at 2 hours. Since the ruptured steam generator is isolated, no change in the ruptured steam generator conditions is assumed to occur until subsequent depressurization.

The RCS cooldown is assumed to continue after 2 hours until the RHR system in-service temperature of 350°F is reached. Depressurization of the ruptured steam generator is assumed to be performed immediately following the completion of the RCS cooldown. The ruptured steam generator is assumed to be depressurized to the RHR in-service pressure of 375 psia via steam release from the ruptured steam generator ASDV, since this maximizes the steam release from ruptured steam generator to the atmosphere, which is conservative for the evaluation of the offsite radiation doses. The RCS pressure is assumed to be reduced concurrently with the ruptured steam generator pressure. It is assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions is completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours are determined for the intact steam generators from a mass and energy balance using the RCS and steam generator conditions at 2 hours and at the RHR system in-service conditions. The steam released from the ruptured steam generator from 2 to 8 hours is

determined based on a mass and energy balance for the ruptured steam generator using the conditions at the time of leakage termination and saturated conditions at the RHR in-service pressure.

After 8 hours, it is assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR in-service conditions are assumed to be reached at 8 hours.

Limiting Case

The limiting case with respect to the thermal and hydraulic input to the radiological consequences analysis considers operation at the highest allowable operating temperature (580.0°F), with the highest allowable main feedwater temperature (455°F), the lowest allowable tube plugging level (0%), and the failure of an ASDV on the ruptured steam generator in the full open position.

5.4.3.4 Acceptance Criteria and Results

The analysis is performed to calculate the mass transfer data for input to the radiological consequences analysis. As such no acceptance criteria are defined. The results of the analysis are used as input to the radiological consequences analysis for the SGTR event.

LOFTTR2 Analysis Results

The primary-to-secondary break flow rate throughout the recovery operations is presented in Figure 5.4.3-6. The calculated break flow flashing fraction and integrated flashed break flow are presented in Figures 5.4.3-7 and 5.4.3-8, respectively. The ruptured steam generator ASDV steam release rate is presented in Figure 5.4.3-9. The total intact steam generator ASDV steam release rate is presented in Figure 5.4.3-10. The ruptured steam water volume is shown in Figure 5.4.3-11. For this case, the water volume in the ruptured steam generator when the break flow is terminated is less than the volume for the margin to overfill case and significantly less than the total steam generator volume of 5730 ft^3. The ruptured steam water mass is shown in Figure 5.4.3-12.

Mass Release Results

The mass release calculations were performed using the methodology discussed above. For the time period from initiation of the accident until leakage termination, the releases are determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip will be through the condenser vacuum exhaust. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator ASDVs.

The mass releases for the SGTR event assuming failure and isolation of the ruptured steam generator ASDV are presented in Table 5.4.3-2, at the NSSS power of 2910 MWt. The results indicate that approximately 74,200 lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours, 43,600 lbm of steam is released to the atmosphere from the ruptured steam generator. A total of 199,500 lbm of primary water is transferred to the secondary side of

the ruptured steam generator before break flow is terminated. A total of 9226.2 lbm of this break flow is assumed to flash to steam upon entering the steam generator.

5.4.3.5 Conclusions

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis is complete. Data for the limiting case is tabulated in Tables 5.4.3-1 and Table 5.4.3-2. The transient is outlined in Figures 5.4.3-1 through 5.4.3-12.

The results and conclusions of the SGTR thermal-hydraulic analysis for offsite radiological consequences performed for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

5.4.3.6 References

1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Lewis, Huang, Behnke, Fittante, Gelman, August 1987.

2. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986.

3. WCAP-12737, "LOFTTR2 Analysis for a Steam Generator Tube Rupture for Beaver Valley Power Station Unit 2," October 1990.

Table 5.4.3-1
BVPS-2 Sequence of Events for Offsite Radiation Dose Analysis

Event	Time (seconds)
SG Tube Rupture	0
Reactor Trip – Overtemperature Delta-T	116
SI Initiated	146
Isolate AFW to Ruptured SG	790
Ruptured SG Isolated	1018
SG ASDV Fails Open	1020
SG ASDV Block Valve Closed	1620
RCS Cooldown Initiated	1740
RCS Cooldown Terminated	2908
RCS Depressurization Initiated	3152
RCS Depressurization Terminated	3236
SI Terminated	3416
Steam Relief to Maintain Subcooling	3516
Break Flow Terminated	3932

Table 5.4.3-2
BVPS-2 Operation at 2910 MWt Mass Releases Total Mass Flow (Pounds)

	Time Period			
	Time Zero to Time of Reactor Trip[1]	Time of Reactor Trip to Time at Which Break Flow is Terminated[2]	Time at Which Break Flow is Terminated to 2 Hours	2 Hours to Time at Which RCS Reaches RHR In-Service Conditions[3]
Ruptured SG				
- condenser	142,300	0.0	0.0	0.0
- atmosphere	0.0	74,200	0.0	43,600
- feedwater	133,200	36,200	0.0	0.0
Intact SGs				
- condenser	281,900	0.0	0.0	0.0
- atmosphere	0.0	172,400	230,000	775,600
- feedwater	281,900	526,600	259,700	785,100
Break Flow	9,200	190,300	0.0	0.0
Flashed Break flow	1730.2	7,496	0.0	0.0

Note:
(1) Reactor trip time = 116 seconds
(2) Time when primary to secondary break flow terminates = 3932 seconds
(3) Time when RCS reaches RHR In-Service Conditions = 8 hours

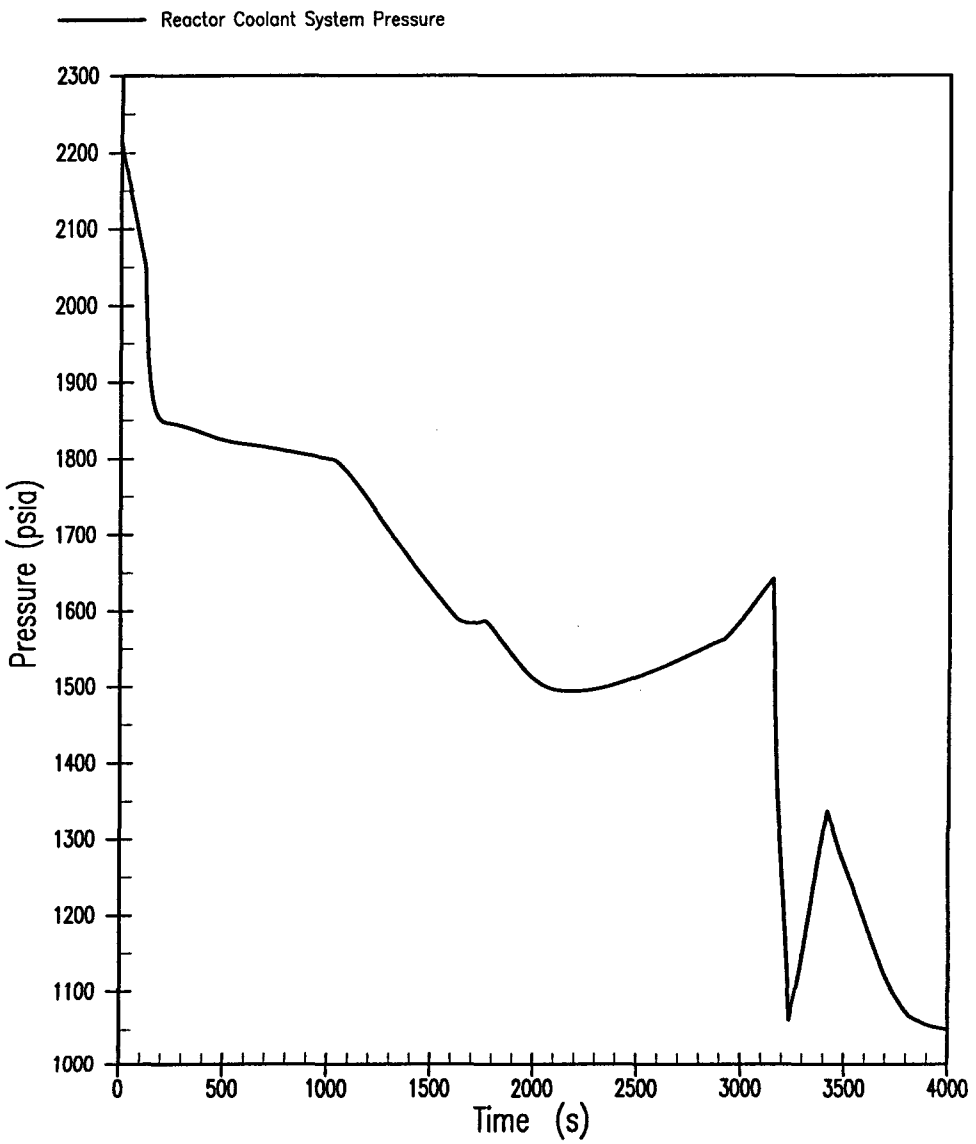

Figure 5.4.3-1
BVPS-2 Offsite Radiation Dose Analysis
Pressurizer Pressure

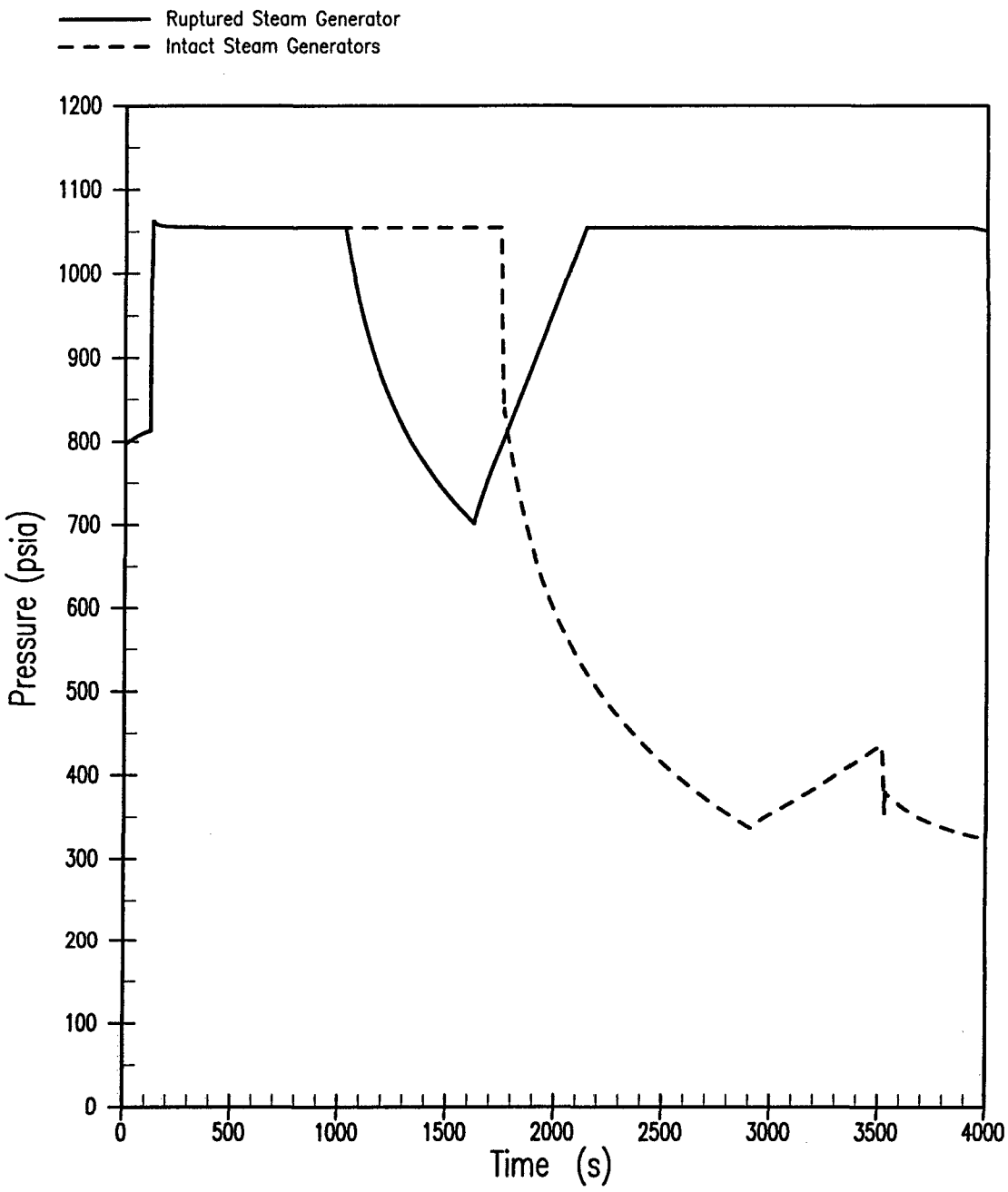

Figure 5.4.3-2
BVPS-2 Offsite Radiation Dose Analysis
Secondary Pressure

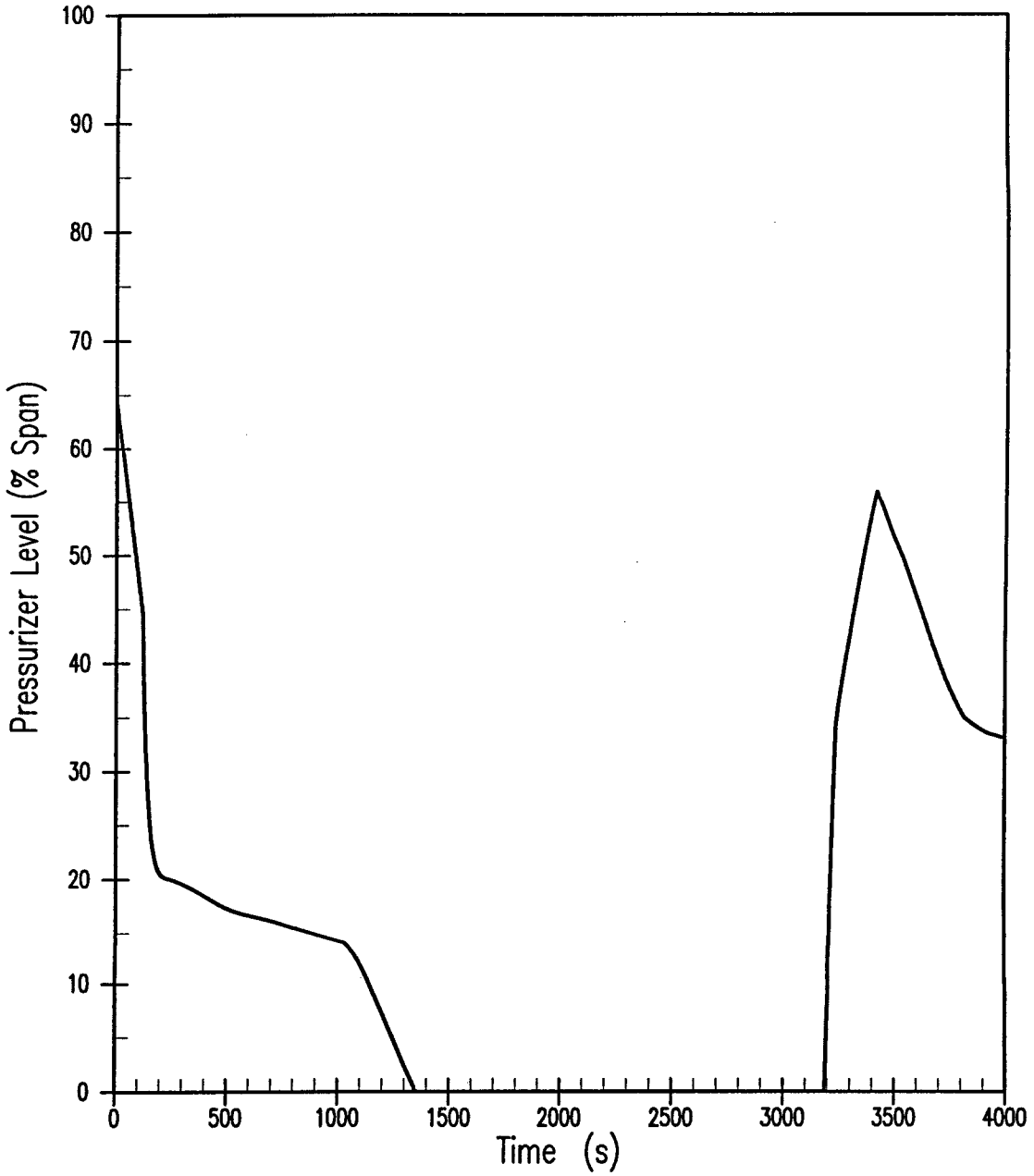

Figure 5.4.3-3
BVPS-2 Offsite Radiation Dose Analysis
Pressurizer Level

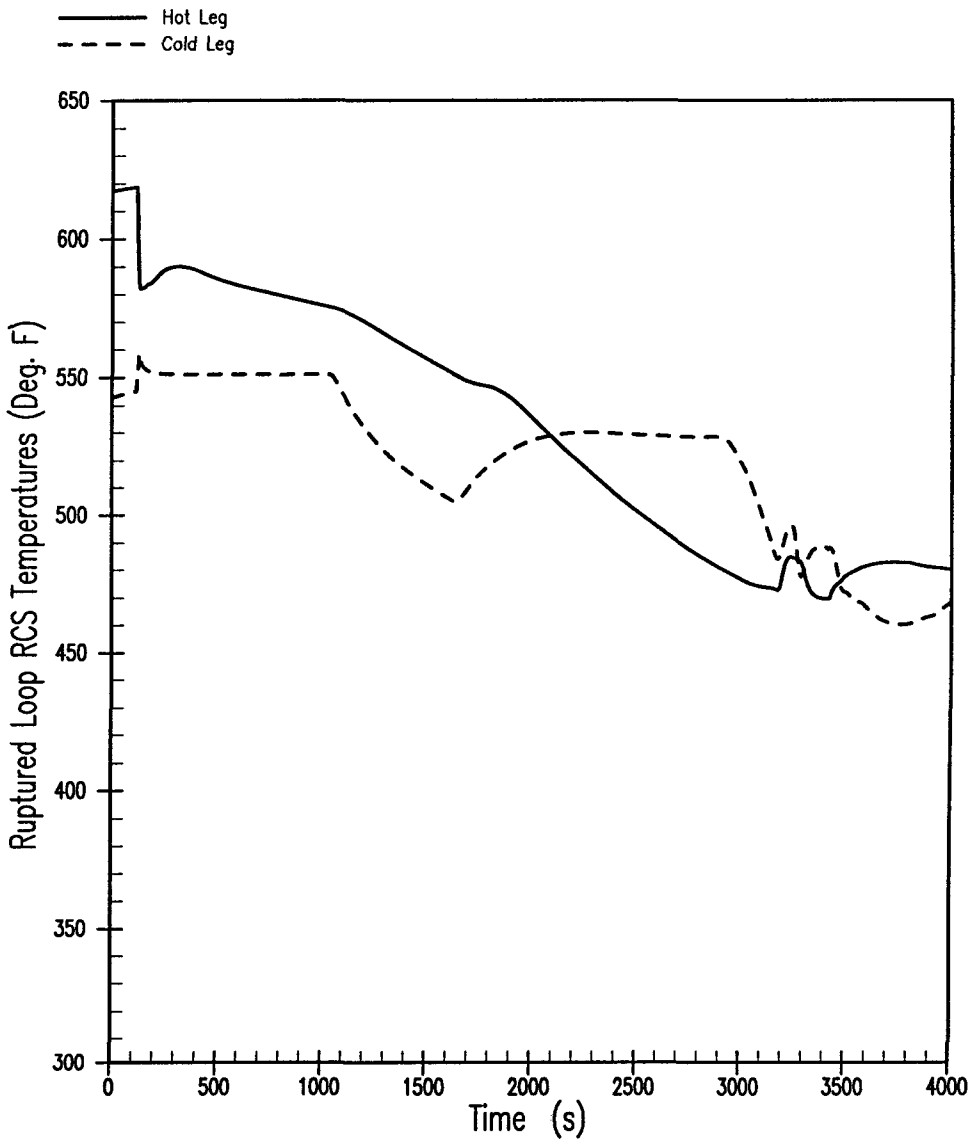

Figure 5.4.3-4
BVPS-2 Offsite Radiation Dose Analysis
Ruptured Loop Hot & Cold Leg Temperatures

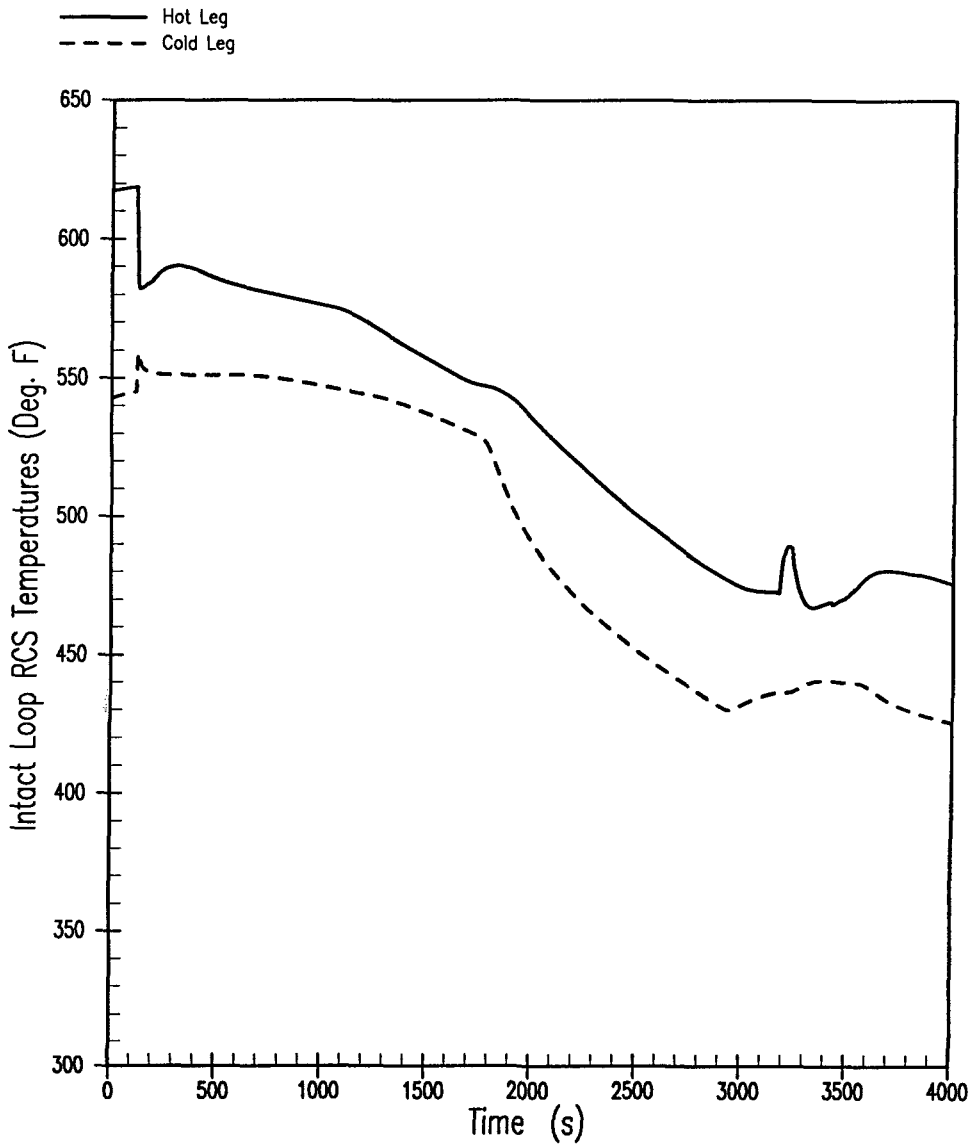

Figure 5.4.3-5
BVPS-2 Offsite Radiation Dose Analysis
Intact Loop Hot & Cold Leg Temperatures

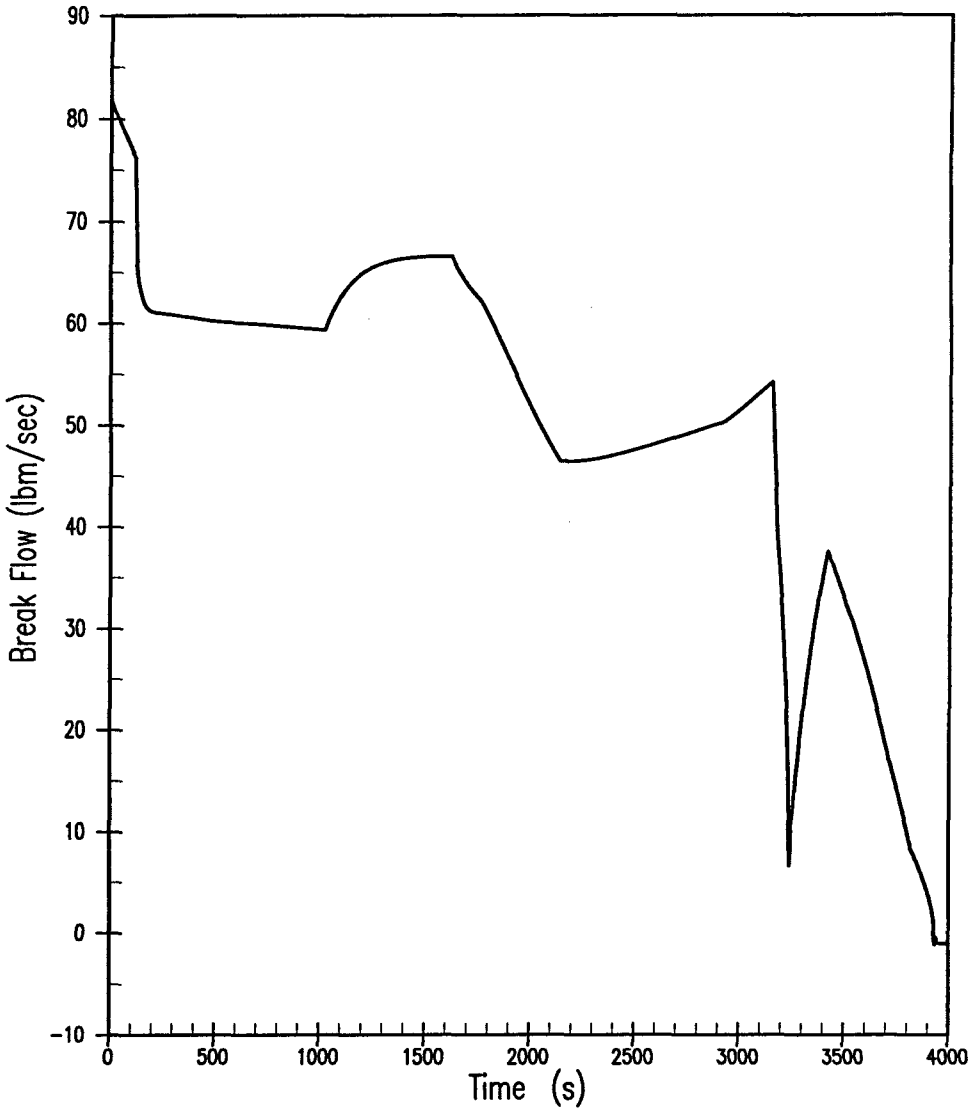

Figure 5.4.3-6
BVPS-2 Offsite Radiation Dose Analysis
Primary-to-Secondary Break Flow

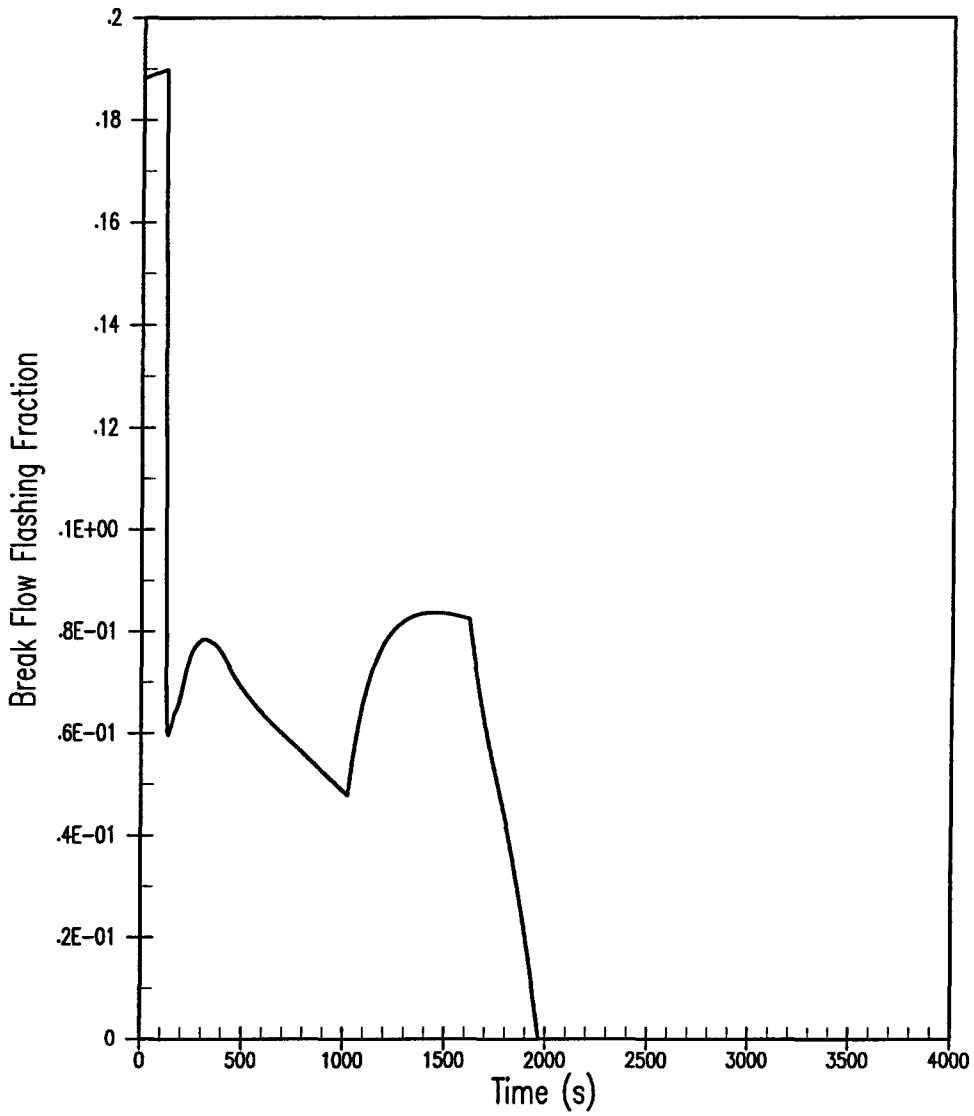

Figure 5.4.3-7
BVPS-2 Offsite Radiation Dose Analysis
Break Flow Flashing Fraction

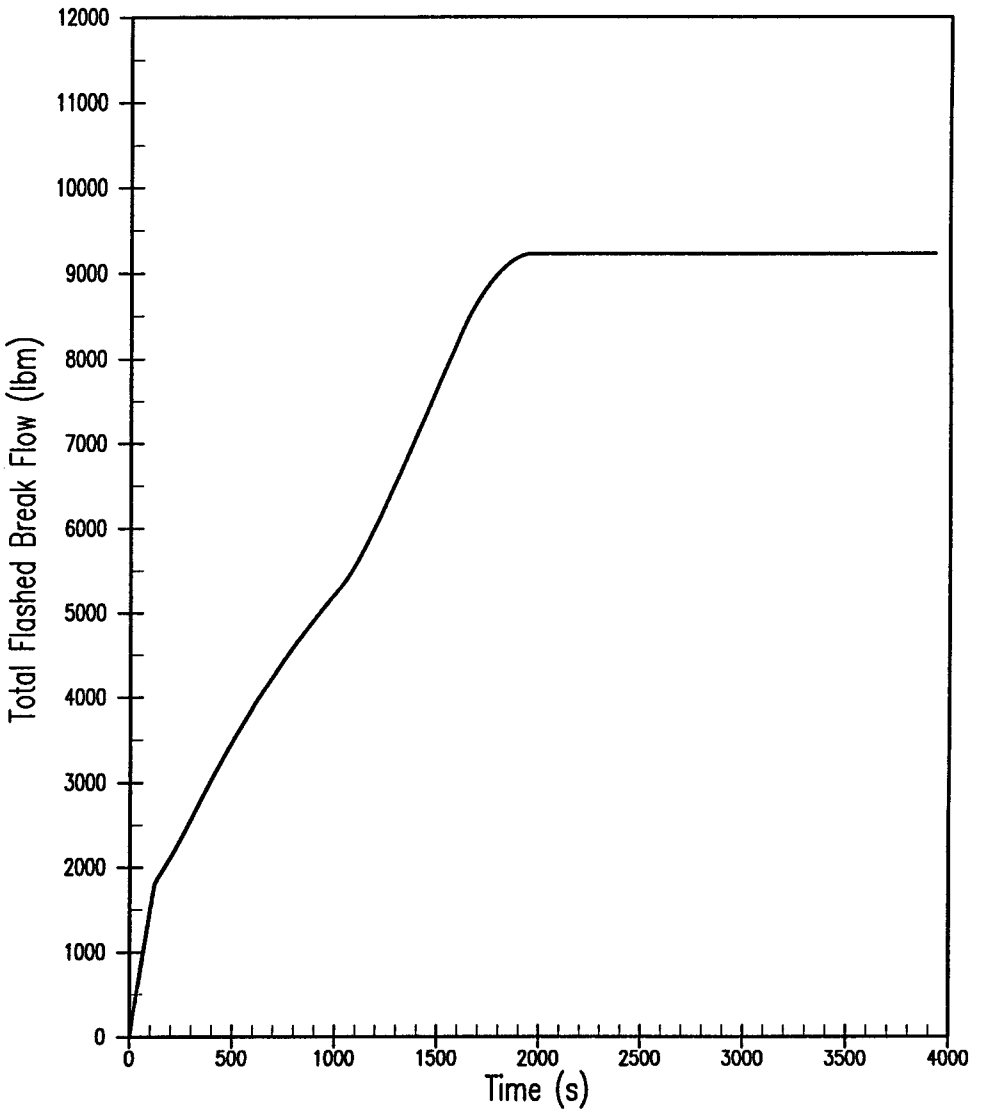

Figure 5.4.3-8
BVPS-2 Offsite Radiation Dose Analysis
Total Flashed Break Flow

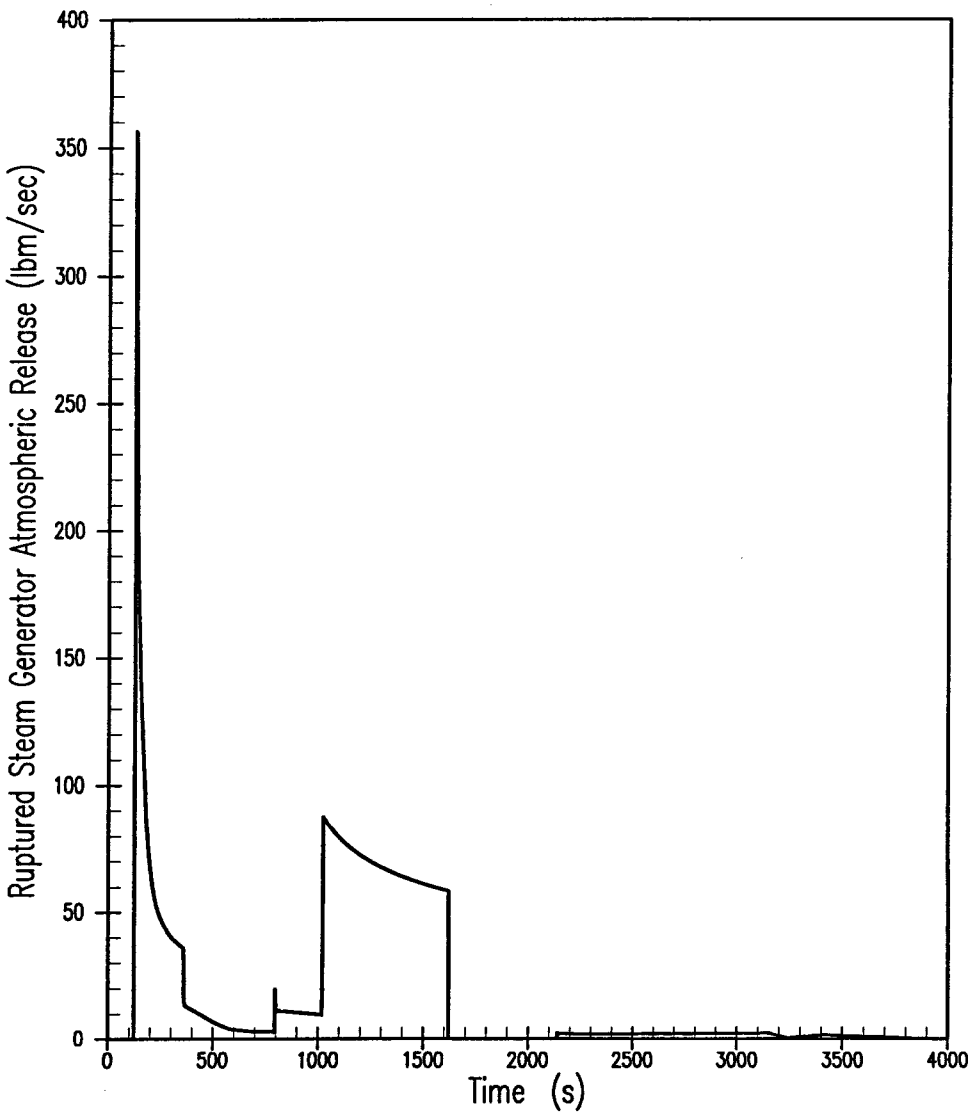

Figure 5.4.3-9
BVPS-2 Offsite Radiation Dose Analysis
Ruptured SG Mass Release Rate to the Atmosphere

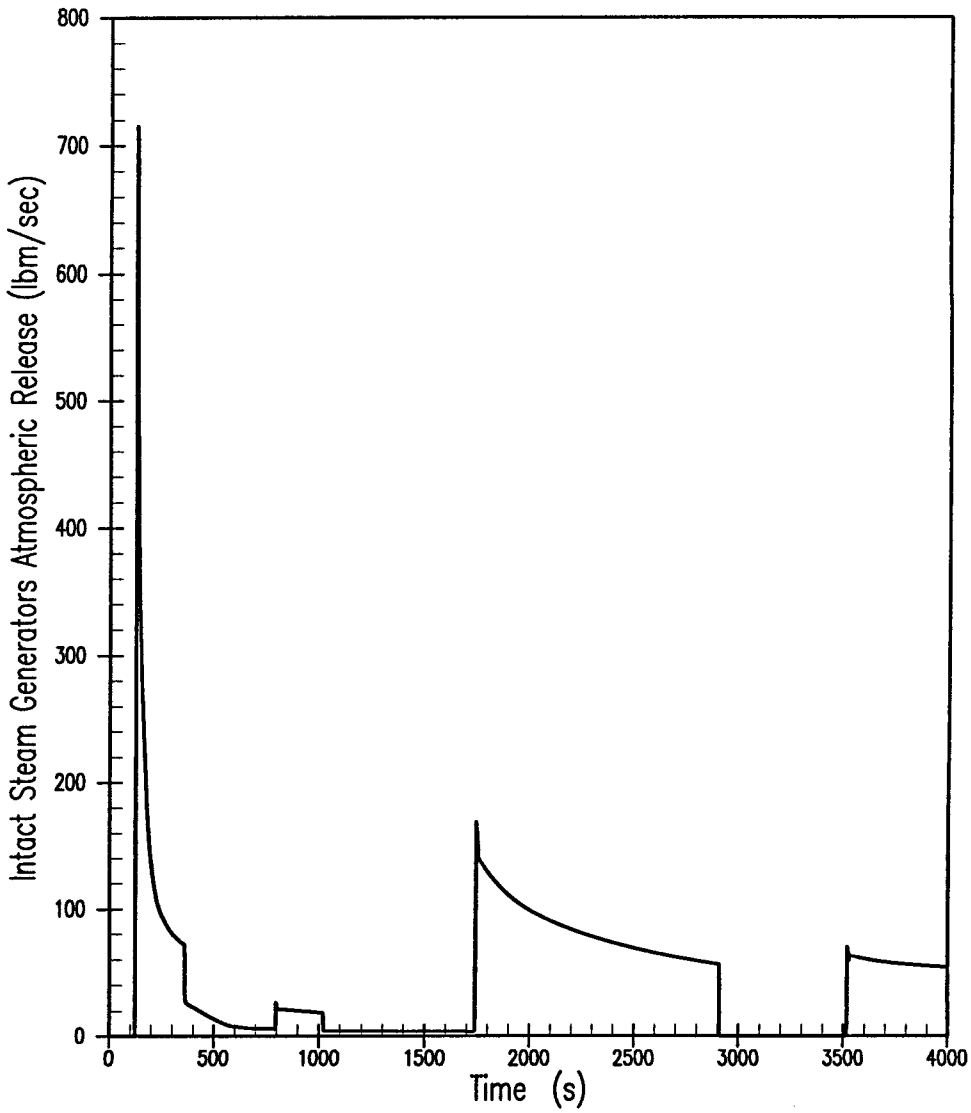

Figure 5.4.3-10
BVPS-2 Offsite Radiation Dose Analysis
Intact SGs Mass Release Rate to the Atmosphere

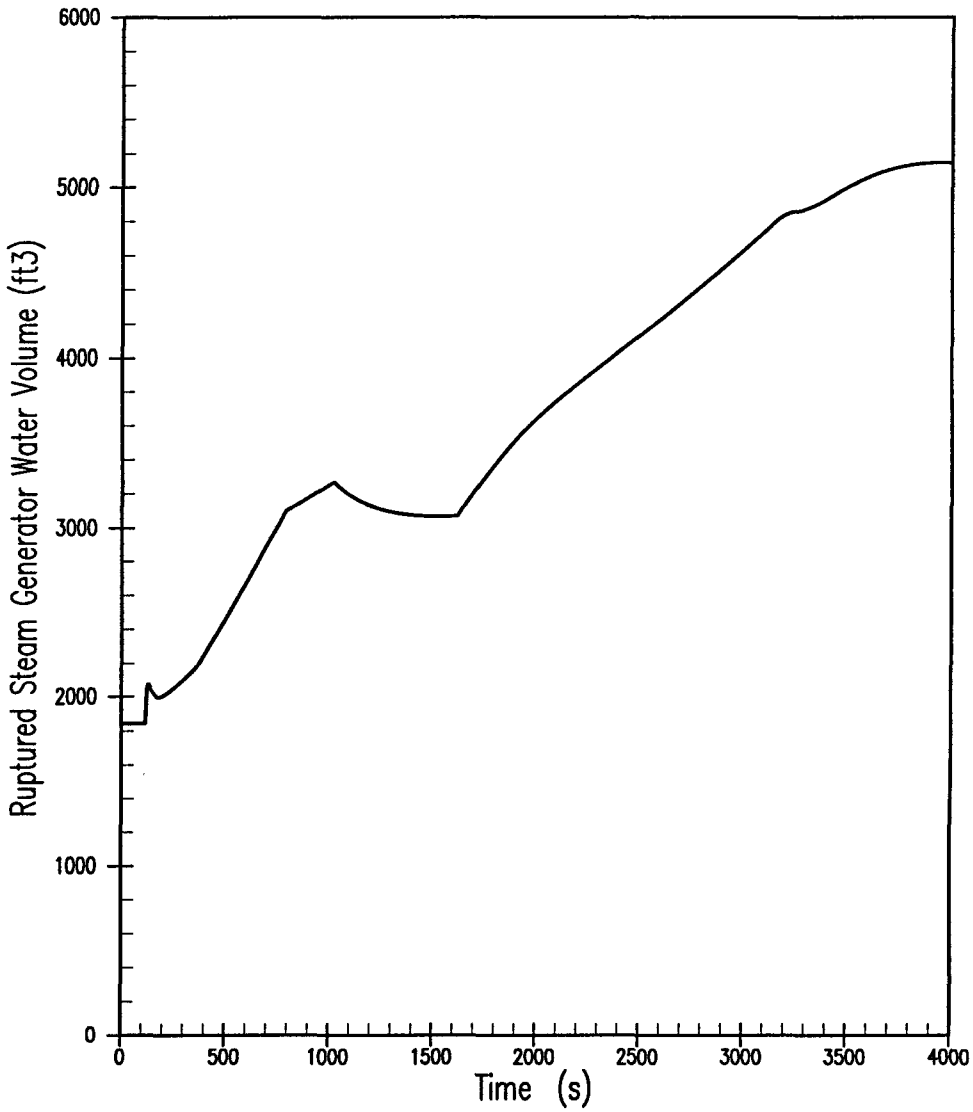

Figure 5.4.3-11
BVPS-2 Offsite Radiation Dose Analysis
Ruptured SG Water Volume

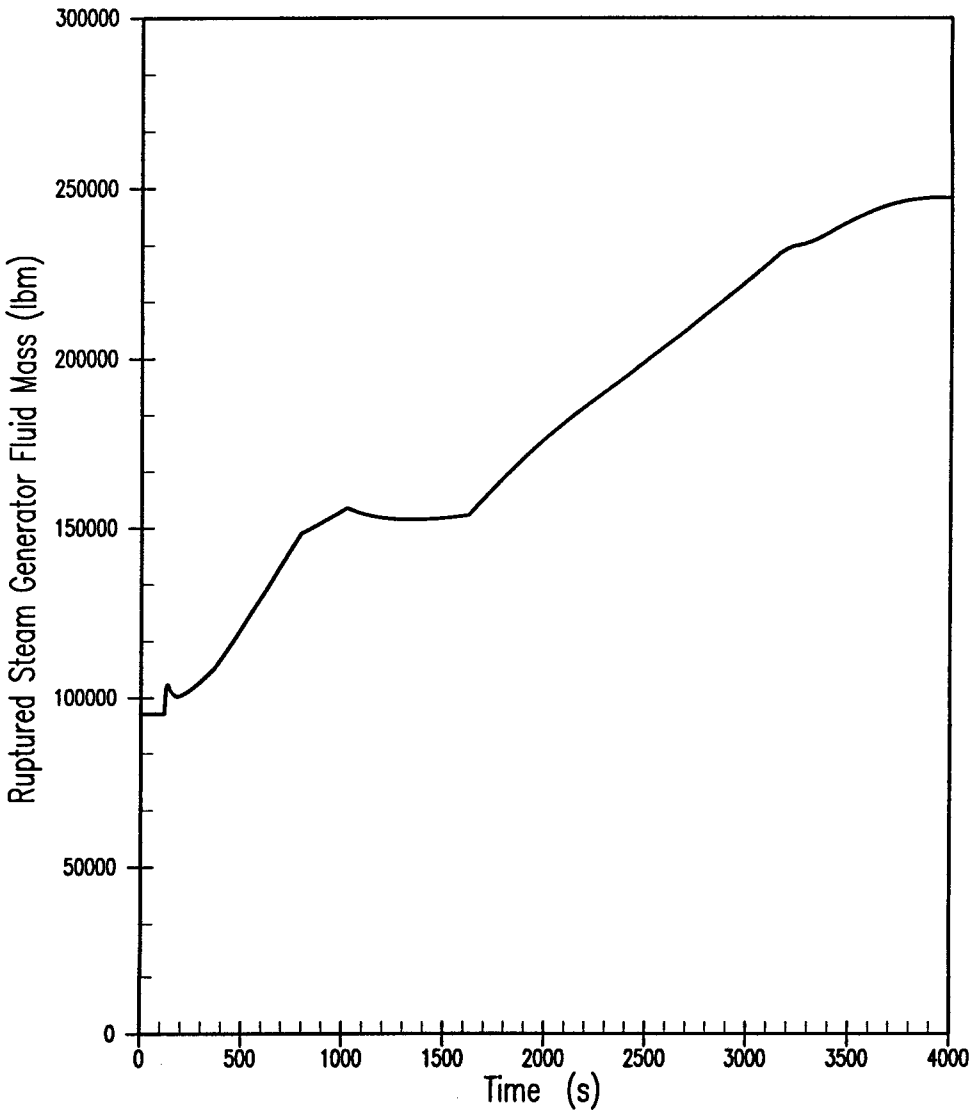

Figure 5.4.3-12
BVPS-2 Offsite Radiation Dose Analysis
Ruptured SG Water Mass

Enclosure 2 of L-05-204

Updated Radiological Assessments for the BVPS-1 & 2 Main Steam Line Break (MSLB) and the BVPS-2 Steam Generator Tube Rupture (SGTR) for the EPU LAR

Section 5.11 "Radiological Assessments" of the Extended Power Uprate (EPU) License Amendment Request (LAR) Licensing Report addresses the radiological dose consequences impact of EPU at BVPS-1 and BVPS-2. The site boundary and control room dose consequence analyses have been updated and a summary of the changes to Section 5.11 of the Licensing Report is provided below:

- The Unit 1 Main Steam Line Break (MSLB) radiological assessment has been replaced with that provided in Section 5.11 of the Replacement Steam Generator (RSG) LAR (Reference 2) Licensing Report to maintain consistency between the EPU and RSG submittals.

- The Unit 2 MSLB radiological assessment has been updated to be consistent with the methodology used for the Unit 1 MSLB in the RSG LAR (Reference 2). Specifically, the change reflects elimination of the steam generator (SG) tube leakage reduction credited previously in the faulted SG at Residual Heat Removal (RHR) initiation (i.e., at T=8 hours). This reduction had taken into consideration the reduced Reactor Coolant System (RCS) pressure at RHR cut-in versus accident initiation (425 psid vs 2235 psid). Credit for this tube leakage reduction is not required to meet the regulatory limits for dose consequences following a MSLB. Removal of this credit simplifies the dose model and makes it consistent with that used for BVPS-1.

- The BVPS-2 Steam Generator Tube Rupture (SGTR) evaluation has been updated to address the following:
  a) a reduced flash duration in the defective SG to make it consistent with the details of the thermal-hydraulic model, and
  b) additional operator action time for isolation of a failed open steam generator atmospheric dump valve (ADV) on the ruptured steam generator.

- The Waste Gas System Rupture (WGSR) radiological assessment has been revised to demonstrate compliance with current licensing basis dose consequence criteria instead of the previously proposed 500 mrem TEDE at the Exclusion Area Boundary / Low Population Zone, and 5 rem TEDE in the control room. The updated assessment utilizes the current BVPS-1 and BVPS-2 licensing basis dose models with input parameters that reflect the EPU, and control room atmospheric dispersion factors that are based on ARCON96 methodology. It is noted that the accident parameters impacted by the EPU are limited to the reactor coolant mass and radioactivity concentrations. This change meets the criteria for implementation via the 10 CFR 50.59 process, and therefore is not being included for NRC review. Accordingly, the revisions provided reflect the elimination of details not necessary for NRC review.

Revised Section 5.11 "Radiological Assessments" of the EPU LAR Licensing Report is attached. Text changes are provided in bold and highlighted.

5.11 RADIOLOGICAL ASSESSMENTS

5.11.1 Introduction

This section addresses the radiological impact of EPU at Beaver Valley Power Station (BVPS-1 and BVPS-2). The current licensing basis core power level is 2689 MWt. The EPU core power level is 2900 MWt. The EPU NSSS power level is 2910 MWt which includes an additional 10 MWt of net heat input from operation of the reactor coolant pumps.

Additionally, as holder of operating licenses issued prior to January 10, 1997, and in accordance with 10CFR50.67 (Reference 1) and Standard Review Plan 15.0.1 (Reference 2), the accident source terms used in the BVPS-1 and BVPS-2 EPU design basis site boundary and control room dose analyses have been revised to reflect the full implementation of Alternative Source Terms (AST) as detailed in Regulatory Guide 1.183. (Reference 3)

The first use of the AST for BVPS was a selective application to revise the Fuel Handling Accident (FHA) in order to justify certain changes in plant operation and configuration during fuel movements. The analysis was reviewed and approved by the NRC in its SER for OL Amendments No. 241 and 121 (Reference 4). In June 2002, the selective application of AST at BVPS was expanded by Reference 5 to include those accidents (i.e., the Loss of Coolant Accident and the Control Rod Ejection Accident) that were impacted by the change in BVPS containment operating conditions from sub-atmospheric to atmospheric pressure (i.e., containment conversion). The expansion of the selective application of AST at BVPS, submitted by Reference 5, was approved by the NRC in its SER for OL Amendment Nos. 257 and 139 (Reference 38). Reference 5 also contained an application for containment conversion that was withdrawn by FENOC letter L-03-135, dated September 5, 2003 (Reference 39). A revised application for containment conversion has been submitted to the NRC for review and approval as License Amendment Request Nos. 317 (Unit 1) and 190 (Unit 2).

The radiological impact of EPU is evaluated for the following:

- Normal Operation Dose Rates and Shielding
- Normal Operation Annual Radwaste Effluent Releases
- Radiological Environmental Doses for Equipment Qualification (EQ)
- Post-LOCA Access to Vital Areas
- Post-Accident Site Boundary and Control Room Doses

In accordance with regulatory guidance, radiological evaluations for accident related issues are assessed at a core power level of 2918 MWt to include an uncertainty of 0.6%. Installation of improved feedwater measurement instrumentation used for calorimetric power calculation allows for instrument error to be reduced from the traditional 2% as recommended in Regulatory Guide 1.49 (Reference 6). The reduction of the uncertainty allowance for calorimetric thermal power measurement to 0.6% was approved by the NRC in its SER for the License Amendments No. 243/122 for BVPS-1/BVPS-2, respectively (Reference 7).

Except as noted, radiological evaluations for normal operation related issues are assessed for EPU at a core power level of 2900 MWt. The impact on the normal operation "design basis" dose rates/shielding

and the normal operation component of equipment qualification doses, is assessed based on a core power level of 2918 MWt. In addition, in accordance with regulatory guidance, the radwaste effluent assessment assumes a core power level of 2918 MWt, but utilizes flow rates and coolant masses at the NSSS power level of 2910 MWt.

With the exception of the site boundary and control room dose assessments, the EPU evaluations discussed in this section (i.e., those associated with normal operation dose rate/shielding adequacy, normal operation radwaste effluents, environmental levels for equipment qualification and vital access) are based on scaling techniques. The scaled increase in radiation levels also includes the impact of the change in fuel cycle length, and the use of current computer codes, methodology and nuclear data in developing the EPU core and reactor coolant inventory, vs. the methodology, computer tools and nuclear data used in the development of the original licensing basis core/reactor coolant inventory. Note that for the most part, the percentage of the estimated increase that can be attributed directly to the EPU is approximately the percentage of the core uprate.

The impact of EPU on the site boundary and control room doses are discussed for the following accidents applicable to BVPS licensing basis:

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

Note that the LOCA and the CREA are addressed in this application by reference only, since these accident analyses, which were performed in support of containment conversion, are based on EPU conditions. The application for containment conversion submitted by Reference 5, was withdrawn by FENOC letter L-03-135, dated September 5, 2003. A revised application for containment conversion is being submitted to the NRC for review and approval as License Amendment Request Nos. 317 (Unit 1) and 190 (Unit 2).

At BVPS, the SLB Outside Containment, LACP, MSLB, SGTR and WGSR are not directly impacted by the implementation of the AST as there is no accident initiated fuel damage associated with these events. However, with this application and the full implementation of AST at BVPS, the dose acceptance criteria of 10CFR50.67 become applicable to all of the accidents listed in Regulatory Guide 1.183 which include the MSLB, and the SGTR. It is noted that the SLB Outside Containment, the LACP and the WGSR are not addressed in Regulatory Guide 1.183. The dose criteria to which they are evaluated are discussed in Section 5.11.2.

The updated site boundary and control room dose analyses reflect EPU conditions, AST (as applicable), and except as noted, bounding parameter values to encompass an event at either unit. In addition the parameter values assigned to the BVPS-1 steam generators reflect the Replacement Steam Generators.

The MSLB, the SGTR, the WGSR and the FHA dose analyses are unit specific. In accordance with current licensing basis, the BVPS-2 EPU MSLB dose analysis reflects the use of Alternative Repair Criteria (ARC) and addresses an accident induced Steam Generator tube leakage. Note that ARC is not utilized in the BVPS-1 EPU MSLB dose analysis because it is not applicable to the Model 54F Replacement Steam Generators. The SGTR dose analyses for BVPS-1 and BVPS-2 reflect environmental releases based on the unit specific licensing basis mass and energy release calculation methodology.

It is noted that the control room dose analyses reflect a control room design consistent with that approved by the NRC in its SER for OL Amendment Nos. 257 and 139. Specifically, the approved design changes include:

- Conservative estimates of control room unfiltered inleakage that envelope the results of recent tracer gas testing performed in the year 2001, and provide margin for surveillance tests.
- Revised Technical Specification acceptance criteria for the BVPS-1 control room HEPA and charcoal filters which will make the BVPS-1 acceptance criteria similar to the more limiting criteria currently listed for the BVPS-2 control room filters.
- Elimination of credit for the automatic initiation feature of the safety related control room area radiation monitors to initiate the control room emergency pressurization system.
- For those events that take credit for the control room emergency ventilation system (CREVS), manual initiation of CREVS pressurization occurs such that the control room is pressurized by T=30 mins.
- Updated control room atmospheric dispersion factors using ARCON96 methodology **for release points associated with LOCA and CREA**.

In addition, the BVPS-1 FHA, and the BVPS-1 and BVPS-2 MSLB and SGTR take credit for a 30 minute control room purge after the accident sequence is complete and the environmental release has been terminated.

The analyses and evaluations for EPU conditions bound and support operation at the current power level, which supports the staged implementation of EPU at BVPS-1 and BVPS-2.

5.11.2 Regulatory Approach

Summarized below are the regulatory acceptance criteria being utilized for the EPU assessments.

5.11.2.1 Normal Operation Assessments

The regulatory commitments currently associated with normal operation assessments are not impacted by this application and remain applicable for the EPU assessment:

- Normal operation on-site dose rates/available shielding will meet the requirements of 10CFR20 (Reference 16) as it relates to allowable operator exposure and access control

- Normal operation off-site releases and doses will meet the requirements of 10CFR20 and 10CFR50, Appendix I (Reference 17). Performance and operation of installed equipment, and reporting of offsite releases and doses will continue to be controlled by the requirements of the Technical Specifications and the Offsite Dose Calculation Manual.

5.11.2.2 Accident Assessments

The regulatory commitments associated with accident assessments are revised as noted by this application and are summarized below:

- Site Boundary and Control Room Doses: As part of the EPU application, BVPS proposes a "full" implementation of the AST as defined in RG 1.183, Section 1.2.1.

  The acceptance criteria for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses are based on 10CFR Part 50 § 50.67 and Section 4.4 Table 6 of Regulatory Guide 1.183 (also noted in Table 1 of SRP 15.0.1):

  - An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, should not receive a radiation dose in excess of the accident specific total effective dose equivalent (TEDE) value noted in Reference 3, Table 6.

  - An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), should not receive a radiation dose in excess of the accident specific TEDE value noted in Reference 3, Table 6.

    Notes: Since the following events are not specifically addressed in RG 1.183:

    a. The acceptance criterion utilized for the WGSR **remains in accordance with current licensing basis and** is 500 mrem **whole body** per BTP ETSB 11-5 (Reference 8

    b. The acceptance criterion utilized for the SLB outside containment and the LACP represent the most limiting dose criterion in Table 6 of RG 1.183.

  The acceptance criteria for the Control Room Dose are based on 10CFR Part 50.67**, or current licensing basis as determined by GDC 19 and SRP 6.4 (References 1, 34 & 33):**

  - In accordance with 10CFR50.67, and except as noted below for the WGSR, adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

  - In accordance with current licensing basis, the acceptance criteria utilized for the WGSR control room dose reflect the guidance provided in GDC 19 and SRP 6.4; i.e., adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

- Equipment Qualification: The EPU EQ assessment takes into consideration the impact of EPU using scaling techniques and TID 14844 source terms (Reference 9). This approach is acceptable based on Section 1.3.5 of Regulatory Guide 1.183, which indicates that no plant modification is required to address the impact of the difference in source term characteristics (i.e., AST vs. TID 14844) on EQ doses.

- Vital Access Doses: The vital access dose assessment for EPU takes into consideration the impact of EPU using scaling techniques and TID 14844 source terms. This approach is acceptable based on the AST bench marking study reported in SECY-98-154 (Reference 10) which concluded that results of analyses based on TID 14844 would be more limiting earlier on in the event, after which time the AST results would be more limiting. The NRC SER for Fort Calhoun Station's implementation of AST (Reference 11), referenced the SECY-98-154 study as the source for the conclusion that results of analyses based on TID 14844 would be more limiting for periods up to one to four months after which time the AST results would be more limiting. Post-LOCA access to vital areas, usually occur, within the first one or two weeks when the original TID 14844 source term is more limiting.

  The EPU assessment takes into consideration the BVPS-1 and the BVPS-2 specific regulatory commitments associated with post-LOCA vital access.

  BVPS-1: As documented in the NRC SER issued to BVPS-1 relative to compliance with NUREG-0737 II.B.2, (References 12 & 13), the BVPS-1 licensing basis does not include estimated doses per operator mission; rather it is a documented evaluation of the worst case post-accident dose rates in plant areas that may need access following a LOCA.

  BVPS-2: As documented in UFSAR Section 12.3.2.10 and the associated NRC SER (Reference 14), the BVPS-2 compliance with NUREG 0737 II.B.2 is based on ensuring that the vital access dose estimates for identified post-accident missions remain within 5 Rem whole body.

  Note that habitability of the BVPS Emergency Response Facility (ERF)/Technical Support Center (TSC) following a LOCA is addressed in this application by reference only, since the LOCA analysis approved by the NRC in its SER for OL Amendment No. 257 and 139 addressed ERF/TSC habitability, and was performed at EPU conditions.

5.11.3 Computer Codes

The QA Category 1 computer codes utilized in EPU analyses specifically developed to support this application, are listed below. The referenced computer codes have been used extensively to support nuclear power plant design and are a part of BVPS current licensing basis.

1. S&W Proprietary Computer Program RADIOISOTOPE, NU-007, V01, L03.

2. Industry Computer Code ARCON96, "Atmospheric Relative Concentrations in Building Wakes" developed by PNL (S&W Program EN-292, V00, L00).

3. S&W Proprietary Computer Code, PERC2, "Passive Evolutionary Regulatory Consequence Code," NU-226, V00, L01.

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

5.11.4 Radiation Source Terms

5.11.4.1 Core Inventory

The equilibrium core inventory utilized to support the EPU assessment is based on a core power level of 2918 MWt, and current licensed values of fuel enrichment and burnup. The methodology used to develop the core inventory, and the associated isotopic listing, is presented in Section 5.3.3.1 and Table 5.3.3-1 of Reference 5.

5.11.4.2 Coolant Inventory

The design basis primary coolant concentrations and the Technical Specification primary and secondary coolant concentrations utilized to support the EPU assessment reflect an equilibrium core inventory based on a core power level of 2918 MWt, and current licensed values of fuel enrichment and burnup. The methodology to develop the design basis and Technical Specification coolant concentrations is discussed Section 5.3.3.2 of Reference 5.

In accordance with the proposed Technical Specification changes accompanying this amendment request, the primary and secondary coolant concentrations for BVPS-1 are made similar to BVPS-2. The noble gas and halogen primary and secondary coolant Technical Specification activity concentrations for BVPS are presented herein in Table 5.11.4-1

5.11.4.3 Primary Coolant Iodine Concentrations based on Pre-Accident/Accident Initiated Iodine Spike

In accordance with the current BVPS-2 Technical Specifications and the proposed BVPS-1 Technical Specifications, the pre-accident iodine spike concentrations in the reactor coolant is 21 μCi/gm DE I-131 (transient Technical Specification limit for full power operation) or 60 times (based on Reference 3) the reactor coolant iodine Technical Specification concentrations.

The accident generated iodine spike activities in the reactor coolant are based on an accident dependent multiplier, times the equilibrium iodine appearance rate. The equilibrium appearance rates are conservatively calculated based on the Technical Specification reactor coolant activities, along with the maximum design letdown rate, maximum Technical Specification allowed leakage, and an ion-exchanger iodine removal efficiency of 100%. Maximizing the reactor coolant cleanup results in maximizing the equilibrium iodine appearance rates.

The pre-accident iodine spike concentrations and the equilibrium iodine appearance rates (utilized to develop accident initiated iodine spike values), for both BVPS-1 and BVPS-2 are presented herein in Table 5.11.4-2.

5.11.4.4 Gap Fractions for Non-LOCA Events

Table 3 in Regulatory Guide 1.183, specifies the fraction of fission product inventory in the fuel rod gap to be used for non-LOCA accidents. The footnote identifies that the applicability of Table 3 is limited to LWR fuel with peak burnups of 62 GWD/MTU "provided that the maximum linear heat generation rate

does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU." The gap fractions utilized for the non-LOCA events at BVPS which could result in fuel failure, are consistent with the requirements for RG 1.183 and are listed below.

Nuclide Group	Regulatory Guide 1.183 Gap Fraction for Non-LOCA Events
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

The core inventory of noble gases, halogens and alkali metals are presented herein in Table 5.11.4-3. These values are consistent with the values presented for these isotopes in Table 5.3.3-1 of Reference 5.

5.11.5 Normal Operation Dose Rates and Shielding

5.11.5.1 Introduction

Cubicle wall thickness is specified for structural and separation requirements, and to provide radiation shielding to protect radiological Equipment Qualification (EQ), and reduce operator exposure during all modes of plant operation, including maintenance and accidents.

Conservative estimates of the radiation sources in plant systems and components form the bases of normal operation plant shielding and radiation zoning. These radiation source terms are primarily derived from conservative estimates of the reactor core and reactor coolant system (RCS) isotopic inventory and are referred to as "design basis" source terms. EPU will impact the isotopic inventory in the core. In addition, since the "design basis" RCS source term is based on 1% fuel defects, EPU will result in an increase in the "design basis" RCS concentration.

The "expected" radiation source terms in the coolant will also be impacted by EPU. "Expected" source terms are less than that allowable by the plant Technical Specifications and are significantly less than the "design basis" source terms.

The impact of EPU on the plant radiation zones/shielding requirements, and on the calculated "design basis" normal operation dose rates, is assessed using scaling techniques. This section also discusses the impact of EPU on the estimated normal operation component of the total integrated dose used for radiological environmental qualification.

5.11.5.2 EPU Assessment

The original shielding design and calculated dose rates for BVPS were based on a core power of 2766 MWt. The original design RCS source terms were based on 1% defective fuel.

The EPU evaluation is based on a core power of 2918 MWt to include an allowance of 0.6% for instrument uncertainty. The above represents an increase of approximately 5.5% of nuclear fission rate and neutron flux level over the original design value.

The assessment presented below takes into consideration that following EPU, the operation and layout/arrangement of plant radioactive systems will remain consistent with the original design.

The assessment takes into account that following EPU, normal operation dose rates/available shielding must continue to meet the requirements of 10CFR20 related to allowable operator exposure and access control. In addition, the impact of EPU on the normal operation radiological environment must be factored into the EQ Program.

Impact of EPU on Normal Operation Shielding/Radiation Zones: Personnel Exposure

EPU will impact the radiation source terms in the reactor core, irradiated fuels/objects, RCS and downstream radioactive systems. These source terms are expected to increase by approximately 7.9% after a core power uprate from 2689 MWt to 2900 MWt. The radiation exposure received by plant personnel is expected to increase by approximately the same percentage.

The above increase in radiation levels will not affect the radiation zoning or shielding requirements in the various areas of the plant because the increase due to EPU will be offset by the:

- conservatism in the pre-EPU "design basis" source terms used to establish the radiation zones,
- plant Technical Specifications that limit the RCS concentrations to levels well below the design basis source terms, and
- conservative analytical techniques used to establish shielding requirements.

Regardless, individual worker exposures will be maintained within acceptable limits by the site Radiation Protection Program which controls access to radiation areas. In addition, procedural controls and ALARA techniques are used to limit doses in areas having increased radiation levels.

Impact of EPU on Normal Operation Equipment Qualification Doses

The impact of EPU on the existing normal operation EQ dose estimates is summarized below:

1. Areas Near Reactor Vessel/Primary Shield Wall: During normal operation, the radiation source in the reactor core is made up of neutron and gamma fluxes, which are approximately proportional to the core power level.

   The radiation sources during shutdown are the decayed gamma source in the core and the activation activities in the reactor internals, pressure vessel/head, neutron shield tank inner wall, and primary system piping walls, which also vary approximately in proportion to the reactor power.

Area dose rates during normal plant operation at 100% power bound that expected during all other modes of operation including shutdown and is therefore the basis of the dose estimates used for equipment qualification and shielding. Since the dose rates near the reactor vessel at 100% power are dominated by the neutron and gamma fluxes, the impact of the use of extended burn fuel is insignificant. With the EPU core power of 2918 MWt (including 0.6% uncertainty), and since the existing analyses were all performed at 2766 MWt, the normal operation EQ dose rate/dose scaling factor in areas near the reactor vessel is estimated to be 2918/2766, or approximately 1.06.

2. In-Containment Areas Adjacent to RCS/Secondary Shielding: During normal operation, the major radiation source in the reactor coolant system components located within containment is the high energy gamma emitter Nitrogen-16 (N-16). N-16 is produced as the oxygen (of the water moderator) is exposed to the neutron flux present in the reactor core. The amount of activation is defined by the flux (or power) density of the core and the amount of time the moderator is resident in the core. After the moderator exits the core (and neutron field), decay of the N-16 will occur. The amount of decay at any given point in the coolant loop is defined by the time subsequent to exiting the core.

   During shutdown, the major radiation sources in the reactor coolant system components located within containment are the deposited corrosion products on the internal surfaces and the primary coolant activity without N-16.

   Area dose rates during normal plant operation at 100% power bound that expected during all other modes of operation including shutdown and is therefore the basis of the dose estimates used for equipment qualification and shielding. With the EPU core power of 2918 MWt, the neutron flux is expected to increase by the percentage of the EPU. The coolant residence time in the core and the transit time are not expected to change significantly due to EPU. In addition, due to its short half-life, the N-16 activity level is not impacted by the use of extended burn fuel. Therefore, since the existing analyses were all performed at 2766 MWt, the EPU EQ dose rate/dose scaling factor for the areas subjected to the N-16 source is estimated be 2918/2766, or approximately 1.06.

3. Areas near Irradiated Fuels and Other Irradiated Objects: These areas include the refueling canal, spent fuel pool, incore instrumentation drive assembly area, and other areas housing neutron irradiated materials. The relevant radiation source is the gamma radiation resulting from decay of the fission and activation products, which is determined by the fission rate, neutron flux level and the irradiation time. Use of extended burn fuel has insignificant impact since the dose rate near spent fuel is dominated by short-lived isotopes in freshly discharged fuel. Consequently, for dose rate assessment purposes, the radiation source increases approximately in proportion to the core power. Since the analyses were all performed at 2766 MWt, the EPU EQ dose rate/dose scaling factor is 1.06.

4. Areas Outside Containment where the Radiation Source Is Derived from the Primary Coolant Activity: In most areas outside the Reactor Containment the radiation sources are either the primary coolant itself or down-stream sources originating from the primary coolant activity. The radiation source terms are based on a conservative assumption that 1% of the reactor fuels are

defective and the fission products are migrating from the defective fuels to the primary coolant and subsequently to the secondary coolant via steam generator leakage. The defective fuel percentage of the earlier operating PWRs (earlier 1970s) was approximately 0.12% (Reference 15). With the improvement of new fuel design and advance of metallurgy, the actual defective fuel percentage of current PWRs is considerably less than the design value of 1%. Moreover, the primary coolant activity is limited by the plant Technical Specifications, which will typically limit the defective fuel percentage by approximately a factor of 3 to 4 below the design value.

The impact of EPU on the normal operation design basis dose rate and shielding is evaluated by comparison of the original design basis source terms to the EPU design basis source terms. Three source terms were considered: a) total primary coolant source, b) degassed primary coolant source, and c) the noble gas source in the primary coolant. Due to the change in isotopic compositions and gamma energy spectrum, the comparison is based on the dose rate shielded by 0, 1, 2, and 3 ft of concrete for representative source geometry. The EPU evaluation reflects the difference of computer codes used in generating the source terms and shielding analyses, difference in nuclear libraries, and takes into consideration the conservative simplified modeling typically employed in shielding design. In addition, the evaluation considers the operation limits imposed by the plant Technical Specification on the primary coolant activity.

The dose rate ratios resulting from the EPU source to the pre-EPU source for the various design basis source term/shielding configurations discussed above ranged from 0.82 to 1.97. However, since, the design basis primary coolant activity is a very conservative source term (1% defective fuel, maximum core inventory, minimum letdown flow, minimum RCS volume in calculating concentration, etc), credit is taken for a more realistic but limiting upper bound primary coolant activity based on the plant Technical Specification.

In accordance with the BVPS current licensing basis, the estimated isotopic Technical Specification 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 Technical Specification LCO 3.4.8, which for BVPS, is the Technical Specification limit on the iodines. 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, and this is already factored into the design basis coolant isotopic mix.

The iodine concentration in the EPU 1% failed fuel design basis RCS corresponds to 3.69 μCi/g I-131 Dose Equivalent. This represents a factor of 3.69 higher than the activity level allowed by a nominal Technical Specification limit of 1 μCi/g I-131 DE in the reactor coolant. Compared to the proposed BVPS-1 and current BVPS-2 RCS Technical Specification limit of 0.35 μCi/g I-131 DE, the EPU design basis source term is a factor of 10.5 higher than that allowed by the BVPS Technical Specifications. Therefore, taking into consideration the limits on RCS concentrations imposed by the plant Technical Specifications, it is concluded that the calculated dose based on the original design basis primary coolant activity remain bounding at the EPU conditions.

5.11.5.3 Conclusions

Normal Operation Radiation Zones and Personnel Exposure

The normal operation radiation levels in most areas of the plant are expected to increase by approximately 7.9%, i.e., the percentage of EPU.

The above increase in expected radiation levels will not affect radiation zoning or shielding requirements in the various areas of the plant because it is offset by the conservatism in the original "design basis" source terms which were used to establish shielding, the plant Technical Specifications which limit the concentrations in primary coolant (and thus all plant process fluids) to well below "design" levels, and conservative analytical techniques typically used during original plant design to establish shielding requirements.

Regardless, individual worker exposures will be maintained within acceptable limits by the site ALARA program which controls access to radiation areas. In addition, procedural controls may be used to compensate for increased radiation levels

Normal Operation Dose for Equipment Qualification

The estimated normal operation EQ dose EPU scaling factors listed below reflect the fact that the existing values are based on a power level of 2766 MWt, and 1% defective fuels (design basis source terms) in the RCS. Note that for the primary coolant and down stream sources, the EPU scaling factor takes into consideration a limiting upper bound primary coolant activity based on the Plant Technical Specifications.

Radiation Source	Scaling Factor
Reactor core	1.06
Irradiated fuels/objects	1.06
N-16 source in RCS loops	1.06
Primary coolant activity and down stream sources	1.0

5.11.6 Normal Operation Annual Radwaste Effluent Releases

5.11.6.1 Introduction

Liquid and gaseous effluents released to the environment during normal plant operations contain small quantities of radioactive materials.

- Liquid Radioactive Waste: Liquids from reactor process systems, or liquids that have become contaminated with these process system liquids, are considered liquid radioactive waste. These wastes are then processed according to their purity level (boron concentration, conductivity, insoluble solids content, organic content, and activity) before being recycled within the plant, discharged to the environment, or reprocessed through the radioactive waste system for further purification until the dose guidelines of 10 CFR 50 Appendix I are met (Reference 17).

- Gaseous Radioactive Waste: Airborne particulates and gases vented from process equipment, and the building ventilation exhaust air is considered gaseous radioactive waste. The major source of gaseous radioactive waste (processing the reactor coolant by the gas stripper and the cover gas system) are decayed using charcoal adsorbers, continuously decayed using separate pressurized decay tanks, filtered and monitored prior to release to ensure that the dose guidelines of 10 CFR 50 Appendix I are not exceeded.

Regulatory guidance relative to methodology to be utilized to establish whether the radwaste effluent releases from a PWR meet the requirements of 10CFR20/10CFR50 Appendix I is provided in NUREG-0017 (Reference 15). It is noted that the NUREG 0017 methodology is independent of the length of the fuel cycle.

The liquid and gaseous radwaste systems' design must be such that the plant is capable of maintaining normal operation offsite releases and doses within the requirements of 40CFR190, 10CFR20, and 10CFR50, Appendix I. (Note that actual performance and operation of installed equipment, and reporting of actual offsite releases and doses continues to be controlled by the requirements of the Offsite Dose Calculation Manual)

The BVPS-1 and BVPS-2 Annual Radioactive Effluent Release Reports for 1997 through 2001 (References 27 through 31), demonstrate that the current gaseous and liquid radwaste releases from the site are well within the release/dose limits set by 40CFR190, 10CFR20, and 10CFR50, Appendix I (References 16, 17, and 32). The impact of the EPU on these releases is evaluated to ensure continued operation within regulatory limits.

5.11.6.2 EPU Assessment

EPU will increase the activity level of radioactive isotopes in the reactor coolant, and, by primary-to-secondary leakage, in the secondary coolant/steam. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radwaste systems where they are processed prior to discharge.

- Liquid Radioactive Waste: As the activity levels in the reactor coolant or secondary fluids increase, the activity level of liquid radwaste inputs /effluents are proportionately increased. Although some wastes such as from the RCS feed and bleed operations may increase due to an EPU, most, if not all, of the water generated by these operations would be recycled within the plant thereby minimizing the impact of additional waste generation on plant effluent analyses.

- Gaseous Radioactive Waste: For the gaseous releases, the analysis is more complex as gaseous effluents are composed of two components:

  a. leakages from the RCS or secondary-plant steam and the normal operation gaseous waste effluents, which are coolant concentration-based and

  b. effluents from the gaseous waste system during shutdown sequences and coolant leakages into containment which are coolant inventory based.

Using the methodology outlined in NUREG-0017 Rev. 0, the change in coolant and steam activity due to EPU is estimated using scaling techniques. To bound the estimated impact of EPU on the annual offsite releases the highest percentage change in activity levels of short lived ($t_{½}$ < 8 days) and long lived isotopes in each chemical grouping found in primary and secondary fluids that characterize each station are applied to applicable pathways for noble gases (short half-life), iodines and particulates (long half-life) and liquids (long half-life). The percentage change is applied to the off-site doses reported in the annual and semi-annual effluent reports for 1997-2001 (adjusted to reflect a 100% capacity factor) to determine whether the estimated off-site doses following EPU, although increased, continue to remain below the regulatory limits and the guidelines of 10CFR20/10CFR50 Appendix I.

Additionally, if the projected increase in offsite doses due to effluents either approaches or exceeds 10CFR50, Appendix I guidelines and/or the direct radiation doses are expected to significantly increase, then the Radiological Effluent Technical Specifications will be examined in order to determine compliance and to determine whether 40CFR190 (Reference 32) is met.

In the EPU assessment summarized below, the baseline pre-EPU operating doses (obtained from BVPS Annual Effluent Reports, References 27 to 31 and adjusted to reflect a 100% capacity factor) reflect a core power level of 2652 MWt (note that for a portion of the year 2001, BVPS operated at a core power level of 2689 MWt, but the assessment conservatively assumes a core power level of 2652 MWt for the entire year of 2001). The plant operating parameters utilized for pre-EPU conditions are consistent with values currently noted in the plant operations manuals and/or the ODCM.

Included in the list of parameters utilized for the assessment are the following pre-EPU parameter values:

Pre-EPU Conditions (1997 to 2001)	
Licensed core thermal power during 1997 to 2001	2652 MWt
Mass of coolant in primary system, (excluding pressurizer and primary coolant purification system), at full power	BVPS-1: 374,000 lbm at 0% SG tube plugging BVPS-2: 374,700 lbm at 0% SG tube plugging (0% tube plugging is used to maximize the EPU scaling factor)
Average primary system letdown rate to primary coolant purification	Evaluated at 60 gpm and 120 gpm for maximum impact
Average flow rate through the primary coolant purification system cation demineralizer	6 gpm
Average shim bleed flow	Derived from minimum shim bleed for boron control between 1600 ppm to 10 ppm, and a cycle length of 518 days
SG carryover factor used for evaluation of iodine and nonvolatiles	1% Iodine 0.1% Nonvolatiles
Total steam flow in secondary system	11.8E6 lbm/hr
Mass of liquid in each steam generator at full power	BVPS-1: 100, 140 lbm BVPS-2: 99,860 lbm

Pre-EPU Conditions (1997 to 2001)	
Average steam generator blowdown rate used in evaluation (total)	BVPS-1: 100 gpm BVPS-2: 97 gpm
Fraction of steam generator feedwater processed through the condensate demineralizers	BVPS-1: No C.D. BVPS-2: 0.70
DF's used in evaluation of the condensate demineralizer system	BVPS-2: DF=1

The remaining parameter values utilized in the baseline evaluation are assumed to be unaffected by the EPU. The values for the above listed parameters at EPU conditions are as follows:

EPU Conditions	
Power Level including power level uncertainty	2918 MWt
Mass of coolant in primary system, (excluding pressurizer and primary coolant purification system), at full power	BVPS-1: 353,100 lbm (RSG) at 22% SG tube plugging BVPS-2: 349, 900 lbm at 22% SG tube plugging (22% tube plugging is used to maximize the EPU scaling factor)
Average primary system letdown rate to primary coolant purification	Same as pre-EPU
Average flow rate through the primary coolant purification system cation demineralizer	Same as pre-EPU
Average shim bleed flow	Derived from minimum shim bleed for boron control between 1879 ppm to 10 ppm, and a cycle length of 518 days
SG Carryover factor used for evaluation of iodine and nonvolatile	Same as pre-EPU
Total steam flow in secondary system MFW Temp = 400°F MFW Temp = 455°F interpolated to 430°F	0 to 22% SG tube plugging 12.08 to 12.06 x 10^6 lb/hr 13.04 to 13.01 x 10^6 lb/hr 0% steam generator tube plugging: 12.60 x 10^6 lb/hr 22% steam generator tube plugging: 12.58 x 10^6 lb/hr
Mass of liquid in each steam generator at full power	BVPS-1 (RSG): 101, 799 lbm BVPS-2 (OSG): 105,076 lbm
Average steam generator blowdown rate used in evaluation total	Same as pre-EPU
Fraction of steam generator feedwater process through the condensate demineralizers	BVPS-1: No C.D. BVPS-2: 0.73
DF's used in evaluation of the condensate demineralizer system	Same as pre-EPU

Summarized below is the estimated impact of EPU on normal operation annual effluent releases and Appendix I doses.

Expected Reactor Coolant Source Terms

Based on a comparison of pre-EPU vs. EPU input parameters, and the methodology outlined in NUREG 0017, the maximum expected increase in the reactor coolant source is approximately 18%.

Gaseous and Liquid Effluent Releases

EPU will increase the liquid effluent release concentrations by approximately 14%, as this activity is based on the long-term RCS and secondary side activity and on waste volumes, the latter being essentially independent of power level within the applicability range of NUREG 0017. Tritium releases in liquid effluents will increase in proportion to their increased production, which is directly related to core power and is allocated between the gaseous and liquid releases in this analysis in the same proportion as pre-EPU releases.

Gaseous releases of Kr-85 will increase by approximately the percentage of power increase. Isotopes with shorter half-lives will have varying EPU increase percentages up to a maximum of 18%. This increase percentage is primarily due to the small EPU RCS mass used in the analysis to accommodate future operations with maximum SG tube plugging. The impact of the EPU on iodine releases is slightly greater than the percentage increase in power level. The other components of the gaseous release (i.e., particulates via the building ventilation systems and water activation gases) are not impacted by EPU using the methodology outlined in NUREG-0017. Tritium releases in the gaseous effluents increase in proportion to their increased production, which is directly related to core power and is allocated in this analysis in the same ratio as pre-EPU releases.

Appendix I Doses

The Appendix I doses following EPU are estimated by scaling up the 1997-2001 average annual doses with the following maximum increase in isotopic concentration / inventory due to EPU in either the reactor coolant or the secondary coolant: a 14% increase for doses due to releases via the liquid effluent pathway, an 18% increase for doses due to releases via the gaseous effluent pathway, and a 14% increase for doses due to releases of iodines and particulates to the atmosphere. The average plant capacity factors during the 5 years utilized for this assessment (68.3% for BVPS-1, 73.6% for BVPS-2) are factored in the estimate.

Type of Dose	Appendix I Design Objectives	5 yrs (1997-2001) Annual Average Dose Normalized to 100% Capacity Factor	EPU Scaled Dose	Percentage of Appendix I Design Objectives for EPU Case
Liquid Effluents				
Dose to total body from all pathways	3 mrem/yr per unit	BVPS-1: 0.102 mrem/yr BVPS-2: 0.095 mrem/yr	BVPS-1: 0.12 mrem/yr BVPS-2: 0.11 mrem/yr	BVPS-1: 4% BVPS-2: 3.7%
Dose to any organ from all pathways	10 mrem/yr per unit	BVPS-1: <0.138 mrem/yr BVPS-2: <0.128 mrem/yr	BVPS-1: 0.16 mrem/yr BVPS-2: 0.15 mrem/yr	BVPS-1: 1.6% BVPS-2: 1.5%
Gaseous Effluents				
Gamma Dose in Air	10 mrad/yr per unit	BVPS-1: 0.141 mrem/yr BVPS-2: 0.116 mrem/yr	BVPS-1: 0.17 mrem/yr BVPS-2: 0.14 mrem/yr	BVPS-1: 1.7% BVPS-2: 1.4%
Beta Dose in Air	20 mrad/yr per unit	BVPS-1: 0.149 mrem/yr BVPS-2: 0.111 mrem/yr	BVPS-1: 0.18 mrem/yr BVPS-2: 0.13 mrem/yr	BVPS-1: 0.9% BVPS-2: 0.7%
Dose to total body of an individual	5 mrem/yr per unit	BVPS-1: 1.048 mrem/yr BVPS-2: 0.510 mrem/yr	BVPS-1: 1.24 mrem/yr BVPS-2: 0.60 mrem/yr	BVPS-1: 25% BVPS-2: 12%
Dose to skin of an individual	15 mrem/yr per unit	Not Reported in Annual Radioactive Release Report	18% increase	It is expected that, like the others, the dose to the skin is a small fraction of the Appendix I Limits
Radioiodines and Particulates Released to the Atmosphere				
Dose to any organ from all pathways	15 mrem/yr per unit	BVPS-1: 1.063 mrem/yr BVPS-2: 0.512 mrem/yr	BVPS-1: 1.21mrem/yr BVPS-2: 0.58 mrem/yr	BVPS-1: 8.1% BVPS-2: 3.9%

Solid Radwaste

Though solid radwaste is not addressed in 10 CFR 50, Appendix I, for completeness relative to radwaste assessments, the impact of EPU on solid radwaste generation is summarized below.

For a "new" facility, the estimated volume and activity of solid waste is linearly related to the core power level. However, for an existing facility that is undergoing EPU, the volume of solid waste would not be expected to increase proportionally, since the EPU neither appreciably impacts installed equipment performance, nor does it require drastic changes in system operation or maintenance. Only minor, if any, changes in waste generation volume are expected. This would include the small increase in volume of condensate polishing resins in BVPS-2. However, it is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity.

Thus while the total long lived activity contained in the waste is expected to be bounded by the percentage of the EPU, the increase in the overall volume of waste generation resulting from EPU is expected to be minor.

5.11.6.3 Conclusions

As discussed earlier in Section 5.11.6.1, the commitment is to both 10 CFR 20 and 10 CFR 50 Appendix I, however, 10 CFR 50 Appendix I is more limiting. 10 CFR 20 does have a release rate criteria that does not exist in 10 CFR 50 Appendix I, but as noted earlier, the plant Technical Specifications and the ODCM control actual performance and operation of installed equipment and releases thus maintaining compliance with that aspect of 10 CFR 20. The projected increase in offsite doses due to effluents following EPU is well below the 10CFR50, Appendix I guidelines, and the direct radiation doses are not expected to increase significantly. In accordance with the evaluation requirements of the ODCM, no further effort is needed to confirm continued compliance with the 40CFR190 annual dose limits (25 mrem for the whole body, 75 mrem to the thyroid and 25 mrem to any other organ for the site) following EPU.

The EPU has no significant impact on the expected annual radwaste effluent releases/doses (i.e., all doses remain a small percentage of allowable Appendix I doses). It is therefore concluded that following EPU, the liquid and gaseous radwaste effluent treatment system will remain capable of maintaining normal operation offsite doses within the requirements of 10 CFR 50 Appendix I and 40CFR190.

5.11.7 Radiological Environmental Doses for Equipment Qualification

5.11.7.1 Introduction

In accordance with 10CFR50.49 safety-related electrical equipment must be qualified to survive the radiation environment at their specific location during normal operation and during an accident (Reference 18).

For the purposes of equipment qualification (EQ), BVPS is divided into various environmental zones. The radiological environmental conditions noted for these zones are the maximum conditions expected to occur and are representative of the whole zone. Normal operation values represent 40 years of operation. Except as noted, the post-accident radiation levels at BVPS-1 and 2 are based on the LOCA and TID 14844 source terms with an accident duration of six months. The accident contribution to the environmental doses in most of the areas in the BVPS-2 Fuel Building is based on the FHA. Also, based on existing analyses, the bounding radiation levels in the top floor of the BVPS-2 Service Building occur following a MSLB. However, as part of the EPU assessment performed in support of the EQ Program, it has been determined that there are no safety related equipment located on this elevation of the service building that are required to be functional following a MSLB. Consequently, the environmental zone in this area is based on the LOCA.

The impact of EPU on the normal operation component of the total integrated dose used for radiological equipment qualification is discussed in Section 5.11.5 and its results are summarized in the conclusions of this section.

The methodology used to estimate the impact of EPU on the post-accident component of the total integrated dose is discussed in this section. It is noted that post-accident environmental doses based on the LOCA and the FHA are developed based on an equilibrium core inventory assuming full power operation plus margin, source term guidance available from regulatory documents relative to post-

accident core/damaged fuel assembly releases, and plant specific mitigation system design features/layout. EPU impacts the equilibrium core inventory and therefore the post-accident radiological source terms. Additional factors that can impact the equilibrium core inventory are fuel enrichment and burnup.

At BVPS, the current normal operation as well as accident dose estimates for equipment qualification are based on plant operation at 2766 MWt. As noted in Section 5.11.5, the current normal operation EQ dose estimates also utilize "design basis" primary coolant source terms and reflect 1% fuel defects.

The impact of EPU, with an 18-month fuel cycle, on the normal operation as well as accident component of the total integrated environmental dose estimates at BVPS, is assessed using scaling techniques. Per Section 5.11.5, the impact of EPU on the normal operation component of the total integrated environmental dose in the various plant areas range from 1.0 to 1.06. Theoretically, with all things being equal, the accident component of the total integrated dose estimate should increase by approximately 6% (i.e., 2918/2766). However, because the EPU core reflects: (a) extended burnup; and (b) the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses, the calculated EPU scaling factor values will deviate from the core power ratio.

5.11.7.2 EPU Assessment

Following EPU, the operation and layout/arrangement of plant radioactive systems will remain consistent with original design. The impact of EPU on the normal operation contribution to the EQ dose is discussed in Section 5.11.5. The assessment presented below, focuses on the impact of EPU on the post-accident contribution to the total EQ dose.

Radiological Environmental Zones Based on the LOCA

The impact of EPU on the post-LOCA EQ gamma radiation doses is evaluated based on a comparison of the gamma source terms developed based on the original core inventory used to develop the EQ doses, to the EPU gamma source terms. The approach utilizes scaling techniques based on a source term comparison, rather than developing actual integrated dose estimates at the various zones/component-specific locations, using the new core inventory.

Post-LOCA Gamma Dose Scaling Factor: Review of the post-LOCA EQ dose calculations supporting both BVPS-1 and BVPS-2, indicate, that four sets of core inventories (all based on a core power level of 2766 MWt), developed during different phases of plant design, form the basis of the EQ dose estimates. Comparison of these core inventories, indicate, that for the most part, the isotopic inventory is the same. To maximize the EPU scaling factor, a composite pre-EPU core inventory was developed that utilized the lowest value of activity, per nuclide, from all four of the core inventory sets. This approach allows the development of bounding scaling factors that are applicable to both units.

Computer code PERC2 is used to develop the post-LOCA gamma energy integrated releases (Mev-hr/sec) per energy group vs. time for the EPU as well as the composite pre-EPU core inventory for the following post-accident radiation sources:

- Pressurized recirculating fluid (i.e., the radiation source contains 100% core noble gases, 50% of the core halogens and 1% of the remainder fission products/actinides; this type of source could be present in the post-accident sampling system).

- Containment atmosphere in containment (i.e., the radiation source contains 100% core noble gases, 50% of the core halogens)

- Containment atmosphere in lines outside containment (i.e., the radiation source contains 100% core noble gases, 25% of the core halogens)

- Sump water (i.e., the radiation source contains 50% of the core halogens and 1% of the remainder fission products/actinides; this is the source expected in recirculating fluids following a large break LOCA)

- Plateout surfaces resulting in accumulated halogens (i.e., the radiation source are the core halogens)

For the "unshielded" case, the factor impact on post-accident gamma doses is estimated by ratioing the gamma energy integrated releases weighted by the flux to dose rate conversion factor, as a function of time, for the EPU power level, to the corresponding weighted source terms based on the pre-EPU power level. Note that the unshielded case for sump water includes a thin shield of several inches of water to eliminate the weak photons that will not escape the self-absorption of sump water.

To evaluate the factor impact of the EPU on post-LOCA gamma doses (vs. time) in areas that are "shielded", the pre-EPU and EPU source terms discussed above are weighted by the concrete shielding factors for each energy group. The concrete shielding factors, for 2 feet of concrete (representative of moderate shielding), and 4 feet of concrete (representative of heavy shielding), provide a basis for comparison of the post-LOCA spectrum hardness of source terms with respect to time for both original design and EPU cases.

Since the EPU gamma dose scaling factors will vary with source, time as well as shielding, to cover all types of analysis models/assessments, the maximum dose scaling factor developed from all of the above assessments is used, for the most part, for all source/receptor combinations, with or without shields, and at all time periods after LOCA. Exceptions include a few cases where analysis refinement was needed to support component qualification at EPU conditions.

The gamma dose scaling factor, taking into consideration EPU, extended burn fuel, and the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses, is determined to be 1.19.

Post-LOCA Beta Dose Scaling Factor: Computer code PERC2 is utilized to develop a beta dose vs. time per source type used in the BVPS beta dose EQ assessments (i.e., pressurized recirculating fluid, containment atmosphere in containment, sump water and plateout surfaces), for both the EPU as well as pre-EPU condition. The EPU beta dose scaling factors are developed by ratioing the EPU vs. pre-EPU beta doses per source type analyzed.

The beta dose scaling factor, taking into consideration EPU, extended burn fuel, and the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses, is determined to be 1.17.

Radiological Environmental Zones Based on the FHA

The impact of EPU on the post-accident EQ gamma radiation doses based on the FHA is evaluated based on a comparison of the original source terms used to develop the EQ doses in the Fuel Building to that applicable following EPU. The approach utilizes scaling techniques based on a source term comparison.

The original analysis addressed both a water source and an airborne source. The source terms for releases from the damaged fuel assembly including the airborne source in the fuel building were based on Safety Guide 25 (Reference 19) and reflect a 100-hr decay after shutdown and a radial peaking factor of 1.65. The water source was intended to reflect the impact of iodines retained in the fuel pool.

Post-FHA Gamma Dose Scaling Factor: Computer code PERC2 is used to develop the post-FHA airborne and pool water integrated gamma doses vs. time for the EPU as well as the composite pre-EPU core inventory for a source term based on a gap release mix from Safety Guide 25 (i.e., 10% noble gases, 30% Krypton, and 10% halogens for the airborne source, and 10% halogens for the pool water source). A decontamination factor, (DF) of 100 is applied to the airborne halogen source.

The impact of EPU on the post-FHA airborne gamma dose contribution is estimated by ratioing the integrated gamma dose, for the gap release source from a worst case assembly (i.e., with a radial peaking factor of 1.75) that has decayed for 100 hours, as a function of time, at the EPU power level, to the corresponding integrated gamma dose based on the pre-EPU power level and associated peaking factor. The resulting scaling factor of 1.14 is applicable to the airborne gamma dose in the Fuel Building. Shielded cases are not considered, as the original analyses did not take credit for shielding.

Similarly, the factor impact on the post-FHA pool water gamma dose contribution is calculated using the PERC2 calculated integrated gamma energy release vs. time for a source that comprises only of core halogens. The scaling factor is determined to be 1.26.

Post-FHA Beta Dose Scaling Factor: Computer code PERC2 is utilized to develop a beta dose vs. time following a FHA in the Fuel Building, for an airborne source term that reflects a gap mix resulting from a FHA, for both the EPU as well as pre-EPU condition. The EPU beta dose scaling factor of 1.14 is developed by ratioing the EPU vs. pre-EPU beta doses. Due to the short range of beta radiation and the shielding provided by the pool water, there is no beta dose contribution from the fuel pool to equipment located in the Fuel Building.

5.11.7.3 Conclusions

The impact of EPU on the estimated normal operation and post-accident total integrated environmental doses in the various areas of the plant is summarized below. Note that the estimated scaling factors reflect EPU, extended burn fuel and the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses.

Normal Operation

The estimated EPU normal operation environmental dose scaling factors are discussed in Section 5.11.5 and are summarized below: Note that for primary coolant and down stream sources, the EPU scaling factor takes into consideration a limiting upper bound primary coolant activity based on the Plant Technical Specifications.

Radiation Source	Scaling Factor
Reactor core	1.06
Irradiated fuels/objects	1.06
N-16 source in RCS loops	1.06
Primary coolant activity and down stream sources	1.0

Accident Dose

The estimated EPU accident environmental dose scaling factors are summarized below

- Doses based on the LOCA: The bounding EPU gamma and beta dose scaling factors are 1.19 and 1.17, respectively
- Doses based on the FHA: The bounding EPU gamma and beta dose scaling factor is 1.26 and 1.14, respectively.

The scaling factors discussed above are applied to the existing environmental conditions, to establish the new radiation environmental levels in the EQ Zones or at component specific locations. Note that in some cases, analysis refinement will be needed to support component specific equipment qualification at EPU conditions. The impact of the EPU radiological environmental conditions on equipment qualification is discussed in Section 10.10.

5.11.8 Post-LOCA Access to Vital Areas

5.11.8.1 Introduction

In accordance with NUREG 0737, II.B.2, vital areas are those areas within the station that will or may require access/occupancy to support accident mitigation or recovery following a Loss of Coolant Accident (LOCA) (Reference 13).

The post-LOCA environmental gamma dose rates are developed based on an equilibrium core inventory assuming full power operation plus uncertainty, source term guidance available from regulatory documents relative to post-accident core releases, and plant specific mitigation system design features/layout. EPU impacts the equilibrium core inventory and therefore the post-LOCA radiological source terms.

At BVPS, the current post-LOCA gamma dose rate estimates used for vital access assessments are based on plant operation at 2766 MWt. The impact of EPU, and with an 18-month fuel cycle, on the estimated

post-LOCA environmental gamma dose rates at BVPS, is assessed, using scaling techniques. Theoretically, with all things being equal, the post-LOCA environmental gamma dose rates, and the operator dose per identified mission, should increase by approximately 6% (i.e., 2918/2766). However, because the EPU core reflects: (a) extended burnup; and (b) the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses, the calculated EPU scaling factor values will deviate from the core power ratio.

The EPU assessment takes into consideration the BVPS-1 and the BVPS-2 specific regulatory commitments associated with vital access.

> BVPS-1: As documented in the NRC SER issued to BVPS-1 relative to compliance with NUREG-0737 II.B.2, (References 12 & 13), the BVPS-1 licensing basis does not include estimated doses per operator mission; rather it is a documented evaluation of the worst case post-accident dose rates in plant areas that may need access following a LOCA.
>
> BVPS-2: As documented in UFSAR Section 12.3.2.10 and the associated NRC SER (Reference 14), the BVPS-2 compliance with NUREG 0737 II.B.2 is based on ensuring that the vital access dose estimates for identified post-accident missions remain within 5 Rem whole body. Consistent with Licensing Amendment Request No. 189 (Unit 2) requesting removal of the Hydrogen Recombiners, Access Route Nos. 7-I and 7-II (operation of the Hydrogen Control System) will not be addressed as the BVPS licensing basis no longer includes the hydrogen control system location as a vital area that needs access following an accident. (Reference 40)

Note that habitability of the BVPS Emergency Response Facility (ERF)/Technical Support Center (TSC) following a LOCA is addressed in Section 5.11.8.3 by reference only, since, the LOCA analysis approved by the NRC in its SER for OL Amendment No. 257 and 139 addressed ERF/TSC habitability, and was performed at EPU conditions.

5.11.8.2 EPU Assessment

The original calculations supporting vital access assessments at BVPS were based on a core power of 2766 MWt, and a one-year fuel cycle.

The EPU evaluation is based on a core power of 2918 MWt to include a value of 0.6% for instrument uncertainty, with an 18-month fuel cycle which will increase the fuel activity inventory of isotopes with long half-lives. The equilibrium core inventory associated with the EPU core, applicable to both units, is discussed in Section 5.11.4.1.

The EPU assessment assumes that following EPU, the operation and layout/ arrangement of plant radioactive systems will remain consistent with the original design.

Review of the post-LOCA vital access gamma dose rate/dose calculations supporting BVPS-1 and BVPS-2, determined that, though the core inventory utilized for BVPS-2 vital access assessments is available in the supporting analyses, the core inventory/source terms utilized by QUADREX Corp. to develop the vital access dose rates at BVPS-1 was not retrievable. However, it is reasonable to assume that since the NSSS of Units 1 and 2 are identical, the pre-EPU core inventories utilized for the original

assessments were essentially similar, with minor differences resulting from library differences between the two different (but of the same vintage) computer codes utilized to develop the cores (i.e., the BVPS-1 core inventory had been developed by QUADREX Corp using industry computer code ORIGEN, whereas the BVPS-2 core inventory was developed via the S&W QA I Computer Code ACTIVITY2 which uses fission yields taken from an ORIGEN library)

The impact of EPU on the post-LOCA gamma radiation dose rates at BVPS-1 and BVPS-2 is therefore evaluated based on a comparison of the gamma source terms developed based on the original core inventory used to develop the post-LOCA dose rates at BVPS-2, to the EPU gamma source terms applicable to both units. The approach utilizes scaling techniques based on a source term comparison, rather than developing new dose rate estimates at the various locations, using the new core inventory.

Computer code PERC2 is used to develop the post-LOCA gamma energy release rates (Mev/sec) per energy group vs. time for the EPU as well as the composite pre-EPU core inventory for the following post-accident radiation sources:

- Pressurized recirculating fluid (i.e., the radiation source contains 100% core noble gases, 50% of the core halogens and 1% of the remainder fission products/actinides; this type of source could be present in the post-accident sampling system).

- Containment atmosphere in containment (i.e., the radiation source contains 100% core noble gases, 50% of the core halogens).

- Containment atmosphere in lines outside containment (i.e., the radiation source contains 100% core noble gases, 25% of the core halogens).

- Sump water (i.e., the radiation source contains 50% of the core halogens and 1% of the remainder fission products/actinides; this is the source expected in recirculating fluids following a large break LOCA).

- Plateout surfaces resulting in accumulated halogens (i.e., the radiation source are the core halogens).

For the "unshielded" case, the factor impact on post-accident gamma dose rates is estimated by ratioing the gamma energy release rates weighted by the flux to dose rate conversion factor, as a function of time, for the EPU power level, to the corresponding weighted source terms based on the pre-EPU power level. Note that the unshielded case for sump water includes a thin shield of several inches of water to eliminate the weak photons that will not escape the self-absorption of sump water.

To evaluate the factor impact of the EPU on post-LOCA gamma dose rates (vs. time) in areas that are "shielded," the pre-EPU and EPU source terms discussed above are weighted by the concrete shielding factors for each energy group. The concrete shielding factors, for 2 feet of concrete (representative of moderate shielding), and 4 feet of concrete (representative of heavy shielding), provide a basis for comparison of the post-LOCA spectrum hardness of source terms with respect to time for both original design and EPU cases.

Since the EPU gamma dose rate scaling factors will vary with source, time as well as shielding, to cover all types of analysis models/assessments, the maximum dose rate scaling factor developed from all of the above assessments is used, for the most part, for all source/receptor combinations, with or without shields, and at all time periods after LOCA. Exceptions include a few cases where analysis refinement was performed to support operator exposure compliance within regulatory limits at EPU conditions.

5.11.8.3 Conclusions

The impact of EPU on the estimated post-LOCA environmental dose rates in the various areas of BVPS-1 and BVPS-2, and the associated operator dose per licensing basis mission at BVPS-2 has been assessed.

The worst case post-LOCA dose rate scaling factor at BVPS-1 and BVPS-2 is estimated to be 1.26, and reflects EPU as well as the impact of (a) extended burn fuel and (b) the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses.

The impact of the above assessment on the unit specific vital access licensing commitments are as follows:

BVPS-1: In support of operations at EPU conditions, the estimated post-LOCA dose rates in areas identified in BVPS-1 Health Physics Procedure as requiring access, will increase by a factor of 1.26

BVPS-2: Except as noted, the operator dose estimates while performing vital functions following a LOCA will increase by a maximum factor of 1.26. The EPU assessment for the following access route included additional refinement to ensure that the operator dose remained within 5 Rem.

- The operator dose for Access Route No. 13 (Obtaining and Analyzing a Post-Accident Effluent Sample) is updated by utilizing the "source specific" (i.e., the analysis takes into consideration that the only source the operator is exposed to is containment atmosphere with 100% noble gases, 50% halogens) "time specific" maximum (i.e., most limiting of the estimated scaling factors, at the time of access and subsequent sample and analysis period, for containment atmosphere sources that are unshielded, have moderate shielding or have heavy shielding) dose rate scaling factor which is estimated to be 1.0. It is therefore concluded that EPU will not impact the operator dose estimates associated with this access route.

The post-LOCA BVPS-2 vital access mission doses summarized in Table 5.11.8-1 demonstrates continued compliance with the NUREG 0737 II.B 2 "5 Rem" operator exposure limit per mission following EPU.

5.11.9 Post-Accident Site Boundary and Control Room Doses.

5.11.9.1 Introduction

As discussed in Sections 5.11.1 and 5.11.2, as holder of operating licenses issued prior to January 10, 1997, and in accordance with 10CFR50.67 and Standard Review Plan 15.0.1, BVPS proposes to revise the accident source terms used in the BVPS-1 and BVPS-2 EPU design basis site boundary and control room dose analyses to reflect the full implementation of Alternative Source Terms (AST) as detailed in Regulatory Guide 1.183.

The impact of EPU on the site boundary and control room doses are discussed for the following accidents applicable to BVPS licensing basis:

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

Note that the LOCA and the CREA are addressed in this application by reference only, since the referenced accident analyses, approved by the NRC in its SER for OL Amendment Nos. 257 and 139, were performed at EPU conditions.

At BVPS, the SLB Outside Containment, LACP, MSLB, SGTR and WGSR are not directly impacted by the implementation of the AST as there is no accident initiated fuel damage associated with these events. However, with this application and the full implementation of AST at BVPS, the dose acceptance criteria of 10CFR50.67 become applicable to all of the accidents listed in Regulatory Guide 1.183 which include the MSLB, and the SGTR. It is noted that the SLB Outside Containment, the LACP and the WGSR are not addressed in Regulatory Guide 1.183. The dose criteria to which they are evaluated are discussed in Section 5.11.2.

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

The 0 to 30-day dose to an operator in the control room due to airborne radioactivity releases is developed for each of the design basis accidents. This represents the post-accident dose to the operator due to inhalation and submersion. The CR shielding design is based on the LOCA, which represents the worst

case DBA relative to radioactivity releases. The direct shine dose due to contained sources/external cloud is included in the CR doses reported for the LOCA.

The updated site boundary and control room dose analyses reflect EPU conditions, AST (as applicable), and except as noted, bounding parameter values to encompass an event at either unit. In addition the parameter values assigned to the BVPS-1 steam generators reflect the Replacement Steam Generators. The analyses for both units reflect a SG tube leakage rate of 150 gpd/SG. The MSLB, the SGTR, the WGSR, and the FHA dose analyses are unit specific. In accordance with the current licensing basis, the BVPS-2 EPU MSLB dose analysis reflects the use of Alternative Repair Criteria (ARC) and addresses an accident induced Steam Generator tube leakage. Note that ARC is not utilized in the BVPS-1 MSLB dose analysis because it is not applicable to the Model 54F Replacement Steam Generators. The SGTR dose analyses for BVPS-1 and BVPS-2 reflect environmental releases based on the unit-specific licensing basis mass and energy release calculation methodology.

It is noted that the control room dose analyses reflect a control room design consistent with that approved by the NRC in its SER for OL Amendment Nos. 257 and 139. Specifically, the approved design changes include:

- Conservative estimates of control room unfiltered inleakage that envelope the results of recent tracer gas testing performed in the year 2001, and provide margin for potential surveillance tests.
- Revised Technical Specification acceptance criteria for the BVPS-1 control room HEPA and charcoal filters which will make the BVPS-1 acceptance criteria similar to the more limiting criteria currently listed for the BVPS-2 control room filters.
- Elimination of taking credit for the automatic initiation feature of the safety related control room monitors to initiate the control room emergency pressurization system.
- Manual initiation of CREVS at T=30 minutes for those events that take credit for the control room emergency ventilation system.
- Updated control room atmospheric dispersion factors using ARCON96 methodology for release points associated with LOCA and CREA.

In addition, the BVPS-1 FHA and the BVPS-1 and BVPS-2 MSLB and SGTR take credit for a 30 minute control room purge after the accident sequence is complete and the environmental release has been terminated.

Except as noted, the accident analyses considers a Loss of Offsite Power (LOOP) at T=0 hours or immediately subsequent to the accident if determined by the accident progression (e.g., the SGTR). The impact of a LOOP "significantly later" on in the accident, (such as during the fuel release phase of a LOCA), is not addressed per NRC Information Notice 93-17 (Reference 25). IN 93-17 concludes that plant design should reflect all credible sequences of the LOCA/LOOP, but states that a sequence of a LOCA and an unrelated LOOP is of very low probability and is not a concern. Likewise, a LOOP is not taken into consideration when evaluating the dose consequences of accidents such as the FHA and the WGSR, as these events, in themselves, cannot cause a reactor trip.

5.11.9.2 Accident Atmospheric Dispersion Factors

Site Boundary Atmospheric Dispersion Factors

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (χ/Q) for BVPS-1 and BVPS-2 remain unchanged by this application and are consistent with current licensing basis. These values are also the same as utilized for the LOCA and the CREA in Reference 5 and are presented in Table 5.11.9-1. As discussed earlier, with the exception of the MSLB, the SGTR, the WGSR and the FHA which have unit specific analysis, for the purposes of performing bounding analyses representative for both units, the BVPS-2 EAB χ/Q's will be utilized in assessing the impact of EPU on all of the remaining design basis accidents listed in Section 5.11.9.1. As noted in Table 5.11.9-1, the LPZ χ/Q's are the same for both units.

On-Site Atmospheric Dispersion Factors

The control room χ/Q values for the environmental release paths associated with the design basis accidents listed for each unit in Section 5.11.9.1, are calculated using the latest version of the "Atmospheric Relative Concentrations in Building Wakes" (ARCON96) methodology (Ramsdell, 1997, Reference 20).

The methodology utilized to develop these atmospheric dispersion factors is discussed in detail in Section 5.3.4.2 of Reference 5. All releases are conservatively treated as ground-level as there are no releases at this site that are high enough to escape the aerodynamic effects of the plant buildings (i.e., 2.5 times Containment Building height, U.S. NRC, 1982).

As noted in Reference 5, the control room air intake χ/Q values are representative of the worst case χ/Q values for control room unfiltered in-leakage since the distances and directions from the release points to these receptors are very similar.

- Control room tracer gas tests have indicated that potential sources of unfiltered inleakage into the control room during the post-accident pressurization mode are the normal operation dampers associated with the control room ventilation system. The same χ/Q as that of the Control Room air intake are assigned to this location.

- The other source of inleakage is potentially that associated with ingress /egress and leakage via door seals. This inleakage is assigned to the door leading into the control room that is considered the point of primary access. This door is located in-between the BVPS-1 and BVPS-2 control room air intakes and is located close enough to the air intakes to allow the χ/Q associated with the air intakes to be assigned to this source of inleakage.

The BVPS-1 and 2 χ/Q values for all release-receptor combinations utilized for the design basis accidents discussed in Section 5.11.9.1 are summarized in Tables 5.11.9-2a and 5.11.9-2b, respectively.

5.11.9.3 Dose Calculation Methodology

With the exception of the WGSR which maintains the current licensing basis methodology, the dose calculation methodology is similar to that outlined in Section 5.3.5 of Reference 5. As noted in Reference 5, computer program PERC2 is used to calculate the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from submersion due to halogens, noble gases and other nuclides at the offsite locations and in the control room. The CEDE is calculated using the Federal Guidance Report No.11, Sept. 1988 (Reference 21) dose conversion factors. The committed doses to other organs due to inhalation of halogens, noble gas, other nuclides and their daughters are also calculated. PERC2 is a multiple compartment activity transport code with the dose model consistent with the regulatory guidance. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included.

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

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

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

Where:

Dh(j)	=	Committed Effective Dose Equivalent (rem) from isotope j
A(j)	=	Integrated Activity (Ci-s/m3)
h(j)	=	Isotope j Committed Effective Dose Equivalent (CEDE) dose conversion factor (mrem/pCi) based on Federal Guidance Report No. 11, Sept. 1988
C2	=	Unit conversion of 1x1012 pCi/Ci
C3	=	Unit conversion of 1x10-3 rem/mrem
CB	=	Breathing rate (m^3/s)
CO	=	Occupancy factor

Deep Dose Equivalent (DDE) from External Exposure – According to the guidance provided in Section 4.1.4 and Section 4.2.7 of RG 1.183, the Effective Dose Equivalent (EDE) may be used in lieu of DDE in determining the contribution of external dose to the TEDE if the whole body is irradiated uniformly. The EDE in the control room is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the control room. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes, the integrated activity in the control room air space, the photon energy release rates per energy group from activity airborne in the control room, and the ANSI/ANS 6.1.1-1991 "neutron and gamma-ray fluence-to-dose factors" (Reference 22).

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

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

$E\gamma_{BAR}$	=	average gamma released per disintegration (Mev/dis)
$\psi(x,y,0)$	=	concentration time integral (Ci-sec/m3)
0.25	=	$[\ 1.11 \times 1.6\text{x}10^{-6} \times 3.7\text{x}10^{10}\ ] / [\ 1293 \times 100 \times 2\ ]$

where:

1.11	=	ratio of electron densities per gm of tissue to per gm of air
$1.6\text{x}10^{-6}$ (erg/Mev)	=	number of ergs per Mev
$3.7\text{x}10^{10}$ (dis/sec-Ci)	=	disintegration rate per curie
1293 (g/m3)	=	density of air at S.T.P.
100	=	ergs per gram per rad
2	=	factor for converting an infinite to a semi-infinite cloud

5.11.9.4 Control Room Design/Operation/Transport Model

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

The BVPS-2 Control Room Emergency Ventilation System (CREVS) system is safety-related, fully automated, and fully compliant with relevant regulatory requirements. In the unlikely event that neither of the BVPS-2 trains can be put in service, operator action may be utilized to initiate the BVPS-1 control room filtered emergency pressurization system. To ensure bounding values, the atmospheric dispersion factors utilized for the identified release paths per accident reflect the limiting control room intake for each time period.

During normal plant operation, both ventilation intakes are operable providing a total supply of 500 cfm of unfiltered outside air makeup which includes all potential inleakage and uncertainties.

For purposes of dose assessment, no credit is taken for automatic initiation of the control room emergency ventilation system following any design basis accident. For events that take credit for operation of the CREVS, the analyses assume manual initiation, and that a pressurized control room is available at T=30 minutes after the accident. For selected accidents, credit is taken for control room clean-up via a half-hour control room purge, at a purge flow rate of 16,200 cfm, after the environmental release due to the accident is terminated.

As discussed in Section 5.11.9.3, and Reference 5, and in accordance with the current licensing basis, the atmospheric dispersion factors associated with control room inleakage are considered to be the same as those utilized for the control room intake. The estimated inleakage envelopes the results of recent tracer

gas testing performed in the year 2001, include 10 cfm for ingress/egress, and provide margin for potential deterioration between surveillance tests.

As noted in Reference 5, the control room emergency filtered ventilation intake flow varies 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 600 cfm in the pressurized mode as it is more limiting. Although the filtered intake of radioisotopes is higher at the larger intake rate of 1030 cfm, it is small compared to the radioactivity entering the control room, in both cases, due to unfiltered inleakage. Consequently, the depletion of airborne activity in the control room via the higher outleakage rate of 1030 cfm make the lower intake rate of 600 cfm more limiting from a dose consequence perspective. This argument holds true because the CEDE from inhalation is far more limiting than the DDE from immersion which is principally from noble gases.

Table 5.11.9-3 lists key assumptions and input parameters associated with BVPS control room design utilized in Reference 5 and applicable to the EPU analyses.

5.11.9.5 Loss of Coolant Accident (LOCA)

The dose consequences at the site boundary and in the control room following a LOCA is addressed in this application by reference only, since, the LOCA analysis, approved by NRC in its SER for OL Amendment Nos' 257 and 139, was performed at EPU conditions and utilized Alternative Source Terms. The methodology utilized to analyze the LOCA is discussed in Section 5.3.6.3 of Reference 5.

The EAB, LPZ and Control Room doses following a LOCA at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.6 Control Rod Ejection Accident (CREA)

The dose consequences at the site boundary and in the control room following a CREA is addressed in this application by reference only, since, the CREA analysis, approved by NRC in its SER for OL Amendment Nos' 257 and 139, was performed at EPU conditions and utilized Alternative Source Terms. The methodology utilized to analyze the CREA is discussed in Section 5.3.6.4 of Reference 5.

The EAB, LPZ and Control Room dose following a CREA at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.7 Main Steam Line Break (MSLB) Outside Containment

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

The BVPS-1 and BVPS-2 MSLB dose assessments follow the guidance provided in RG 1.183. In addition, the BVPS-2 assessment supports the implementation of Alternate Repair Criteria (ARC) as defined in USNRC GL 95-05 (Reference 24) and previously approved in BVPS-2 License Amendment Number 101. In accordance with GL 95-05, the BVPS-2 MSLB dose assessment determines the

maximum accident induced leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the EAB, LPZ and the control room.

Tables 5.11.9-4a and 5.11.9-4b list the key parameters utilized to develop the radiological consequences following the MSLB at BVPS-1 and BVPS-2, respectively.

BVPS-1

The radiological model used for the MSLB assessment conservatively assumes immediate dry-out of the faulted SG following a MSLB resulting in the instantaneous release of all of the SG contents, which are assumed at maximum Technical Specification concentrations. Based on an assumption of a simultaneous Loss of Offsite Power, the condenser is unavailable, and environmental steam releases via the MSSVs/ADVs from the intact steam generators are used to cool down the reactor until the Residual Heat Removal (RHR) system starts shutdown cooling. The elevated iodine activity in the RCS due to a postulated pre-accident or concurrent iodine spike, as well as the noble gas (at Technical Specification concentrations), leak into the faulted and intact steam generators, and are released to the environment from the break point, and from the MSSVs/ADVs, respectively.

The steam releases from the intact SGs continue until shutdown cooling is initiated via operation of the RHR system at T= 8 hrs, resulting in the termination of environmental releases via this pathway. The releases from the faulted SG due to primary to secondary leakage continues until T=19 hrs (i.e., estimated time for the RHR System to bring the primary coolant temperature down to 212°F). ~~It is noted that credit is taken for a 56% reduction in the primary to secondary tube leakage in the faulted SG after 8 hrs to reflect the reduced RCS pressure at RHR cut-in vs. RCS pressure at accident initiation (i.e., 425 psid vs. 2235 psid).~~

Since there is no postulated fuel damage associated with this accident, the primary radiation source is the activity in the reactor coolant system. Two iodine spiking cases are addressed: a pre-accident iodine spike and a concurrent iodine spike. The analysis uses the proposed BVPS-1 Technical Specification limits for the primary and secondary coolant activity.

a. Pre-accident spike - initial primary coolant iodine activity is assumed to be at 21 μCi/gm of DE I-131, (proposed Technical Specification transient limit for full power operation). Initial primary coolant noble gas activity is assumed to be at the proposed Technical Specification levels.

b. Concurrent spike - the initial primary coolant iodine activity is assumed to be at 0.35 μCi/gm DE I-131 (proposed equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 0.35 μCi/gm DE I-131 coolant concentration. In accordance with the current design basis, the duration of the assumed spike is four hours. The initial primary coolant noble gas activity is assumed to be at the proposed Technical Specification levels.

The initial secondary coolant iodine activity is assumed to be at the proposed Technical Specification limit of 0.1 μCi/gm DE I-131.

Following a MSLB, the primary and secondary reactor coolant activity is released to the environment via two pathways; i.e., the MSLB location and the MSSVs/ADVs. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes (identified for purposes of assessment as the BVPS-1 MSSVs/ADVs to the BVPS-1 CR intake, and the BVPS-1 MSLB location to the BVPS-1 Intake) are selected to determine a bounding control room dose.

Faulted Steam Generator

The release from the faulted Steam Generator occurs via the postulated break point of the main steam line. The faulted steam generator is conservatively assumed to dry-out instantaneously following the MSLB, releasing all of the iodines in the secondary coolant that was initially contained in the steam generator. The secondary steam initially contained in the faulted steam generator is also released; however, this contribution is not included in this analysis since the associated radioactivity is insignificant compared to the other contributions. The primary to secondary leakage reflects 150 gpd at STP. All iodine and noble gas activities in the referenced tube leakage are released directly to the environment without hold-up or decontamination.

Intact Steam Generator

The release from the two remaining intact steam generators (used to cool the reactor and the primary system) occur via the plant MSSVs/ADVs. The iodine activity in the intact SG liquid is released to the environment in proportion to the steaming rate and the partition factor. The steam releases from the MSSVs/ADVs terminate at 8 hours after the event when shutdown cooling is initiated via the RHR System.

BVPS-2

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

In accordance with the guidance provided in GL95-05, 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 MSLB dose analysis is performed to establish a maximum allowable accident-induced leakage, against which the cycle leakage projections can be compared. The accident induced leakage rate is the maximum primary-to-secondary SG tube leakage that could occur with the offsite or control room operator doses remaining within applicable limits. For analysis purposes, this tube leakage is conservatively assigned to the faulted SG. Consequently, the primary-to-secondary leakage in the faulted steam generator reflects 150 gpd at STP, plus the maximum allowable accident induced tube leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the EAB/LPZ and the control room.

BVPS-2 specific parameters as noted in Table 5.11.9-4b are utilized. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes (identified for purposes of assessment as the BVPS-2 MSSVs/ADVs to the BVPS-2 CR intake, and the BVPS-2 MSL break location to the BVPS-2 Intake) are selected to determine a bounding control room dose.

EAB 2 hr Worst Case Window

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

- The Source/Release for the Pre-incident Spike Case is at its maximum levels between 0 and 2 hours.
- The Source/Release for the Concurrent Spike Case is at its maximum levels between 4 (end of the spiking period) and 6 hours.

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

Accident Specific Control Room Model Assumptions

The control room emergency ventilation system is manually initiated and a pressurized control room is available at T=30 mins after the accident. Following termination of the environmental release, the control room is purged, at T=24 hrs, at a rate of 16,200 cfm, for a period of 30 mins. The remaining CR parameters utilized in this model are discussed in Section 5.11.9.4.

The EAB, LPZ and Control Room dose following a MSLB at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.8 Steam Generator Tube Rupture (SGTR)

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

The dose assessments follow the guidance provided in Regulatory Guide 1.183. Tables 5.11.9-5a and 5.11.9-5b list the key parameters utilized to develop the radiological consequences following a SGTR at BVPS 1 and BVPS-2, respectively.

BVPS-1

The SGTR results in a reactor trip and a simultaneous loss of offsite power at 225 seconds after the event. Due to the tube rupture the primary coolant with elevated iodine concentrations (pre-accident or concurrent iodine spike) flows into the faulted steam generator and the associated activities are released to the environment via secondary side steam releases. Before the reactor trip, the activities are released from the air ejector. After the reactor trip the steam release is via the MSSVs/ADVs. The spiking primary coolant activities leaked into the intact steam generator at the maximum allowable primary-to-secondary leakage value are also released to the environment via secondary steam releases. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes (identified for purposes of assessment as the BVPS-1 MSSVs/ADVs to the BVPS-1 CR intake, and the BVPS-1 Air Ejector to the BVPS-1 Intake) are selected to determine a bounding control room dose.

Since there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary coolant system. Two spiking cases are addressed: a pre-accident iodine spike and a concurrent iodine spike. The analysis uses the proposed BVPS-1 Technical Specification limits for the RCS and secondary coolant.

a. Pre-accident spike - the initial primary coolant iodine activity is assumed to be 21 μCi/gm of DE I-131 (proposed BVPS-1 transient Technical Specification limit for full power operation). The initial primary coolant noble gas activity is assumed to be at the proposed BVPS-1 Technical Specification levels.

b. Concurrent spike - the initial primary coolant iodine activity is assumed to be 0.35 μCi/gm DE I-131 (proposed BVPS-1 equilibrium Technical Specification limit for full power operation). Immediately following the accident, the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 335 times the equilibrium appearance rate corresponding to the 0.35 μCi/gm DE I-131 coolant concentration. In accordance with the current design basis, the duration of the assumed spike is 4 hours. The initial primary coolant noble gas activity is assumed to be at the proposed BVPS-1 Technical Specification levels.

The initial secondary side liquid and steam activity is relatively small and its contribution to the total dose is small compared to that contributed by the rupture flow. However, the release of the secondary side liquid activity and the resultant doses are also included in this analysis. The initial secondary side iodine activity is assumed to be at the proposed BVPS-1 Technical Specification limit of 0.1 μCi/gm DE I-131.

Faulted SG Release

A postulated SGTR will result in a large amount of primary coolant being released into the faulted steam generator via the break location with a significant portion of it flashed to the steam space. The noble gases in the break flow and the iodine in the flashed flow are assumed immediately available for release from the steam generator without retention. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and partition factor. Before the reactor trip at 225 seconds, the activities in the steam are released to the environment from the main condenser air ejector. All steam noble gases and organic iodine are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returned to all three steam generators and assumed to be available for future steaming release. After the reactor trip, the break flow continues until the primary system is fully depressurized. No credit is taken for the condenser, since, a LOOP is assumed to occur simultaneously with the reactor trip. The steam is released from the MSSVs/ADVs. All activity releases from the faulted steam generator cease when it is isolated at 30 minutes after the accident.

Intact SG release

The activity release from the intact steam generator is due to normal primary-to-secondary leakage and steam release from the secondary side. The Primary-to-Secondary leak rate is assumed to be 150 gpd per SG. All of the iodine activity in the referenced leakage is assumed to mix uniformly with the steam generator liquid and released in proportion to the steaming rate and the partition factor. Before the reactor

trip at 225 seconds, the steam is released from the main condenser air ejector. After the reactor trip, the steam is released from the MSSVs/ADVs. The reactor coolant noble gases that enter the intact steam generator are released directly to the environment without holdup. The steam release from the intact steam generator continues until initiation of shutdown cooling 8 hours after the accident.

Release of Initial SG Liquid Activity

The initial iodine inventory in the steam generator liquid is assumed to be at Technical Specification levels and is released to the environment, due to steam releases, via the condenser/air ejector before reactor trip, and via the MSSVs/ADVs after reactor trip. The release from the faulted SG stops at T=30 mins. The release from the intact SGs continue until 8 hrs after the accident.

BVPS-2

Except as noted, the BVPS-2 dose assessment utilizes the same methodology discussed above for BVPS-1. The analysis utilizes BVPS-2 specific parameters as noted in Table 5.11.9-5b. It is noted that the steam release from the faulted SG includes a short period release between 2 and 8 hrs when the faulted SG is manually depressurized in preparation for RHR operation. The most limiting atmospheric dispersion factors for each of the release points relative to the two CR intakes (identified for purposes of assessment as the BVPS-2 MSSVs/ADVs to the BVPS-2 CR intake, and the BVPS-2 air ejector to the BVPS-2 Intake) are selected to determine a bounding control room dose.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. The major source for the SGTR is the flashed portion of the RCS break flow which is terminated **before T=2 hrs.** Therefore the worst 2-hr window dose for both the pre-accident and accident initiated spike case occurs during T=0 hr to T=2 hrs after the accident.

Accident Specific Control Room Model Assumptions

No credit is taken for initiation of the control room emergency ventilation system following a SGTR. Following termination of the environmental release, the control room is purged, at T=8 hrs, at a rate of 16,200 cfm, for a period of 30 mins. The remaining CR parameters utilized in this model are discussed in Section 5.11.9.4.

The EAB, LPZ and Control Room dose following a SGTR at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.9 Locked Rotor (LR) and Loss of AC Power (LACP)

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a LR accident at BVPS-1 or BVPS-2 at EPU conditions. As noted in Section 5.11.1, bounding parameter values are used to encompass an event at either unit.

The dose assessment follows the guidance provided in Regulatory Guide 1.183. Table 5.11.9-6 lists the key assumptions and parameters utilized to develop the radiological consequences following a BVPS LR accident. Table 5.11.9-7 lists the key parameters associated with a BVPS LACP. The transport models associated with the two events are similar with the exception that the LR event results in fuel damage and associated release of gap activity, whereas the LACP has no fuel damage, and the maximum release is associated with Technical Specification concentrations. Since the RCS Technical Specification activity is significantly smaller than the gap activity associated with failed fuel, it is concluded that the dose consequences of the LR bound that of the LACP.

A BVPS LR accident results in less than 20% failed fuel and a release of the associated gap activity. The gap activity (consisting of noble gases, halogens and alkali metals) are instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary steam generator tube leakage assumed to be at the value of 450 gpd @STP.

A radial peaking factor of 1.75 is applied to the activity release. The chemical form of the iodines in the gap are assumed to be 95% CsI, 4.85% elemental and 0.15% organic. At BVPS, the SG tubes remain submerged for the duration of the event; therefore, the gap iodines are assumed to have a partition coefficient of 100 in the SG. The iodine releases from the SG are assumed to be 97% elemental and 3% organic. The gap noble gases are assumed to be released freely to the environment without retention in the SG, whereas the particulates are carried over in accordance with the design basis SG moisture carryover fraction.

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

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

The most limiting atmospheric dispersion factors between the MSSVs/ADVs at each unit relative to the two CR intakes (identified for purposes of assessment as the BVPS-1 MSSVs/ADVs to the BVPS-1 CR intake) is selected to determine a bounding control room dose.

EAB 2-Hour Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For the LR event, the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the 6-8 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=8 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB X/Q is utilized.

Accident Specific Control Room Model Assumptions

The control room is conservatively assumed to remain in the normal operation mode. The remaining CR parameters utilized in this model are discussed in Section 5.11.9.4.

The EAB, LPZ and Control Room dose following a LR event at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.10 Fuel Handling Accident (FHA) in the Fuel Pool or in the Containment

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

The dose assessment follows the guidance provided in Regulatory Guide 1.183. Table 5.11.9-8 lists some of the key assumptions and parameters utilized to develop the radiological consequences following a BVPS FHA.

BVPS-1

At BVPS, plant Technical Specifications prohibit initiation of fuel handling activities in the Fuel Pool or in the Containment until 100 hours after reactor shut down. Analyses of the deflections and resulting stresses on the dropped fuel results in the damage of 137 of the 264 rods in a fuel assembly. All of the fuel gap activity associated with the damaged rods is assumed to be released. A radial peaking factor of 1.75 is applied to the core average gap activity. The activity (consisting of noble gases, halogens, and alkali metals) is released in a "puff" to the fuel pool or reactor cavity.

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

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

The exhaust flows from the containment and Fuel Pool Area may be directed out of the SLCRS release point. However, since the containment and fuel buildings are "open", releases could also occur from anywhere along the containment wall (e.g., via the equipment or personnel hatch) or via the fuel building normal operation release point, i.e., the ventilation vent. Because the location of the release is unknown, the worst case dispersion factor (identified for purposes of assessment as that associated with the BVPS-1

ventilation vent to the BVPS-1 CR intake) is used without taking any credit for SLCRS flows or filtration in this analysis.

BVPS-2

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

The site boundary atmospheric dispersion factors reflect unit specific values. The most limiting atmospheric dispersion factor associated with the potential BVPS-2 release points relative to the two CR intakes (identified for purposes of assessment as the BVPS-2 ventilation vent to the BVPS-2 CR intake) is selected to determine the bounding control room dose.

In addition, as discussed below and in Table 5.11.9-8, the analyses reflect the differing control room ventilation system operational requirements during fuel handling at each unit.

EAB 2 hr Worst Case Window

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

Accident Specific Control Room Model Assumptions

No credit is taken for initiation of the control room emergency ventilation system following a FHA. Subsequent to a BVPS-1 FHA, following termination of the environmental release, the control room is purged, at T=2 hrs, at a rate of 16,200 cfm for a period of 30 mins. For the BVPS-2 FHA, the control room is assumed to remain in the normal operation mode. The remaining CR parameters utilized in these models are discussed in Section 5.11.9.4.

The EAB, LPZ and Control Room dose following a FHA at BVPS-1 and BVPS-2 at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.11 Small Line Break (SLB) Outside Containment

Computer program PERC2 is used to calculate the control room and site boundary doses due to airborne radioactivity releases following a SLB outside Containment at BVPS-1 or BVPS-2 at EPU conditions. As noted in Section 5.11.1, bounding parameter values are used to encompass an event at either unit.

Regulatory Guide 1.183 does not address a SLB outside Containment. The dose assessment herein follows the current BVPS licensing basis model, but for purposes of consistency, uses the most limiting dose limits set by Regulatory Guide 1.183 for accident evaluations. Table 5.11.9-9 lists the key assumptions and parameters utilized to develop the radiological consequences following a SLB outside Containment at BVPS.

The SLB outside containment postulates the break of the 2 inch RCS letdown line in the Auxiliary Building resulting in a maximum break flow of 16.79 lbm/sec. Thirty seven percent of the break flow is calculated to flash. The iodine activity in the break flow is assumed to become airborne in proportion to the flash fraction, whereas the noble gases are assumed to be airborne and discharged to the environment without decontamination or holdup.

Since there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary coolant system. In accordance with current licensing basis and SRP 15.6.2 (Reference 37), a concurrent iodine spike is included in the source term.

The initial primary coolant iodine activity is assumed to be 0.35 μCi/gm DE I-131 (current equilibrium BVPS-2 and proposed equilibrium BVPS-1 Technical Specification limit for full power operation). Immediately following the accident, the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 0.35 μCi/gm DE I-131 coolant concentration. In accordance with the current design basis, the duration of the assumed spike is 4 hours. The iodine released from the RCS is assumed to be 97% elemental and 3% organic.

The activity in the Auxiliary Building is released to the environment via the Ventilation Vent. The most limiting atmospheric dispersion factors between the ventilation vent release point at each unit relative to the two CR intakes (identified for purposes of assessment as the BVPS-1 Ventilation Vent to the BVPS-1 CR intake) is selected to determine a bounding control room dose. No credit is taken for Auxiliary building holdup or filtration. The break flow is isolated by manual operator action after a period of 15 minutes.

EAB 2 hr Worst Case Window

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

Accident Specific Control Room Model Assumptions

The control room is assumed to remain in the normal operation mode. The remaining CR parameters utilized in this model are discussed in Section 5.11.9.4.

The EAB, LPZ and Control Room dose following a SLB outside Containment at EPU conditions are presented in Tables 5.11.9-11 and 5.11.9-12.

5.11.9.12 Waste Gas System Rupture (WGSR)

The WGSR dose analysis was re-done using current licensing basis methodology and criteria, and using parameters that reflect EPU conditions. Also, the control room atmospheric dispersion factors utilized for the dose consequence analysis (see Tables 5.11.9-2A and 5.11.9-2B are based on ARCON96 methodology which was previously approved for use at BVPS for release points associated with LOCA and CREA. The impact of EPU is small and is limited to slight changes in RCS mass and nuclide specific activities while operating with 1 percent failed fuel. Any increase in offsite or control room operator doses resulting from these changes is minimal. This change meets the criteria for implementation via the 10 CFR 50.59 process. Therefore, this accident re-analysis is not being included for NRC review.

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Accident Specific Control Room Model Assumptions

The control room is assumed to remain in the normal operation mode. The CR parameters utilized in this model are discussed in Section 5.11.9.4.

5.11.10 Conclusions

The radiological analyses and evaluations documented in this section demonstrate that EPU to 2900 MWt will not impact compliance with applicable regulatory radiological dose limits for normal operation and for accidents.

The radiological impact of EPU was been evaluated for the following:

- Normal Operation Dose Rates and Shielding
- Normal Operation Annual Radwaste Effluent Releases
- Radiological Environmental Doses for Equipment Qualification (EQ)
- Post-LOCA Access to Vital Areas
- Post-Accident Site Boundary and Control Room Doses

The regulatory acceptance criteria being utilized in the EPU assessments are discussed in Section 5.11.2. It is noted that as part of the EPU application, the accident source term used in the BVPS-1 and BVPS-2 are design basis site boundary and control room dose analyses reflect the *"full"* implementation of Alternative Source Terms (AST) as provided in 10CFR50.67 and Regulatory Guide 1.183.

The conclusions of the EPU evaluation are summarized below.

5.11.10.1 Normal Operation Dose Rates and Shielding

The normal operation radiation levels in most areas of the plant are expected to increase by the percentage of the EPU; i.e., approximately 7.9%. The increase in expected radiation levels will not affect radiation zoning or shielding requirements in the various areas of the plant.

Individual worker exposure is maintained to within acceptable limits by the site Radiological Control program which controls access to radiation areas. In addition, procedural controls and ALARA considerations can be used to control exposure in situations that involve increased radiation levels.

5.11.10.2 Normal Operation Annual Radwaste Effluent Releases

EPU will have negligible impact on the annual radwaste effluent releases/doses (i.e., all doses are expected to remain within a small percentage of the allowable Appendix I doses). It is therefore concluded that following EPU, the liquid and gaseous radwaste effluent treatment system will not require modification in order to keep offsite doses within the requirements of 10 CFR 50 Appendix I.

5.11.10.3 Radiological Environmental Doses for Equipment Qualification (EQ)

The impact of EPU on the estimated normal operation and post-accident total integrated environmental doses in the various areas of the plant is estimated using scaling techniques and is summarized below. Note that the scaling factors reflect EPU as well as the impact of (a) the use of extended burn fuel; and (b) the more advanced fuel burnup modeling/libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses.

Normal Operation Dose

The estimated maximum normal operation EQ dose EPU scaling factor of 1.06 reflects the fact that the existing calculated values are based on a power level of 2766 MWt, and 1% defective fuels (design basis source terms) in the RCS. The EQ dose EPU scaling factor of 1.0 at locations outside containment but near primary coolant and down stream sources takes into consideration a limiting upper bound primary coolant activity based on the Plant Technical Specifications.

Accident Dose

The estimated accident environmental dose EPU scaling factors are based on TID 14844 source terms and are summarized below.

- Doses based on the LOCA: bounding EPU gamma and beta dose scaling factor is 1.19 and 1.17, respectively.
- Doses based on the FHA: bounding EPU gamma and beta dose scaling factors is 1.26 and 1.14, respectively.

The scaling factors discussed above are applied to the existing environmental conditions, to establish the new radiation environmental levels in the EQ zones or at component specific locations. The impact of the EPU radiological environmental conditions on equipment qualification is discussed in Section 10.10.

5.11.10.4 Post-LOCA Access to Vital Areas

The vital access dose assessment for EPU takes into consideration the impact of EPU using scaling techniques and TID 14844 source terms.

The worst case post-LOCA dose rate scaling factor at BVPS-1 and BVPS-2 is estimated to be 1.26, and reflects EPU as well as the impact of (a) the use of extended burn fuel; and (b) the more advanced fuel burnup modeling and radionuclide libraries utilized in development of the EPU core, as compared to the computer code used in the original analyses.

The impact of the above assessment on the unit-specific post-LOCA vital access tasks considering the impacts of EPU concluded that all required EOP steps can be accomplished without exceeding the guidance limits in NUREG-0737 II.B.2.

> **BVPS-1:** As documented in the NRC SER issued to BVPS-1 relative to compliance with NUREG-0737 II.B.2, (References 12 and 13), the BVPS-1 licensing basis does not include estimated doses per operator mission; rather it is a documented evaluation of the worst case post-accident dose rates in plant areas that may need access following a LOCA. The EPU assessment indicates that the estimated post-LOCA dose rates in areas identified in BVPS-1 Health Physics Procedure REOP 2.1 as requiring access, will increase by a maximum of 26% following EPU.
>
> **BVPS-2:** As documented in UFSAR Section 12.3.2.10 and the associated NRC SER (Reference 14), the BVPS 2 compliance with NUREG 0737 II.B.2 is based on ensuring that the vital access dose estimates for identified post-accident missions remain within 5 Rem whole body. The EPU assessment indicates that the operator doses while performing vital functions following a LOCA will remain within the 5 Rem limit imposed by NUREG 0737 II.B.2 following EPU.

Emergency Response Facility (ERF)/Technical Support Center (TSC) Habitability: Post-LOCA habitability of the ERF/TSC is addressed in this application by reference only, since, the LOCA analysis, approved by the NRC in SER for OL Amendment Nos. 257 and 139, addressed ERF/TSC habitability and was performed at EPU conditions. As noted in Section 5.3.7.3.2 of Reference 5, the maximum 30 day dose to the operator in the ERF following a LOCA at either unit, based on containment conversion, AST methodology and EPU conditions will remain within the requirements of 10CFR 50.67 without the need to credit ventilation or filtration systems.

5.11.10.5 Post-Accident Site Boundary and Control Room Doses

In support of EPU, the dose consequences at the site boundary and the control room, for the design accidents applicable to the BVPS licensing basis have been re-analyzed to reflect the full implementation of Alternative Source Terms (AST) as detailed in Regulatory Guide 1.183. Note that the LOCA and the CREA are addressed in this application by reference only, since, the referenced accident analyses, approved by the NRC in SER for OL Amendment Nos. 257 and 139, were performed at EPU conditions.

The SLB outside Containment and the LACP are not addressed in Regulatory Guide 1.183. The acceptance criterion utilized for the SLB outside Containment and the LACP represent the most limiting dose criterion in Table 6 of RG 1.183.

The WGSR is not addressed in Regulatory Guide 1.183. The acceptance criteria for the WGSR remains GDC 19, SRP 6.4 and BTP ETSB 11-5, consistent with current licensing bases.

It is noted that the control room dose analyses reflect a control room design consistent with that approved by the NRC in its SER for OL Amendment Nos. 257 and 139.

In addition, the BVPS-1 FHA and the BVPS-1 and BVPS-2 MSLB and SGTR take credit for a 30 minute control room purge after the accident sequence is complete and the environmental release has been terminated.

It is concluded that following EPU the dose consequences at the site boundary and control room following all design basis accidents will remain within the regulatory requirements of 10CFR50.67, or current licensing basis (applicable only for the WGSR).

5.11.11 References

1. 10CFR50.67, "Accident Source Term."

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

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

4. NRC Safety Evaluation Report enclosing Amendment No. 241 (BVPS-1) and No. 121 (BVPS-2), "Beaver Valley Power Station Units 1&2 – Issuance of Amendment Re: Revised Fuel Handling Accident Safety Analysis and Requirements for Handling Irradiated Fuel Assemblies in the Reactor Containment and in the Fuel Building."

5. Beaver Valley Power Station Units 1 and 2 Licensing Amendment Request (LAR) No's 300 and 172, L-02-069 entitled "Containment Conversion" June 5, 2002.

6. Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1.

7. U.S. Nuclear Regulatory Commission: Amendment No. 243/122 to Facility Operating Licenses No. DPR-66 and NPF-73, Beaver Valley Power Station, BVPS-1 and 2; "1.4 Percent Power Uprate" (TAC No. MB0996, MB0997, and MB2557); Sept. 24, 2001.

8. USNRC Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to a Waste Gas System Leak or Failure", Revision 0, July 1981.

9. TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962.

10. SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," June 30, 1998.

11. NRC SER for Fort Calhoun Station, Amendment No. 201 to Facility Operating License, dated December 5, 2001.

12. Safety Evaluation Report, "NUREG-0737, Item II.B.2 – Design Review of Plant Shielding – Corrective Actions for Access to Vital Areas, Beaver Valley Power Station Unit No. 1," date November 8, 1982.

13. NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

14. NUREG-1057, "Safety Evaluation Report related to the Operation of Beaver Valley Power Station BVPS-2," October 1985.

15. NUREG-0017, Revision 0, April 1976, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors.

16. Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

17. Code of Federal Regulations, Title 10, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Reasonably Achievable" for Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents."

18. Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

19. Safety Guide 25, "Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurizer Water Reactors," March 23, 1972.

20. Ramsdell, J. V. Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes." Prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Rev. 1, May 1997.

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

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

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

24. NRC Generic Letter 95-05, August 3, 1995, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

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

26. NRC Safety Guide 24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," March, 1972.

27. Beaver Valley Power Station 1 & 2 "Annual Radioactive Effluent Release Report – 1997."

28. Beaver Valley Power Station 1 & 2 "Annual Radioactive Effluent Release Report – 1998."

29. Beaver Valley Power Station 1 & 2 "Annual Radioactive Effluent Release Report – 1999."

30. Beaver Valley Power Station 1 & 2 "Annual Radioactive Effluent Release Report – 2000."

31. Beaver Valley Power Station 1 & 2 "Annual Radioactive Effluent Release Report – 2001."

32. Code of Federal Regulations, Title 40, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

33. NUREG 0800, SRP 6.4, Revision 2, "Control Room Habitability System"

34. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 19, "Control Room"

35. Not used.

36. EPA 402-R-93-081, Federal Guidance Report No. 12, "External Exposure to Radio-nuclides in Air, Water and Soil, September, 1993.

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

38. NRC Safety Evaluation Report Enclosing Amendment No. 257 (BVPS-1) and No. 139 (BVPS-2), "Beaver Valley Power Station Units 1 & 2 – Issuance of Amendments Re: Selective Implementation of Alternate Source Terms and Control Room Habitability Technical Specification Changes," September 10, 2003.

39. FENOC Letter L-03-135 to NRC dated September 5, 2003, "Withdrawal of the Containment Conversion Portion (Phase 2) of License Amendment Request Nos. 300 and 172."

40. Beaver Valley Power Station Units 1 & 2 Licensing Amendment Request (LAR) Nos. 316 and 189, L-04-012, January 28, 2004, "Application for Technical Specification Change to Eliminate the Requirements for Hydrogen Recombiners and Hydrogen Analyzers Using the Consolidated Line Item Improvement Process," approved by the NRC on May 19, 2004 in Amendments 259 and 142.

Table 5.11.4-1
Proposed BVPS-1 and Current BVPS-2 Primary and Secondary Coolant Technical Specification Iodine and Noble Gas Activity Concentrations

Nuclide	Primary Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	2.74E-01	8.33E-02
I-132	1.08E-01	1.40E-02
I-133	4.10E-01	9.39E-02
I-134	6.00E-02	1.95E-03
I-135	2.36E-01	3.39E-02
Kr-83m	3.89E-02	--
Kr-85m	1.35E-01	--
Kr-85	1.18E+01	--
Kr-87	9.00E-02	--
Kr-88	2.52E-01	--
Xe-131m	4.84E-01	--
Xe-133m	3.99E-01	--
Xe-133	2.95E+01	--
Xe-135m	9.09E-02	--
Xe-135	9.16E-01	--

Table 5.11.4-2 Proposed BVPS-1 and Current BVPS-2 Primary Coolant Pre-Accident Iodine Spike Activity Concentrations and Equilibrium Iodine Appearance Rates		
Nuclide	**Pre-Accident Iodine Spike Activity Concen. (μCi/gm)**	**Activity Appearance Rates (μCi/sec)**
I-131	16.4	2.53E+03
I-132	6.5	2.66E+03
I-133	24.6	4.42E+03
I-134	3.6	3.00E+03
I-135	14.1	3.41E+03

Table 5.11.4-3
BVPS Core Inventory of Dose Significant Isotopes in the Gap (2918 MWt)

Noble Gases		Halogens		Alkali Metals & BA137M	
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

Table 5.11.8-1
BVPS-2 Post-Accident Vital Access Doses (2918 MWt)

No.	Description	Area or Route	Time after DBA	Occupancy Time of Vital Area	Calculated Whole Body Dose (Rem) Primary	Alternate
1	Control room (CR)	Area	0	Continuous to 30 days	Table 5.11.9-12	NA
2	Emergency Response Facility (ERF)	Area	0	Continuous to 30 days	See Section 5.11.8	NA
4	Operation of post-accident H2 analyzer	Route	30 min	15 min	1.26	2.77
6	Lubrication oil for the diesel generator	Route	96 hrs	3 hr	0.01	*****
8	Personnel movement (ERF to CR or CR to ERF)	Route	Variable	3 min (walking) 1 min (riding)	< 1	*****
9	Manual operations of valves in safeguards building	Route	24 hrs	15 min	3.15	3.40
10	Residual heat removal (RHR) suction valve transfer	Route	24 hrs	15 min	2.90	3.53
11	Electrical connection of permanently installed spare equipment	Route	24 hrs	15 min	0.01	0.18
12	Ventilation fan for service water system	Route	24 hrs	30 min	0.01	*****
13	Obtaining and Analyzing a Post- Accident Effluent Sample	Route	1 hr	10 min (ERM) and 1 hr (lab)	4.4*** 4.3****	NA NA
14	Re-energizing ECCS Valve Circuits	Route	25 min	5 min	0.40	0.59

*** Doses are calculated assuming the sample will be analyzed at the primary chemistry laboratory.

**** Doses are calculated assuming the sample will be analyzed at the emergency response facility.

***** Alternate route not considered because significant dose contribution is due to travel outside the buildings.

Table 5.11.9-1 BVPS Site Boundary Atmospheric Dispersion Factors (sec/m^3)				
Exclusion Area Boundary Averaging Period				
Release Point	**0-2 hr**	–	–	–
BVPS-1 Release Points	1.04E-3	–	–	–
BVPS-2 Release Points	1.25E-3	–	–	–
Low Population Zone Averaging Period				
Release Point	**0-8 hr**	**8-24 hr**	**1-4 day**	**4-30 day**
BVPS-1 and BVPS-2 Release Points	6.04E-5	4.33E-5	2.10E-5	7.44E-6

Table 5.11.9-2A
BVPS-1 On-Site Atmospheric Dispersion Factors (sec/m^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 1 Containment Edge	BVPS-1 CR Intake	7.48E-04	5.77E-04	2.53E-04	2.00E-04	1.78E-04
U 1 Containment Top	BVPS-1 CR Intake	8.16E-04	5.78E-04	2.27E-04	1.71E-04	1.47E-04
U 1 Ventilation Vent	BVPS-1 CR Intake	4.75E-03	3.66E-03	1.43E-03	1.02E-03	8.84E-04
U 1 RWST Vent	BVPS-1 CR Intake	7.34E-04	6.17E-04	2.54E-04	1.96E-04	1.57E-04
U 1 MS Relief Valves	BVPS-1 CR Intake	1.24E-03	9.94E-04	4.08E-04	3.03E-04	2.51E-04
U 1 MSL (break)/AEJ	BVPS-1 CR Intake	1.05E-02	7.72E-03	3.01E-03	2.14E-03	2.00E-03
U 1 Gaseous Waste Storage Vault	BVPS-1 CR Intake	1.40E-03	8.78E-04	3.16E-04	2.93E-04	2.62E-04
U 1 Containment Equipment Hatch	BVPS-1 CR Intake	6.25E-04	4.23E-04	1.76E-04	1.27E-04	1.11E-04
U 1 Cooling Tower	BVPS-1 CR Intake	1.19E-04	8.79E-05	3.41E-05	2.76E-05	2.09E-05
U 1 Containment Edge	BVPS-2 CR Intake	4.88E-04	4.07E-04	1.79E-04	1.41E-04	1.22E-04
U 1 Containment Top	BVPS-2 CR Intake	5.93E-04	4.63E-04	1.84E-04	1.34E-04	1.16E-04
U 1 Ventilation Vent	BVPS-2 CR Intake	2.00E-03	1.62E-03	6.76E-04	5.05E-04	4.06E-04
U 1 RWST Vent	BVPS-2 CR Intake	4.76E-04	4.10E-04	1.70E-04	1.33E-04	1.07E-04
U 1 MS Relief Valves	BVPS-2 CR Intake	7.46E-04	6.31E-04	2.62E-04	1.98E-04	1.62E-04
U 1 MSL (break)/AEJ	BVPS-2 CR Intake	4.24E-03	3.87E-03	1.69E-03	1.18E-03	1.06E-03
U 1 Gaseous Waste Storage Vault	BVPS-2 CR Intake	1.42E-03	8.19E-04	3.38E-04	2.78E-04	2.49E-04
U 1 Containment Equipment Hatch	BVPS-2 CR Intake	4.48E-04	3.33E-04	1.36E-04	1.02E-04	8.70E-05
U 1 Cooling Tower	BVPS-2 CR Intake	1.33E-04	9.49E-05	3.61E-05	2.87E-05	2.25E-05
U 1 Containment Edge	BVPS-2 Aux. Bldg. NW Corner	3.34E-04	2.85E-04	1.23E-04	9.62E-05	8.37E-05
U 1 Containment Top	BVPS-2 Aux. Bldg. NW Corner	4.37E-04	3.41E-04	1.39E-04	1.02E-04	8.79E-05
U 1 RWST Vent	BVPS-2 Aux. Bldg. NW Corner	3.23E-04	2.83E-04	1.18E-04	9.32E-05	7.52E-05
U 1 Cooling Tower	BVPS-2 Aux. Bldg. NW Corner	1.57E-04	1.12E-04	4.13E-05	3.35E-05	2.60E-05
U 1 Containment Edge	BVPS-1 Service Bldg.	1.90E-03	1.57E-03	4.54E-04	5.08E-04	4.55E-04

Table 5.11.9-2A (continued)
BVPS-1 On-Site Atmospheric Dispersion Factors (sec/m^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 1 Containment Top	BVPS-1 Service Bldg.	1.64E-03	8.59E-04	3.35E-04	2.71E-04	2.29E-04
U 1 RWST Vent	BVPS-1 Service Bldg.	2.37E-03	1.88E-03	7.58E-04	5.71E-04	4.48E-04
U 1 Cooling Tower	BVPS-1 Service Bldg.	1.09E-04	8.10E-05	3.28E-05	2.65E-05	1.92E-05
U 1 Containment Edge	ERF Intake	4.53E-05	2.97E-05	1.41E-05	1.23E-05	1.09E-05
U 1 Containment Top	ERF Intake	4.57E-05	3.74E-05	1.50E-05	1.44E-05	1.23E-05
U 1 RWST Vent	ERF Intake	4.53E-05	2.87E-05	1.39E-05	1.21E-05	1.05E-05
U 1 Cooling Tower	ERF Intake	5.75E-05	4.97E-05	2.31E-05	1.80E-05	1.66E-05
U 1 Containment Edge	ERF Edge Closest to Cont.	4.70E-05	3.16E-05	1.54E-05	1.32E-05	1.14E-05
U 1 Containment Top	ERF Edge Closest to Cont.	5.00E-05	3.94E-05	1.62E-05	1.52E-05	1.30E-05
U 1 RWST Vent	ERF Edge Closest to Cont.	4.54E-05	3.14E-05	1.50E-05	1.29E-05	1.13E-05
U 1 Cooling Tower	ERF Edge Closest to Cont.	7.67E-05	6.28E-05	3.10E-05	2.36E-05	2.17E-05

Table 5.11.9-2B
BVPS-2 On-Site Atmospheric Dispersion Factors (sec/m^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 2 Contain. Edge	BVPS-1 CR Intake	3.19E-04	2.38E-04	1.06E-04	8.08E-05	6.19E-05
U 2 Containment Top	BVPS-1 CR Intake	3.83E-04	3.10E-04	1.34E-04	9.83E-05	6.65E-05
U 2 Ventilation Vent	BVPS-1 CR Intake	5.32E-04	3.89E-04	1.75E-04	1.30E-04	9.02E-05
U 2 RWST Vent	BVPS-1 CR Intake	1.70E-04	1.30E-04	5.56E-05	4.40E-05	3.31E-05
U 2 MS Relief Valves	BVPS-1 CR Intake	3.33E-04	2.38E-04	1.09E-04	7.88E-05	5.66E-05
U 2 MSL (break)/AEJ	BVPS-1 CR Intake	6.21E-04	4.87E-04	2.30E-04	1.65E-04	1.10E-04
U 2 Gaseous Waste Storage Vault	BVPS-1 CR Intake	7.71E-04	4.90E-04	2.26E-04	1.76E-04	1.31E-04
U 2 Containment Equipment Hatch	BVPS-1 CR Intake	2.47E-04	1.69E-04	7.94E-05	6.05E-05	4.56E-05
U 2 Contain. Edge	BVPS-2 CR Intake	4.82E-04	3.59E-04	1.55E-04	1.21E-04	9.18E-05
U 2 Containment Top	BVPS-2 CR Intake	5.56E-04	4.45E-04	1.91E-04	1.39E-04	9.35E-05
U 2 Ventilation Vent	BVPS-2 CR Intake	9.39E-04	6.69E-04	3.08E-04	2.23E-04	1.54E-04
U 2 RWST Vent	BVPS-2 CR Intake	2.18E-04	1.58E-04	7.31E-05	5.53E-05	4.12E-05
U 2 MS Relief Valves	BVPS-2 CR Intake	5.01E-04	3.58E-04	1.61E-04	1.19E-04	8.32E-05
U 2 MSL (break)/AEJ	BVPS-2 CR Intake	1.03E-03	7.84E-04	3.57E-04	2.64E-04	1.86E-04
U 2 Gaseous Waste Storage Vault	BVPS-2 CR Intake	1.55E-03	9.04E-04	4.08E-04	3.30E-04	2.45E-04
U 2 Containment Equipment Hatch	BVPS-2 CR Intake	3.45E-04	2.23E-04	1.06E-04	8.29E-05	6.14E-05
U 2 Contain. Edge	BVPS-2 Aux. Bldg. NW Corner	9.12E-04	7.13E-04	3.05E-04	2.35E-04	1.79E-04
U 2 Containment Top	BVPS-2 Aux. Bldg. NW Corner	1.14E-03	8.87E-04	3.83E-04	2.74E-04	1.83E-04
U 2 RWST Vent	BVPS-2 Aux. Bldg. NW Corner	3.19E-04	2.25E-04	1.06E-04	7.95E-05	5.84E-05
U 2 Contain. Edge	BVPS-1 Service Bldg.	1.96E-04	1.54E-04	6.37E-05	5.05E-05	3.89E-05
U 2 Containment Top	BVPS-1 Service Bldg.	2.46E-04	2.07E-04	8.84E-05	6.56E-05	4.49E-05
U 2 RWST Vent	BVPS-1 Service Bldg.	1.24E-04	9.81E-05	4.10E-05	3.24E-05	2.51E-05
U 2 Contain. Edge	ERF Intake	6.02E-05	4.67E-05	2.22E-05	1.78E-05	1.59E-05
U 2 Containment Top	ERF Intake	6.16E-05	5.36E-05	2.42E-05	2.08E-05	1.81E-05

Table 5.11.9-2B (continued)
BVPS-2 On-Site Atmospheric Dispersion Factors (sec/m^3)

Release	Receptor	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
U 2 RWST Vent	ERF Intake	7.28E-05	6.58E-05	3.01E-05	2.31E-05	2.08E-05
U 2 Contain. Edge	ERF Edge Closest to Containment	6.72E-05	5.69E-05	2.65E-05	2.13E-05	1.89E-05
U 2 Containment Top	ERF Edge Closest to Containment	7.22E-05	6.43E-05	2.96E-05	2.48E-05	2.15E-05
U 2 RWST Vent	ERF Edge Closest to Containment	9.42E-05	8.37E-05	3.81E-05	2.97E-05	2.58E-05

Table 5.11.9-3 Analysis Assumptions and Key Parameter Values BVPS Common Control Room	
Control Room Parameters	
Free Volume	173,000 ft^3
Normal Operation Unfiltered Intake	500 cfm
Isolation Mode Unfiltered Inleakage (includes 10 cfm for egress/ingress)	300 cfm
Emergency Mode Intake Rate	600 to 1030 cfm
Emergency Mode Recirculation Rate	N/A
Emergency Mode Intake Filter Efficiency	99% (aerosols) 98% (elemental/organic iodine)
Emergency Mode Recirculation Filter Efficiency	NA
Emergency Mode Unfiltered Inleakage (includes 10 cfm for egress/ingress)	30 cfm
Occupancy Factors	(0-24 hr) 1.0 (1 - 4 days) 0.6 (4-30 days) 0.4
Operator Breathing Rate	(0-30 days) 3.5E-04 m^3/sec
Delay in Initiation of Control Room Emergency Ventilation due to LOOP	
Auto-Start on receipt of CIB (not credited in analysis)	
CR isolated (includes diesel start up/sequencing)	T=77 seconds
CR in emergency pressurization mode	T=137 seconds
Manual	
CR in emergency pressurization mode	T=30 minutes

Table 5.11.9-4A
Analysis Assumptions and Key Parameter Values
Main Steam Line Break[1] – BVPS-1

Core Power Level	2918 MWt
Reactor Coolant Mass (min)	340,711 lbm
Leakrate into Faulted Steam Generator	150 gpd @ STP
Amount of Accident Induced Leakage (AIL) into Faulted SG.	N/A
Maximum time to cool RCS to 212F	19 hrs
~~SG tube leakage reduction fraction after RHR cut-in at T= 8 hrs~~	~~0.44~~
Leakrate into Intact Steam Generators	300 gpd total from 2 SGs @ STP
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine and NG Concentration	Table 5.11.4-1 (0.35 μCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 5.11.4-2 (0.35 μCi/gm DE-I131)
Pre-Accident Iodine Spike Activity	Table 5.11.4-2 (21 μCi/gm DE-I131)
Accident Initiated Spike Appearance Rate	500 times equilibrium appearance rates
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Tech Spec Activity in SG liquid	Table 5.11.4-1(0.1μCi/gm DE-I131)
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (Released without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (Released without holdup)
Fraction of Noble Gas Released from faulted SG	1.0 (Released without holdup)
Minimum Post-Accident Intact SG Liquid Mass	101,799 lbm per SG
Maximum Initial Liquid in each SG	101,799 lbm
Steam Releases from Intact SG	0-2 hr (345,000 lbm) 2-8 hr (734,000 lbm)
Dryout of Faulted SG	Instantaneous
Termination of release from Faulted SG	19 hours
Termination of release from Intact SG	8 hours
Release Point: Faulted SG	Break Point
Release Point: Intact SG	MSSV/ADVs
CR emergency Ventilation: Initiation Signal/Timing Manual	
CR pressurized and in Emergency Mode	T=30 minutes
Control Room Purge (Time/Rate)	24 hours after DBA @16,200 cfm (min) for 30 min

Notes:

(1) Steam generator parameter values reflect the Replacement Steam Generators.

Table 5.11.9-4B
Analysis Assumptions and Key Parameter Values
Main Steam Line Break – BVPS-2

Core Power Level	2918 MWt
Reactor Coolant Mass (min)	341,332 lbm
Leakrate into Faulted Steam Generator	150 gpd @ STP
Amount of Accident Induced Leakage (AIL) into Faulted SG.	2.1 gpm @ STP
Maximum time to cool RCS to 212F	21 hrs
~~SG tube leakage reduction fraction after RHR cut in at T= 8 hrs~~	~~0.44~~
Leakrate to Intact Steam Generators	300 gpd total from 2 SGs @ STP
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 5.11.4-1 (0.35 μCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 5.11.4-2 (0.35 μCi/gm DE-I131)
Pre-Accident Iodine Spike Activity	Table 5.11.4-2 (21 μCi/gm DE-I131)
Accident Initiated Spike Appearance Rate	500 times equilibrium appearance rates
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Tech Spec Activity in SG liquid	Table 5.11.4-1 (0.1 μCi/gm DE-I131)
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (Released without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (Released without holdup)
Fraction of Noble Gas Released from faulted SG	1.0 (Released without holdup)
Minimum Post-Accident Intact SG Liquid Mass	105,076 lbm per SG
Maximum Initial Liquid in each SG	105,076 lbm
Steam Releases from Intact SG	0-2 hr (350,000 lbm) 2-8 hr (730,000 lbm)
Dryout of Faulted SG	Instantaneous
Termination of release from Faulted SG	21 hours
Termination of release from Intact SG	8 hours
Release Point: Faulted SG	Break Point
Release Point: Intact SG	MSSV/ADVs
CR emergency Ventilation: Initiation Signal/Timing Manual	
CR pressurized and in Emergency Mode	T= 30 minutes
Control Room Purge (Time/Rate)	24 hours after DBA @16,200 cfm (min) for 30 min

Table 5.11.9-5A
Analysis Assumptions and Key Parameter Values
Steam Generator Tube Rupture(1) – BVPS-1

Core Power Level	2918 MWt
Reactor Coolant Mass	373,100 lbm
Break Flow to Faulted Steam Generator	0-225 sec (21,900 lbm) 225-1800 sec (128,000 lbm)
Time of Reactor Trip	225 sec
Termination of Release from Faulted SG	1800 seconds
Fraction of Break Flow that Flashes	0-225 sec (0.2227) 225-1800 sec (0.1645)
Leakage Rate to Intact Steam Generators	150 gpd @ STP for each SG
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 5.11.4-1 (0.35 μCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 5.11.4-2 (0.35 μCi/gm DE-I131)
Pre-Accident Iodine Spike Activity	Table 5.11.4-2 (21 μCi/gm DE-I131)
Accident Initiated Spike Appearance Rate	335 times equilibrium
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	91,000 lbm
Faulted SG Liquid Mass (min)	91,000 lbm
Initial SG Liquid Mass per Steam Generators	96,000 lbm
Tech Spec Activity in SG liquid	Table 5.11.4-1 (0.1 μCi/gm DE-I131)
Form of All Iodine Released to the Environment via Steam Generators	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion)	1.0 (Released without holdup)
Fraction of Noble Gas Released from any SG	1.0 (Released without holdup)
Partition Factor in Condenser	100 elemental iodine 1 organic iodine / Noble Gases
Steam Flowrate to Condenser	0-225 sec (1207.407 lbm/sec per SG)
Faulted SG Steam Releases via MSSV/ADVs	225 sec – 1800 sec (68,900 lbm)
Intact SG Steam Releases via MSSV/ADVs	225 sec – 7200 sec (417,100 lbm) 2 hr – 8 hr (979,500 lbm)
Termination of Release from SGs	8 hours
Environmental Release Points	0-225 sec (Condenser Air Ejector) 225 sec –8 hr (MSSVs/ADVs)
CR Emergency Ventilation: Initiation Signal/Timing	
Control room is maintained in normal ventilation mode	
CR Purge Initiation (Manual)Time and Rate	8 hours after DBA @16,200 cfm (min) for 30 min

Notes:

(1) Steam generator parameter values reflect the Replacement Steam Generators.

Table 5.11.9-5B
Analysis Assumptions and Key Parameter Values
Steam Generator Tube Rupture – BVPS-2

Core Power Level	2918 MWt
Reactor Coolant Mass	368,000 lbm
Break Flow to Faulted Steam Generator	(0-116.4 sec) 9,200 lbm (116.4-3932 sec) 190,300 lbm
Time of Reactor Trip	116.4 sec
Termination of Break Flow to Faulted SG	3932 sec
Amount of Break Flow that Flashes	(0-116.4 sec) 1,730.2 lbm (116.4-1968.5 sec) 7496 lbm
Leakage Rate to Intact Steam Generators	150 gpd @ STP for each steam generator
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 5.11.4-1 (0.35 μCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 5.11.4-2 (0.35 μCi/gm DE-I131)
Pre-Accident Iodine Spike Activity	Table 5.11.4-2 (21 μCi/gm DE-I131)
Accident Initiated Spike Appearance Rate	335 times equilibrium
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	95,150 lbm
Faulted SG Liquid Mass (min)	95,150 lbm
Initial Mass in Steam Generators	95,150 lbm
Tech Spec Activity in SG liquid	Table 5.11.4-1 (0.1 μCi/gm DE-I131)
Form of All Iodine Released to the Environment via Steam Generators	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion))	1.0 (Released without holdup)
Fraction of Noble Gas Released from any SG	1.0 (Released without holdup)
Partition Factor in Condenser	100 elemental iodine 1 organic iodine / Noble Gases
Steam Flow to Condenser before Reactor Trip	142,300 lbm (Faulted SG) 281,900 lbm (Intact SGs)
Faulted SG Steam Releases via MSSV/ADVs	(116.4-3932 sec) 74,200 lbm (3932-7200 sec) 0 lbm (7200-28,800 sec) 43,600 lbm
Intact SG Steam Releases via MSSV/ADVs	(116.4-3932 sec) 172,400 lbm (3932-7200 sec) 230,000 lbm (7200-28,800 sec) 775,600 lbm
Termination of Release from SGs	8 hours
Environmental Release Points	(0-116.4 sec) Main Condenser Air Ejector (116.4 sec–8 hr) MSSVs/ADVs
CR Emergency Ventilation: Initiation Signal/Timing	
CR is maintained in normal ventilation mode	
CR Purge Initiation (Manual)Time and Rate	8 hours after DBA @16,200 cfm (min) for 30 min

Table 5.11.9-6
Analysis Assumptions & Key Parameter Values
Locked Rotor Accident(1) – BVPS-1 and BVPS-2

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	340,711 lbm
Primary to Secondary SG tube leakage	450 gpd @ STP
Melted Fuel Percentage	0%
Failed Fuel Percentage	20%
Core Activity of Isotopes in Gap	Table 5.11.4-3
Radial Peaking Factor	1.75
Fraction of Core Inventory in Fuel gap	I-131 (8%) Kr-85 (10%) Other Noble Gases (5%) Alkali Metals (12%)
Iodine Chemical Form in Gap	4.85% elemental 95% CsI 0.15% Organic
Secondary Side Parameters	
Minimum Post-Accident SG Liquid Mass	101,799 lbm per SG
Iodine Species released to Environment	97% elemental; 3% organic
Iodine Partition Coefficient in SGs	100 (all tubes submerged)
Particulate Carry-Over Fraction in SGs	0.0025
Steam Releases from SGs	0-2 hr (348,000 lbm) 2-8 hr (778,000 lbm)
Termination of releases from SGs	8 hours
Fraction of Noble Gas Released	1.0 (Released to Environment without holdup)
Environmental Release Point	MSSVs/ADVs
CR emergency Ventilation: Initiation Signal/Timing CR is maintained under Normal Operation ventilation	

Note:

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

Table 5.11.9-7
Analysis Assumptions & Key Parameter Values
Loss of AC Power(1) – BVPS-1 and BVPS-2

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	340,711 lbm
Primary to Secondary SG tube leakage	450 gpd @ STP
Melted Fuel Percentage	0%
Failed Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 5.11.4-1(0.35 μCi/gm DE-I131)
Secondary Side Parameters	
Minimum Post-Accident SG Liquid Mass	101,799 lbm per SG
Iodine Species released to Environment	97% elemental; 3% organic
Tech Spec Activity in SG liquid	Table 5.11.4-1 (0.1 μCi/gm DE-I131)
Iodine Partition Coefficient in SGs	100 (all tubes submerged)
Fraction of Noble Gas Released from SGs	1.0 (Released without holdup)
Steam Releases from SGs	0-2 hr (348,000 lbm) 2-8 hr (778,000 lbm)
Termination of releases from SGs	8 hours
Environmental Release Point	MSSVs/ADVs
CR emergency Ventilation: Initiation Signal/Timing	
CR is maintained under Normal Operation ventilation	

Note:

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

Table 5.11.9-8 Analysis Assumptions and Key Parameter Values Fuel Handling Accident in Fuel Pool Area or Containment – BVPS-1 and BVPS-2	
Core Power Level	2918 MWt
Number of Rods in Fuel Assemblies	264
Total Number of Fuel Assemblies	157
Number of Damaged Rods	137
Decay Time Prior to Fuel Movement	100 hours
Radial Peaking Factor	1.75
Fraction of Core Inventory in gap	I-131 (8%) Kr-85 (10%) Other Noble Gases (5%) Other Halides (5%) Alkali Metals (12%)
Core Activity of Isotopes in Gap	Table 5.11.4-3
Iodine Form of gap release before scrubbing	99.85% elemental 0.15% Organic
Min depth of water in Fuel Pool or Reactor Cavity	23 ft
Scrubbing Decontamination Factors	Iodine (200) Noble Gas (1) Particulates (∞)
Rate of Release from Fuel	PUFF
Environmental Release Rate (unfiltered) within a 2-hour period	All airborne activity
<u>Environmental Release Points</u>	
Accident in Fuel Pool Area	More Restrictive of Ventilation Vent or SLCRS
Accident in Containment	More Restrictive of Equipment Hatch or SLCRS
CR Emergency Ventilation: Initiation Signal/Timing **BVPS-1 and 2** Control room is maintained in normal ventilation mode. **BVPS-1**	
Control room purge initiation (Manual) Time and Rate	2 hours after DBA @16,200 cfm for 30 min

Table 5.11.9-9
Analysis Assumptions and Key Parameter Values
Small Line Break Outside Containment – BVPS-1 and BVPS-2

Core Power Level	2918 MWt
Minimum Reactor Coolant Mass	340,711 lbm
CVCS letdown line break - mass flow rate	16.79 lbm/s
Break Flow Flash Fraction	37%
Time to isolate break-	15 minutes
Melted Fuel Percentage	0%
Failed Fuel Percentage	0%
RCS Tech Spec NG & Iodine Concentration	Table 5.11.4-1 (0.35 μCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 5.11.4-2 (0.35 μCi/gm DE-I131)
Accident Initiated Spike Appearance Rate	500 times equilibrium
Duration of Accident Initiated Spike	4 hours
Iodine Species released to Environment	97% elemental; 3% organic
SLCRS Filter Efficiency	0%
Environmental Release Point	Ventilation Vent
CR Emergency Ventilation: Initiation Signal/Timing	
CR is maintained under Normal Operation ventilation	

TABLE 5.11.9-10 Analysis Assumptions and Key Parameter Values Waste Gas System Rupture – BVPS-1 and BVPS-2 (Table has been deleted)	

Table 5.11.9-11
Beaver Valley Power Station BVPS-1 and BVPS-2
Exclusion Area Boundary and Low Population Doses (TEDE) (8)

Accident	EAB Dose (rem)(1,3)	LPZ Dose (rem)(2)	Regulatory Limit (rem)
Loss of Coolant Accident	14	2.5	25
Control Rod Ejection Accident(4)	3.1	1.5	6.3
Main Steam Line Break (U1)(7)	0.08 0.11	0.01 0.04	25(PIS) 2.5(CIS)
Main Steam Line Break (U2)(5)(7)	0.4 2.5	0.1 0.7	25(PIS) 2.5(CIS)
Steam Generator Tube Rupture (U1)(7)	2.27 0.93	0.14 0.06	25(PIS) 2.5(CIS)
Steam Generator Tube Rupture (U2)(7)	1.3 0.68	0.07 0.05	25(PIS) 2.5(CIS)
Locked Rotor Accident	2	0.33	2.5
Loss of AC Power	(Note 6)	(Note 6)	2.5
Fuel Handling Accident BVPS-1 BVPS-2	 2.02 2.43	 0.12 0.12	6.3
Small Line Break Outside Containment	0.23	0.012	2.5

Notes:

(1) EAB Doses are based on the worst 2-hour period following the onset of the event.

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

(3) Except as noted, the maximum 2 hr dose for the EAB is based on the 0-2 hr period:

- LOCA: 0.5 to 2.5 hr
- MSLB (CIS): 4 to 6 hr (Unit 2 only)
- LR: 6 to 8 hr

(4) Dose values are based on the containment release scenario. The dose consequences based on the secondary side release scenario are 1 Rem (EAB) and 0.1 Rem (LPZ).

(5) Doses are based on the maximum allowable Accident Induced Leakage (2.1 gpm) 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) The WGSR dose analysis was redone using current licensing basis methodology and criteria, parameters that reflect EPU conditions and ARCON96 methodology. This change meets the criteria for implementation via the 10 CFR 50.59 process. Therefore, the results of this accident re-analysis is not being included for NRC review.

Table 5.11.9-12
30 Day Integrated Control Room Doses (TEDE)

Accident	Control Room Operator	
	Dose (rem)	Reg. Limit (rem)
Loss of Coolant Accident(1) (LOCA)	2 (0.6)	5
Control Rod Ejection Accident(2) (CREA)	1.3	5
Main Steam Line Break (U1)(5) (MSLB)	0.66	5
Main Steam Line Break (U2)(3)(5)	0.6	5
Steam Generator Tube Rupture (U1)(5) (SGTR)	1.95	5
Steam Generator Tube Rupture (U2)(5)	0.32	5
Fuel Handling Accident(6) BVPS-1(5) BVPS-2	 2.36 1.4	5
Locked Rotor Accident(6) (LRA)	2.2	5
Loss of AC Power(6) (LACP)	(Note 4)	5
Small Line Break Outside Containment(6) (SLB)	0.7	5

Notes:

(1) Portion shown in parenthesis for the LOCA represents that portion of the total dose of 2 rem that is the contribution of direct shine from contained sources/external cloud.

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

(3) Dose is based on the maximum allowable Accident Induced Leakage (2.1 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 16, 200 cfm following termination of the environmental releases 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 operations: SGTR, LRA, LACP, SLB outside Containment, WGSR and FHA.

(7) The WGSR dose analysis was re-done using current licensing basis methodology and criteria, parameters that reflect EPU conditions and ARCON 96 methodology. This change meets the criteria for implementation via the 10 CFR 50.59 process. Therefore, the results of this accident re-analysis is not being included for NRC review.