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L-07-042

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

Subject: Beaver Valley Power Station, Unit No. 1
BV-1 Docket No. 50-334, License No. DPR-66
Supplemental Information – BVPS Unit No. 1 Steam Generator Tube Rupture Analysis Update

On November 30, 2006, FirstEnergy Nuclear Operating Company (FENOC) transmitted letter L-06-157 (Reference 1) which included a commitment to provide changes to previously docketed information relative to the Extended Power Uprate (EPU) as a result of the Steam Generator Tube Rupture (SGTR) re-analysis for Beaver Valley Power Station (BVPS) Unit No. 1.

The BVPS Unit No. 1 SGTR re-analysis has been completed and a 10 CFR 50.59 evaluation was performed to address the analytical changes and the Emergency Operating Procedure updates relative to SGTR re-analysis for BVPS Unit No. 1. The evaluation concluded that prior NRC approval was not required to implement the changes. The attachments to this letter are provided to satisfy the commitment described above.

Attachment 1 provides a summary of the updated SGTR analysis results for BVPS Unit No. 1. Attachments 2 through 7 provide updated information reflecting the inclusion of the SGTR re-analysis results into previously docketed information.

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There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Henry L. Hegrat, Supervisor - FENOC Fleet Licensing, at (330) 374-3114.

Sincerely,

James H. Lash

Attachments:

1. Summary Results: Updated Steam Generator Tube Rupture (SGTR) Analysis for Beaver Valley Power Station Unit 1 (BVPS-1)
2. Changes to Sections 5.4 and 5.11 of the Extended Power Uprate (EPU) License Amendment Request (LAR) Licensing Report
3. Changes to FENOC Letter L-05-112
4. Changes to FENOC Letter L-05-137
5. Changes to FENOC Letter L-05-195
6. Changes to FENOC Letter L-06-003
7. Changes to FENOC Letter L-06-157

Reference:

1. FENOC Letter L-06-157, Supplemental Information – SGTR Analysis Update, dated November 30, 2006.

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

Attachment 1 of L-07-042

Summary Results: Updated Steam Generator Tube Rupture (SGTR) Analysis for Beaver Valley Power Station Unit 1 (BVPS-1)

The BVPS-1 Atmospheric Steam Dump Valves (ASDV) [PCV-1MS-101A, B, & C] and Residual Heat Removal Valve [HCV-1MS-104] capacities used in Extended Power Uprate (EPU) analysis for BVPS-1 did not include the effect of the upstream and downstream piping friction losses. Therefore, the actual valve capacities are effectively less when the friction losses are considered. As a result, the BVPS-1 SGTR analysis was updated using the corrected effective valve capacities.

A summary of the results of the BVPS-1 SGTR re-analysis using the corrected valve capacities, which include the piping friction losses, is provided herein.

Summary Results:

The specified function of the affected valves is to open when required to release steam to the environment. This function may be required during normal plant operations for cooldown, or to provide heat removal following accident conditions. This function is required to maintain system integrity and, consequently, minimize offsite doses in the event of a SGTR.

The impact on overfill of the Steam Generator (SG) as well as Reactor Coolant System (RCS) cooldown to Residual Heat Removal (RHR) conditions has been evaluated. The BVPS-1 SGTR analyses update addressed the impact on the following areas:

- Overfill – this analysis was updated and shows that no water relief occurs through the ASDVs or Main Steam Safety Valves (MSSVs). With the reduced valve capacities, the water in the ruptured SG could partially fill the main steam line. The piping was analyzed and remains acceptable under the postulated event. Based on the above, it was concluded that the current dose analysis methodology remains valid since the regulatory based partitioning of radioactive iodine remains applicable within the partially filled steam lines.
- Cooldown – the cooldown analysis using NOTRUMP showed that placing RHR in service and, hence, termination of releases (including the intact SGs) could be accomplished in 24 hours. This time period for placing RHR in service was a change from the original analysis assumption of 8 hours, and this change was evaluated under 10 CFR 50.59. The extension of the cooldown time period from 8 hours to 24 hours was due to the reduced effective valve capacities which lengthened the cooldown period.
- Dose - The dose consequence re-analysis indicated a small increase in the control room dose as a result of the lengthened cooldown period. The evaluation concluded that the increases were minimal and that prior NRC approval was not required to implement the changes. No other dose consequence analyses were affected as a result of the changes to the BVPS-1 SGTR analysis.

During the investigation into quantifying the impact of the lower ASDV capacities for the BVPS-1 SGTR event, the following additional changes were identified and addressed in the Emergency Operating Procedures (EOPs) as described below:

- To locally trip the Turbine-Driven Auxiliary Feedwater (TDAFW) pump should the control room action fail.
- To secure the motor-driven AFW pump(s) from the control room to isolate AFW to the ruptured steam generator in the event the AFW motor-operated isolation valves do not close from the control room.

The BVPS-1 EOP changes have been implemented, and simulator and field validations were conducted to demonstrate that the operator actions can be performed within the time constraints and assumptions of the SGTR analysis.

In conclusion, a 10 CFR 50.59 evaluation was performed to address the analytical changes and the EOP updates relative to the SGTR re-analysis for BVPS-1. The evaluation concluded that prior NRC approval was not required to implement the changes, and the documentation for BVPS-1 EPU has been supplemented to reflect the inclusion of the BVPS-1 SGTR re-analysis results. (See Attachments 2 through 7 for updated EPU information relative to the BVPS-1 SGTR re-analysis.)

Attachment 2 of L-07-042

Changes to Sections 5.4 and 5.11 of the Extended Power Uprate (EPU) License Amendment Request (LAR) Licensing Report

Sections 5.4 and 5.11 of the Extended Power Uprate (EPU) License Amendment Request (LAR) Licensing Report (FENOC letter L-04-125 dated October 4, 2004) provide the analysis for the Steam Generator Tube Rupture (SGTR) event and post-accident dose consequences. The information relative to the BVPS-1 SGTR evaluation has been updated and provided in this attachment. A summary of the affected changes to the applicable sections of the Licensing Report is provided below:

- Section 5.4 "Steam Generator Tube Rupture" and Section 5.4.1 "BVPS-1 Thermal and Hydraulic Analysis for Offsite Radiological Consequences":
  - Changes to the BVPS-1 SGTR licensing basis analysis for EPU conditions to address single failure criterion to support implementation of the BVPS-1 EPU Amendment.
  - Changes to account for the extension of the cooldown time period from 8 hours to 24 hours for a BVPS-1 SGTR due to the reduced valve capacities.

- Section 5.11 "Radiological Assessments – Specifically, Section 5.11.1 "Introduction," Section 5.11.9 "Post-Accident Site Boundary and Control Room Doses," and Section 5.11.10 "Conclusions":
  - Changes to account for the extension of the cooldown time period from 8 hours to 24 hours for a BVPS-1 SGTR due to the reduced valve capacities.

(Ref. FENOC Letter L-04-125, License Amendment Request Nos. 302 and 173, dated October 4, 2004)

NOTE: The changes have been formatted such that deletions are shown with a strike-through and insertions are shown double-underlined for ease of comparison with previously docketed information.

5.4 STEAM GENERATOR TUBE RUPTURE

BVPS-1 and BVPS-2 include analyses for a design basis steam generator tube rupture (SGTR) event to demonstrate that the potential radiological consequences are acceptable. The analysis methodologies for BVPS-1 and BVPS-2 differ as presented in their Updated Final Safety Analysis Reports (UFSARs). The SGTR analysis methodology for BVPS-1 consists of a thermal-hydraulic analysis to provide tube rupture data (e.g., break flow and steam releases) as input to the BVPS-1 radiological consequences analysis. The SGTR analysis methodology for BVPS-2 consists of a two step thermal-hydraulic analysis, i.e., analysis to first demonstrate margin to steam generator overfill followed by analysis to provide tube rupture data (e.g., break flow and steam releases) as input to the BVPS-2 radiological consequences analysis. The SGTR analysis for BVPS-1 is described in Section 5.4.1 and the SGTR analysis for BVPS-2 is described in Sections 5.4.2 and 5.4.3.

Additionally, an operational response analysis of SGTR was performed for BVPS-1 with the same conservative methods described for BVPS-2~~, with the exception that no single failure was modeled~~. This operational response analysis was performed to develop information for operator training. The analysis demonstrated margin to steam generator overfill and confirmed that the BVPS-1 licensing basis analysis methodology provides conservative tube rupture data (e.g., break flow and steam releases) as input to radiological consequences analysis.

The BVPS-1 licensing basis analysis for a SGTR event assumes that release from the ruptured steam generator is terminated in 30 minutes but the analysis methodology ~~does~~ **did** not explicitly address the single failure criterion of 10 CFR 50, Appendix A. The EPU evaluations reflect the existing BVPS-1 licensing basis, i.e., no single failure is modeled. However, the need to address the single failure criterion ~~has~~ **had** been identified and entered into the BVPS Corrective Action Program. FENOC ~~will resolve this issue by either conducting~~ **has conducted** further analyses ~~or by plant modification, such~~ that **addresses** the single failure criterion ~~is met~~ for a BVPS-1 SGTR event. The necessary actions ~~will be~~ **have been** completed ~~prior to, or concurrent with,~~ **to support** implementation of the BVPS-1 EPU amendment.

5.4.1 BVPS-1 Thermal and Hydraulic Analysis for Offsite Radiological Consequences

5.4.1.1 Introduction

In support of the EPU Project for BVPS-1, a steam generator tube rupture (SGTR) thermal-hydraulic analysis for use in the calculation of radiological consequences has been performed. The SGTR analysis supports a T_{avg} window range of 566.2° up to 580°F, secondary-side conditions (e.g., steam pressure, flow, temperature) based on high and low steam generator tube plugging (SGTP) (0% up to 22%). The SGTR analysis also supports the Model 54F replacement steam generators. In order to bound all possible conditions, four separate cases have been analyzed as follows:

1. T_{avg} = 566.2°F and SGTP = 0%
2. T_{avg} = 566.2°F and SGTP = 22%
3. T_{avg} = 580.0°F and SGTP = 0%
4. T_{avg} = 580.0°F and SGTP = 22%

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. The primary thermal-hydraulic parameters which affect the calculation of offsite doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary-to-secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere.

The accident analyzed is the double-ended rupture of a single steam generator tube. It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the reactor coolant system (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low pressurizer pressure. Loss of offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator atmospheric steam dump valves (ASDVs) and/or safety valves. Following SI actuation, it is assumed that the RCS pressure stabilizes at the value where the SI and break flow rates are equal. In the analysis, the equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR. Break flow and the steam releases from the ruptured steam generator are calculated for the initial 30 minute period.

After 30 minutes, it is assumed in the analysis that steam is released only from the intact steam generators in order to dissipate the core decay heat and to subsequently cool the plant down to the residual heat removal (RHR) system operating conditions. It is assumed that plant cooldown to RHR operating conditions is accomplished within ~~8~~ **24** hours after initiation of the SGTR and that steam releases are terminated at that time. A primary-side and secondary-side mass and energy balance is used to calculate the steam release and feedwater flow for the intact steam generators from 0 to 2 hours, ~~and~~ from 2 to 8 hours**, and from 8 to 24 hours**.

5.4.1.2 Input Parameters and Assumptions

The primary-side and secondary-side operating conditions for EPU are documented in Section 2.1.1. A summary of key input assumptions for the SGTR event follows.

High-Head Safety Injection (HHSI) Flow Rates

A larger SI flowrate results in a greater RCS equilibrium pressure and, consequently, higher break flow. Maximum HHSI flowrates were, therefore, assumed for this analysis.

RHR Cut-in Time

The RHR cut-in time based on the RCS heat load and RHR heat removal capacity is conservatively calculated and modeled in the SGTR analysis. This cut-in time affects the duration of long-term steam releases from the intact steam generators to the atmosphere following termination of the break flow. The effect of RHR cut-in time on long-term doses, however, is not significant since the radiation released from the intact steam generators is small relative to that released by the ruptured steam generator. An RHR cut-in time of ~~8~~ **24** hours has been assumed.

Break Flow Flashing Fraction

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured steam generator. Since a transient break flow calculation is not performed for BVPS-1, a detailed time-dependent flashing fraction that incorporates the expected changes in primary-side temperatures cannot be calculated. Instead, a conservative calculation of the flashing fraction is performed using the limiting conditions from the break flow calculation cases. Two time intervals are considered, as in the break flow calculations; pre-reactor trip and post-reactor trip (SI initiation occurs concurrently with reactor trip). Since the RCS and steam generator conditions are different before and after the trip, different flashing fractions would be expected.

The flashing fraction is based on the difference between the primary-side fluid enthalpy and the saturation enthalpy on the secondary side. Therefore, the highest flashing will be predicted for the case with the highest primary-side temperatures. For the flashing fraction calculations, it is conservatively assumed that all of the break flow is at the hot leg temperature (the break is assumed to be on the hot leg side of the steam generator). Similarly, a lower secondary-side pressure maximizes the difference in the primary and secondary enthalpies, resulting in more flashing. The highest pre-trip flashing fraction based on the range of operating conditions covered by this analysis is for the case with a hot leg temperature of 603.9°F, an initial RCS pressure of 2250 psia, and an initial secondary pressure of 623 psia. The case with a hot leg temperature of 617°F would have a lower flashing fraction because the corresponding conservatively high secondary pressure is 831 psia and the flashing is more dependent on secondary pressure than hot leg temperature. All cases consider the same post-trip RCS pressure of 1888.4 psia and post-trip steam generator pressure of 932.75 psia. The highest post-trip flashing fraction, based on the range of operating temperatures covered by this analysis, is for a case with a hot leg temperature of 617°F. It is conservatively assumed that the hot leg temperature is not reduced for the 30 minutes in which break flow is calculated.

Miscellaneous Parameter Assumptions

- Low pressurizer pressure SI actuation setpoint = 1860 psia
- Lowest steam generator safety valve reseat pressure = 932.75 psia, and includes 11.6% main steam safety valve (MSSV) blowdown and 3% safety valve setpoint tolerance.

5.4.1.3 Description of Analyses Performed

A T_{avg} window of 566.2° up to 580.0°F is considered. Section 2.1.1 documents four Performance Capability Working Group (PCWG) cases that have been used for the BVPS-1 SGTR analysis.

Cases are analyzed at a T_{avg} of 566.2° and 580.0°F, with 0% and 22% SGTP. All the cases support a power of 2910 MWt (NSSS power) and thermal design flow (TDF) of 87200 gpm/loop.

Break Flow, Steam Releases, and Feedwater Flows

In total, four cases were considered in the SGTR thermal-hydraulic analysis to bound the EPU operating conditions. Note that these four cases are individually analyzed in order to determine the limiting steam release and limiting break flow between 0 and 30 minutes for the radiological consequences calculation. A single calculation is performed to determine long-term steam releases from, and feedwater flow to, the intact steam generators for the time interval from the start of the event (0 hours) to 2 hours**, from 2 to 8 hours,** and from ~~2~~ **8** hours to RHR cut-in at ~~8~~ **24** hours. The 0 to 2 hour calculations use the 0 to

30 minute intact steam generator steam release and feedwater flow results from the case that resulted in the highest intact steam generator steam and feedwater flow rates.

A mass and energy balance is assumed in the calculation of the break flow and steam releases. The energy balance is based on the following assumed conditions at 30 minutes: (1) the RCS fluid is at the equilibrium pressure and no-load temperature, (2) the pressurizer fluid and steam generator secondary fluid for both the ruptured and intact steam generators is saturated at no-load temperature, and (3) the core and clad, primary system metal, pressurizer metal, and steam generator secondary metal are at no-load temperature. Since the RCS fluid is not at a consistent energy state with the ruptured steam generator and the remainder of the primary and secondary systems, energy must be dissipated to reduce the RCS fluid from equilibrium pressure and no-load temperature to saturation at no-load temperature. It is assumed that the plant is then maintained stable at no-load temperature until 2 hours, and that steam will be released from only the intact steam generators to dissipate the energy from the reduction in the RCS fluid energy state and the core decay heat from 30 minutes to 2 hours.

After 2 hours, it is assumed that plant cooldown to RHR cut-in conditions is initiated by releasing steam from only the intact steam generators. It is assumed that cooldown to RHR cut-in conditions is completed within ~~8~~ 24 hours after the SGTR. After the RHR cut-in conditions are reached, it is assumed that further cooldown is performed using the RHR system and that the steam release from the intact steam generators is terminated.

The energy to be dissipated from 2 to 8 hours is calculated from an energy balance for the primary and secondary systems between no-load conditions at 2 hours and the RHR entry conditions at 8 hours, plus the core decay heat load from 2 to 8 hours. **The energy to be dissipated from 8 to 24 hours assumes only the core decay heat load from 8 to 24 hours.** The amount of steam released from the intact steam generators is calculated from a mass and energy balance for the intact steam generators.

5.4.1.4 Results and Acceptance Criteria

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.

The tube rupture break flow and ruptured steam generator atmospheric steam releases (post-trip) from 0 to 30 minutes for the different SGTR cases are summarized in Table 5.4.1-1. Based on the results of these SGTR cases, bounding values for break flow and steam releases are provided in Table 5.4.1-2 along with flashing fractions, long-term steam releases, and feedwater flows for use in radiological consequences analysis. The maximum break flow and steam releases represent bounding values which are conservative for an offsite dose evaluation. The values in Table 5.4.1-3 include an approximate 10% increase in mass transfer rates for use in a conservative radiological analysis. Increasing the mass transfer data prior to performing the radiological consequences analysis allows future plant changes that result in small increases in the mass transfer rates to be evaluated, without requiring the radiological analysis to be redone.

5.4.1.5 Conclusions

The BVPS-1 SGTR thermal-hydraulic analysis for use in the radiological consequences calculation has been completed in support of the EPU Project. Based on a primary-side and secondary-side mass and energy balance, the break flow and atmospheric steam releases from the ruptured and intact steam generators were calculated for 30 minutes. After 30 minutes, it was assumed that steam is released only from the intact steam generators in order to dissipate the core decay heat and to subsequently cool the plant down to the RHR system operating conditions. For BVPS-1, it was assumed that plant cooldown to RHR operating conditions can be accomplished within ~~8~~ 24 hours after initiation of the SGTR event and that steam releases are terminated at this time. A primary-side and secondary-side mass and energy balance was used to calculate the steam release and feedwater flow for the intact steam generators from 0 to 2 hours, ~~and~~ from 2 to 8 hours, **and from 8 to 24 hours**.

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-1.

Table 5.4.1-1
BVPS-1 Limiting SGTR Thermal-Hydraulic Results *

Tube Rupture Break Flow for 0 to 30 Minutes	
T_{avg} = 566.2°F, 0% SGTP	135,900 lbm
T_{avg} = 566.2°F, 22% SGTP	136,300 lbm
T_{avg} = 580.0°F, 0% SGTP	134,700 lbm
T_{avg} = 580.0°F, 22% SGTP	135,500 lbm
Steam Release from Ruptured SG (Post-Trip) for 0 to 30 Minutes	
T_{avg} = 566.2°F, 0% SGTP	55,800 lbm
T_{avg} = 566.2°F, 22% SGTP	53,100 lbm
T_{avg} = 580.0°F, 0% SGTP	62,600 lbm
T_{avg} = 580.0°F, 22% SGTP	58,600 lbm
* Values rounded up to the nearest 100	

Table 5.4.1-2 BVPS-1 SGTR Thermal-Hydraulic Results for Radiological Dose Analysis	
Reactor Trip, SI Actuation, and Loss Of Offsite Power	224.72 seconds
Pre-Trip (less than 224.72 seconds)	
Tube Rupture Break Flow*	19,900 lbm
Percentage of Break Flow Which Flashes	22.27 %
Steam Release Rate to Condenser	1207.4 lbm/sec for each SG
Post-Trip (after 224.72 seconds)	
Tube Rupture Break Flow for post-trip to 30 minutes*	116,400 lbm
Percentage of Break Flow Which Flashes	16.45 %
Steam Release from Ruptured SG for post-trip to 30 minutes*	62,600 lbm
Steam Release from Intact SGs for post-trip to 2 hours*	379,200 lbm
Feedwater flow to Intact SGs for post-trip to 2 hours*	364,200 lbm
Steam Release from Intact SGs for 2 to 8 hours*	890,500 lbm
Feedwater flow to Intact SGs for 2 to 8 hours*	966,300 lbm
Steam Release from Intact SGs for 8 to 16 hours*	**598,600 lbm**
Feedwater flow to Intact SGs for 8 to 16 hours*	**598,600lbm**
Steam Release from Intact SGs for 16 to 24 hours*	**497,000 lbm**
Feedwater flow to Intact SGs for 16 to 24 hours*	**497,000 lbm**
* Values rounded up to the nearest 100	

Table 5.4.1-3 BVPS-1 SGTR Thermal-Hydraulic Results for Radiological Dose Analysis With Additional 10%	
Reactor Trip, SI Actuation, and Loss Of Offsite Power	224.72 seconds
Pre-Trip (less than 224.72 seconds)	
Tube Rupture Break Flow*	21,900 lbm
Percentage of Break Flow Which Flashes	22.27 %
Steam Release Rate to Condenser	1207.4 lbm/sec for each SG
Post-Trip (after 224.72 seconds)	
Tube Rupture Break Flow for post-trip to 30 minutes*	128,000 lbm
Percentage of Break Flow Which Flashes	16.45 %
Steam Release from Ruptured SG for post-trip to 30 minutes*	68,900 lbm
Steam Release from Intact SGs for post-trip to 2 hours*	417,100 lbm
Feedwater flow to Intact SGs for post-trip to 2 hours*	400,600 lbm
Steam Release from Intact SGs for 2 to 8 hours*	979,500 lbm
Feedwater flow to Intact SGs for 2 to 8 hours*	1,062,900 lbm
Steam Release from Intact SGs for 8 to 16 hours*	**658,400 lbm**
Feedwater flow to Intact SGs for 8 to 16 hours*	**658,400 lbm**
Steam Release from Intact SGs for 16 to 24 hours*	**546,700 lbm**
Feedwater flow to Intact SGs for 16 to 24 hours*	**546,700 lbm**
* Values rounded up to the nearest 100	

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 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.

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**. Except as noted, the control room purge is implemented** after the accident sequence is complete and the environmental release has been terminated. **For the BVPS-1 SGTR, the control room purge is implemented prior to accident termination, but at a time when the remaining environmental releases have minimal effect on the dose consequences. Delaying the control room purge to after the environmental releases are terminated would result in an increase in the operator dose.**

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:

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 analysis 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**. Except as noted, the control room purge is implemented** after the accident sequence is complete and the environmental release has been terminated. **For the BVPS-1 SGTR, the control room purge is implemented prior to accident termination, but at a time when the remaining environmental releases have minimal effect on the dose consequences. Delaying the control room purge to after the environmental releases are terminated would result in an increase in the operator dose.**

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 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~~ **24** 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~~ **24** 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, t~~The control room is purged at T=8 hrs at a rate of 16,200 cfm for a period of 30 mins. **For BVPS-2, the control room purge is implemented upon termination of the environmental releases. For BVPS-1, the control room purge is implemented prior to accident termination, but at a time when the remaining environmental releases have minimal effect on the dose consequences. Delaying the control room purge to after the environmental releases are terminated would result in an increase in the operator dose.**

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.

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**. Except as noted, the control room purge is implemented** after the accident sequence is complete and the environmental release has been terminated. **For the BVPS-1 SGTR, the control room purge is implemented prior to accident termination, but at a time when the remaining environmental releases have minimal effect on the dose consequences. Delaying the control room purge to after the environmental releases are terminated would result in an increase in the operator dose.**

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).

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.

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) **8 hr – 16 hr (658,400 lbm)** **16 hrs – 24 hrs (546,700 lbm)**
Termination of Release from SGs	~~8~~24 hours
Environmental Release Points	0-225 sec (Condenser Air Ejector) 225 sec –~~8~~24 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 5.11.9-12 30 Day Integrated Control Room Doses (TEDE) (7)		
	Control Room Operator	
Accident	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)~~ (8) (SGTR)	~~1.95~~ 1.96	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.

(8) For BVPS-1, the control room purge is implemented at T= 8 hrs, which is prior to accident termination. The remaining environmental releases after T=8 hrs have minimal effect on the dose consequences. Delaying the control room purge to after the environmental releases are terminated (i.e., to T=24hrs) would result in an increase in the operator dose

Attachment 3 to L-07-042

Changes to FENOC Letter L-05-112

Information from FENOC Letter L-05-112, Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173, has been updated and provided in this attachment. A summary of the affected changes to the applicable sections of the Licensing Report is provided below:

- Response to RAI A.1, Table A.1-22 "SGTR Overfill Analysis" has been updated to identify the BVPS-1 EPU values for operator action times for mitigation of the SGTR event.

(Ref. FENOC Letter L-05-112, Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173, dated July 8, 2005.)

NOTE: The changes have been formatted such that deletions are shown with a strike-through and insertions are shown double-underlined for ease of comparison with previously docketed information.

Table A.1-22
Steam Generator Tube Rupture Overfill Analysis
(EPU Licensing Report Section 5.4.2)

Parameter Name	BVPS-1* EPU Value	BVPS-2 EPU Value	BVPS-2 Current Value	Notes
AFW Flow, Total gpm, maximum	930	930	930	Split evenly
Identify and isolate MSIV for ruptured SG, after Rx trip, minutes	16.7	15	10.75 from the beginning of the event	
Initiate RCS cooldown by local operation of Residual Heat Removal Valve (RHRV) and/or ASDV, after MSIV closure, minutes	~~10~~ 15	7	Not used	Single failure case, different failure combinations analyzed
Initiate RCS cooldown by operation of two ASDVs from MCR, after MSIV closure, minutes	2.4	2	Operation of ASDVs in 9 minutes	No failure case, different failure combinations analyzed
Initiate RCS depressurization, after cooldown, minutes	~~4.9~~ 3	4	2.5	
Initiate SI Termination, after depressurization, minutes	~~3~~ 4.9	3	1.25	
Isolate/control AFW to ruptured SG, after Rx trip, minutes	6.8	5.5	10.75 from the beginning of the event	
Main Feedwater Temperature HFP, °F, minimum	400	400	437.5	
Pzr PORV Capacity, per valve, lb/hr, nominal	~~210,000~~ 194,000 @ ~~2500~~ 2365 psia	232,000 @ 2350 psia	210,000	
Pzr Pressure Uncertainty, psi, maximum	+/- 40	+/- 45	30	
RCS Pressure, psia, nominal	2250	2250	2250	
RCS Vessel Average (T_{avg}) Temp, HFP, Low T_{avg} Case, °F, nominal	566.2	566.2	576.2	
SG Tube Plugging,%, maximum	22	22	30	
SG Water Level, Greater than 20% Power, % narrow range, nominal	65 (U1 RSG)	44	44	
SI Act – Pressurizer Pressure, psig, nominal	~~1845~~ 1841	1856	1856	
SI Flow Rate vs. Pressure (Max Safeguards), gpm vs. psig, maximum	See Table ~~A.1-22B~~ A.1-22C	See Table A.1-22B	See Table A.1-22A	
SI Full Flow Delay (without offsite power), seconds, nominal	~~10~~ 20	10	0	

* Unit 1 EPU information presented is the LOFTTR2 operational response analysis. There is no current (pre-EPU) Unit 1 LOFTTR2 operational response analysis.

New Table A.1-22C being added to reflect updated BV-1 SGTR Overfill Analysis

Table A.1-22C
BVPS-1: EPU Total Injected Flow into Core vs. RCS Backpressure For SGTR Overfill
(EPU Licensing Report Section 5.4.2)

RCS Pressure (psig)	Injected Flow (lbm/sec)
0	105.8
100	103.6
200	101.4
400	96.8
600	92.0
800	87.0
1000	81.8
1200	76.3
1400	70.3
1600	63.9
1800	56.9
2000	49.3
2200	40.7
2400	30.1

Attachment 4 to L-07-042

Changes to FENOC Letter L-05-137

Information from FENOC Letter L-05-137, Response to a Request for Additional Information (RAI dated July 28, 2005) in Support of License Amendment Request No. 320, has been updated and provided in this attachment. A summary of the affected changes to the applicable sections of the Licensing Report is provided below:

- Response to RAI B.3 has been updated to identify the extension of the cooldown time period from 8 hours to 24 hours due to the reduced valve capacities.

(Ref. FENOC Letter L-05-137, Response to a Request for Additional Information (RAI dated July 28, 2005) in Support of License Amendment Request No. 320, dated August 26, 2005.)

NOTE: The changes have been formatted such that deletions are shown with a strike-through and insertions are shown double-underlined for ease of comparison with previously docketed information.

B.2 Question

In your July 8, 2005, EPU RAI response to Item X.3, Section 5.4 of Enclosure 1, why wasn't failure of the atmospheric dump valve (ADV) considered at accident initiation? Would it be more limiting than assuming failure of the ADV at accident initiation when the SG with the tube rupture is isolated?

Response:

The subject RAI response stated that the limiting single failure in the supplemental BVPS-1 SGTR operational response analysis case for radiological dose analysis is a failure of the ADV to the open position on the ruptured steam generator at the time that the ruptured steam generator is isolated.

Although a failure of an ADV on the ruptured steam generator at accident initiation may produce greater mass release, it would not produce more conservative radiological dose consequences. The ADV is failed at the time of steam generator isolation in order to maximize the radiological dose consequences. The SGTR dose analysis methodology provides for a scenario that maximizes the radiological dose consequences of the SGTR event. As outlined in Section 5.11.9.8 of Enclosure 2 of the RSG LAR, the dose model assumes that the noble gases in the break flow and the iodine in the portion of the break flow that flashes is released instantaneously without holdup. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released in proportion to the steaming rate and partition factor. Delaying the failure of the ADV on the ruptured steam generator until the time of ruptured steam generator isolation allows for an increase in the activity in the steam generator liquid due to the accumulation of the non-flashed iodine activity in the ruptured steam generator. Likewise, the delay allows for a buildup of activity in the RCS due to the accident initiated iodine spike. In summary, the break flow that does not flash will continue to build up the activity in the steam generator liquid and consequently the mass released at this later time in the transient contains a higher concentration of activity than a comparable release early on in the transient, providing for a conservative radiological release.

The supplemental BVPS-1 operational response analysis case for radiological dose analysis was performed to confirm that the BVPS-1 licensing basis analysis (mass and energy balance calculation) is conservative with respect to radiological dose consequences.

B.3 Question

With regard to enclosure 3 of the April 13, 2005, LAR:

Is it assumed or has it been verified that plant cooldown and steam releases from the intact SG cease at 8 hours following the SGTR initiating event?

Response:

As described in Section 5.4.1 of Enclosure 2 of the April 13, 2005, RSG LAR, the BVPS-1 licensing basis analysis for the SGTR event is a mass and energy balance calculation. This calculation includes assumptions that: 1) following the termination of primary-to-secondary break flow and steam release from the ruptured steam generator, the plant is stabilized at no-load temperature with steam release from the intact steam generators until 2 hours after initiation of the SGTR, and 2) the plant is then cooled down with steam release from the intact steam generators to RHR entry conditions within ~~8~~ **24** hours after initiation of the SGTR, at which time the steam release from the intact steam generators is terminated.

The supplemental BVPS-1 SGTR operational response analysis uses the LOFTTR2 computer code to model the plant response to the SGTR event including the simulation of the operator actions for recovery from a SGTR based on the BVPS-1 Emergency Operating Procedures (EOPs), which are based on the Westinghouse Owners Group Emergency Response Guidelines. The LOFTTR2 analysis is performed for the time period from initiation of the SGTR until the primary and secondary pressures are equalized, at which time primary-to-secondary break flow is terminated. The BVPS-1 SGTR operational response analysis does not model plant response during the stabilization period that is assumed to last up to 2 hours after initiation of the SGTR or the cooldown to RHR entry conditions period that is assumed to last up to ~~8~~ **24** hours after initiation of the SGTR.

Following break flow termination, the plant operators will stabilize and then cool down the plant to RHR entry conditions consistent with the BVPS-1 EOPs. The EOPs direct the operators to use the condenser steam dump valves for steam release to stabilize and cool down the plant. If the condenser steam dump valves are not available, the EOPs direct the operators to use the atmospheric steam dump (ASD) valves. Both the condenser steam dump valves and the ASD valves have adequate capacity to support the cooldown to RHR entry conditions within ~~8~~ **24** hours after the SGTR.

The BVPS-1 plant configuration includes three ASD valves (one for each steam generator) and one residual heat release control valve, which is common for all steam generators. These valves have a total atmospheric steam dump capacity that is sufficient to cool down the plant from no-load temperature of 547°F to RHR entry temperature of 350°F in 4 hours (i.e., 50°F/hour cooldown). The ASD valves on the intact steam generators, which are sized to support a normal cooldown at 50°F/hour, have sufficient capacity to cool down the plant to the RHR entry temperature within ~~8~~ **24** hours after the SGTR event, at which time RHR operation can be initiated and steam release from the intact steam generators can be terminated. Thus, it has been confirmed that the ASD valves on the intact steam generators have sufficient capacity to cool down the plant to RHR entry conditions within ~~8~~ **24** hours following the SGTR.

Part C - Section 5.11 Radiological Analysis

C.1 Question

What is the basis for assuming a 30-minute purge of the control room envelope (CRE) following the completion of the accident sequence and are there procedures directing the operators to take such actions (Pg 5-225)?

Response:

The 30 minute purge of the CRE following completion of the accident sequence **for the MSLB, and completion of the majority of the environmental release following a SGTR** is utilized in the dose consequence analyses to **reduce operator exposure and** support compliance with the regulatory dose criteria of 10 CFR 50.67 and Regulatory Guide 1.183 relative to maximum allowable post-accident exposure of the control room operator.

Attachment 5 to L-07-042

Changes to FENOC Letter L-05-195

Information from FENOC Letter L-05-195, Additional Information Regarding Responses to RAI Dated July 28, 2005 in Support of License Amendment Request No. 320, has been updated and provided in this attachment. A summary of the affected changes to the applicable sections of the Licensing Report is provided below:

- Response to RAI question B.1 (Table B.1) and RAI questions 5 and 6 were updated to identify the termination of the primary-to-secondary break flow for EPU conditions and the extension of the time period from 8 hours to 24 hours due to the reduced valve capacities.
- Changes were made to Table 5-2 "BVPS-1 Margin to Overfill Analysis Sequence of Events" to account for the SGTR re-analysis for BVPS-1.
- Figures 5-1 through 5-6 were revised to reflect the SGTR re-analysis for BVPS-1.

(Ref. FENOC Letter L-05-195, Additional Information Regarding Responses to RAI Dated July 28, 2005 in Support of License Amendment Request No. 320, dated December 6, 2005.)

NOTE: The changes have been formatted such that deletions are shown with a strike-through and insertions are shown double-underlined for ease of comparison with previously docketed information.

UPDATED TABLE B.1 FROM RESPONSE TO RAI QUESTION B.1
(FROM RAI RESPONSE L-05-195 DATED DECEMBER 6, 2005)

Table B.1

Analysis Assumptions and Key Parameter Values
BVPS-1 Steam Generator Tube Rupture(1) – Operational Response Case

Core Power Level Reactor Coolant Mass Break Flow to Faulted Steam Generator (SG) Time of Reactor Trip Amount of Break Flow that Flashes Leakage Rate to Intact SG's Failed/Melted Fuel Percentage RCS Tech Spec Iodine & NG Concentration RCS Equilibrium Iodine Appearance Rates Pre-Accident Iodine Spike Activity Accident Initiated Spike Appearance Rate Duration of Accident Initiated Spike	2918 MWt 373,100 lbm 0-120 sec (9,500 lbm) 120-***3988*** sec (***196,500*** lbm) 120 sec 0-120 sec (1810 lbm) 120-***2142.5*** sec (***8635.3*** lbm) 150 gpd @ $63^{\circ}F$ and 1 ATM for each SG 0% Table 5.11.4-1 (0.35 µCi/gm DE-I131) Table 5.11.4-2 (0.35 µCi/gm DE-I131) Table 5.11.4-2 (21 µCi/gm DE-I131) 335 times equilibrium 4 hours
Secondary System Release Parameters: Intact SG Liquid Mass (min) Faulted SG Liquid Mass (min) Initial SG Liquid Mass per SG's Tech Spec Activity in SG liquid Form of All Iodine Released to the Environment via SG's Iodine Partition Coefficient (unflashed portion) Fraction of Iodine Released (flashed portion) Fraction of Noble Gas Released from any SG Partition Factor in Condenser Steam Flowrate to Condenser Faulted SG Steam Releases via MSSV/ADVs Intact SG Steam Releases via MSSV/ADVs Termination of Release from SGs Environmental Release Points	 91,953 lbm 91,953 lbm 91,953 lbm Table 5.11.4-1 (0.1 µCi/gm DE-I131) 97% elemental; 3% organic 100 (all tubes submerged) 1.0 (Released without holdup) 1.0 (Released without holdup) 100 elemental iodine 1 organic iodine/Noble Gases 0-120 sec (1202 lbm/sec from faulted SG) 0-120 sec (1188 lbm/sec per intact SG) 120 sec – ***3988*** sec (***89,500*** lbm) 2 hr – 8 hr (***42,600*** lbm(2)) 120 sec ***– 3988 sec (197,400 lbm)*** ***3988 sec – 7200 sec (228,900 lbm)*** 2 hr – 8 hr (***768,700*** lbm) **8 hr – 16 hr (658,400 lbm)** **16 hrs – 24 hrs (546,700 lbm)** ~~8~~ 24 hours 0-120 sec (Condenser Air Ejector) 120 sec –~~8~~ 24 hr (MSSVs/ADVs)
CR Emergency Ventilation: Initiation Signal/Timing Control Room (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

Notes:

(1) Steam generator parameter values reflect the Replacement Steam Generators and Operations Assessment

(2) Brief depressurization release in preparation of shutdown cooling

Additional information is being provided to update responses to RAI Questions 5 and 6 relative to the steam generator tube rupture event for BVPS Unit No. 1.

5. Original Question:

Section 5.4 of the April 13, 2005, RSG LAR states that an operational response analysis of steam generator tube rupture (SGTR) was performed for BVPS-1. The NRC staff requests that the licensee provide a table listing the sequence of events and times from break initiation to event termination that shows operators can terminate the break flow from the ruptured SG within 51 minutes of accident initiation for the SG replacement and extended power uprate (EPU) conditions. Additionally, the staff requests that the licensee provide the results of the SGTR thermal-hydraulic analysis over time showing the pressurizer pressure, intact and ruptured SGs pressures, and ruptured SG water volume for the analysis to demonstrate that no overfilling of the SG occurs.

Updated Response (with changes from previous response provided in bold type): Note, there were no changes to Table 5-1, which is being included for completeness.

The SGTR operational response analysis for EPU conditions with RSG included cases to provide thermal-hydraulic tube rupture data for use in radiological dose consequence analysis and for evaluation of margin to overfill. The sequence of events for these cases is shown in Tables 5-1 and 5-2, respectively. The sequence of events tables show that termination of the event (i.e., termination of primary-to-secondary break flow) occurs at 3988 seconds (approximately 66.5 minutes) and ~~3578~~ **3976** seconds (approximately ~~60~~ **66** minutes), respectively, for EPU conditions with RSGs. For BVPS-1, 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 5-1 through 5-6 show the response of pressurizer level, pressurizer pressure, intact and ruptured SG pressure, intact loop RCS temperatures, primary-to-secondary break flow, and ruptured SG water volume for the margin to overfill case. The response to Question 6 provides information regarding the analysis performed in the 1990's for current power conditions that established a break flow termination time of 51 minutes.

Table 5-1
BVPS-1 Thermal-Hydraulic Analysis Sequence of Events

Event (Thermal & Hydraulics for Doses)	Time (seconds)
Steam generator tube rupture	0
Reactor trip – Overtemperature Delta-T	120
Safety injection initiated	148
Isolate auxiliary feedwater to ruptured steam generator	573
Ruptured steam generator steamline isolated	1122
Ruptured steam generator atmospheric dump valve (ADV) fails open	1124
Ruptured steam generator ADV block valve closed *	1724
Reactor coolant system cooldown initiated	1868
Reactor coolant system cooldown terminated	2934
Reactor coolant system depressurization initiated	3116
Reactor coolant system depressurization terminated	3208
Safety injection terminated	3502
Steam relief to maintain sub-cooling	3570
Break flow terminated	3988

* NOTE: Operator action time to isolate the failed-open ADV on the ruptured steam generator was increased to allow for additional operator action time margin.

Table 5-2
BVPS-1 Margin to Overfill Analysis Sequence of Events

Event (Margin to Overfill)	Time (seconds)
Steam generator tube rupture	0
Reactor trip – Overtemperature Delta-T	~~106~~ 124.5
Auxiliary feedwater initiated	~~121~~ 154.5
Safety injection initiated	~~197~~ 261.5
Isolate auxiliary feedwater to ruptured steam generator	~~515~~ 533
Ruptured steam generator steamline isolated	~~1110~~ 1128
Reactor coolant system cooldown initiated	~~1254~~ 1272
Reactor coolant system cooldown terminated	~~2466~~ 2728
Reactor coolant system depressurization initiated	~~2646~~ 2908
Reactor coolant system depressurization terminated	~~2752~~ 3014
Steam relief to maintain subcooling	~~2952~~ 3300
Safety injection terminated	~~3046~~ 3308
Break flow terminated	~~3578~~ 3976
	Volume (ft^3)
Available ruptured steam generator secondary volume*	~~5630~~ 5885
Maximum ruptured steam generator secondary volume	~~5602~~ 5850
Available ruptured steam generator secondary volume to overfill	~~28~~ 35

***NOTE: Includes the steam generator volume of 5630 ft^3 and the steamline volume of 255 ft^3**

Figure 5-1
Pressurizer Level

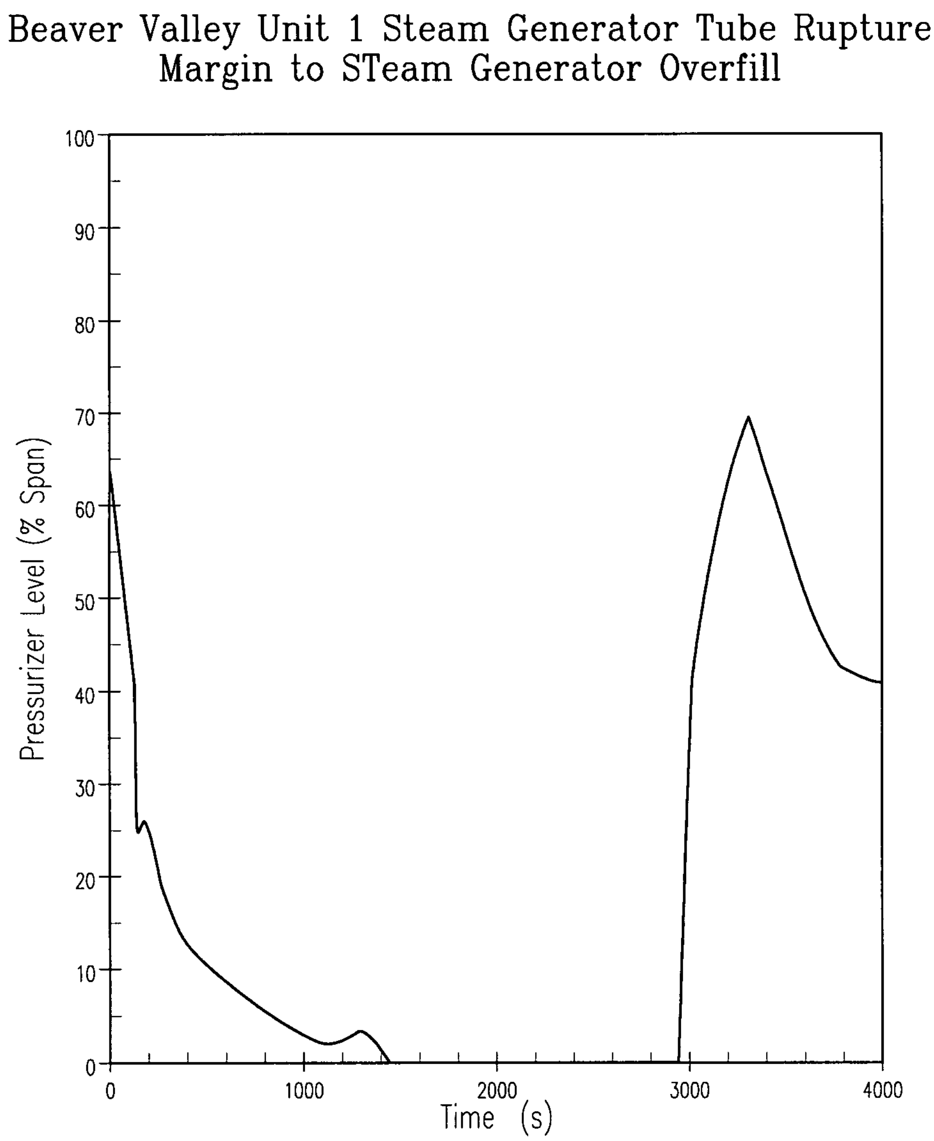


Figure 5-2
Pressurizer Pressure

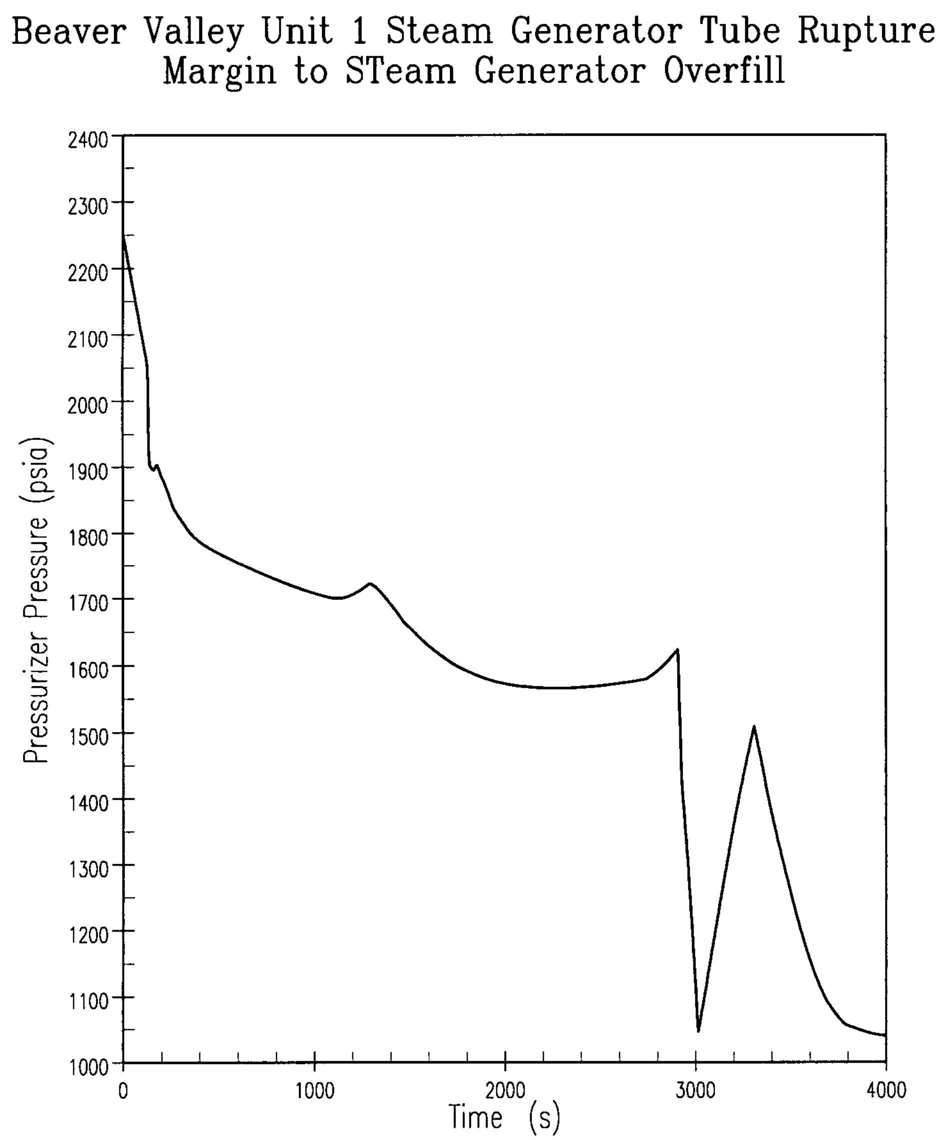


Figure 5-3
Secondary Pressure

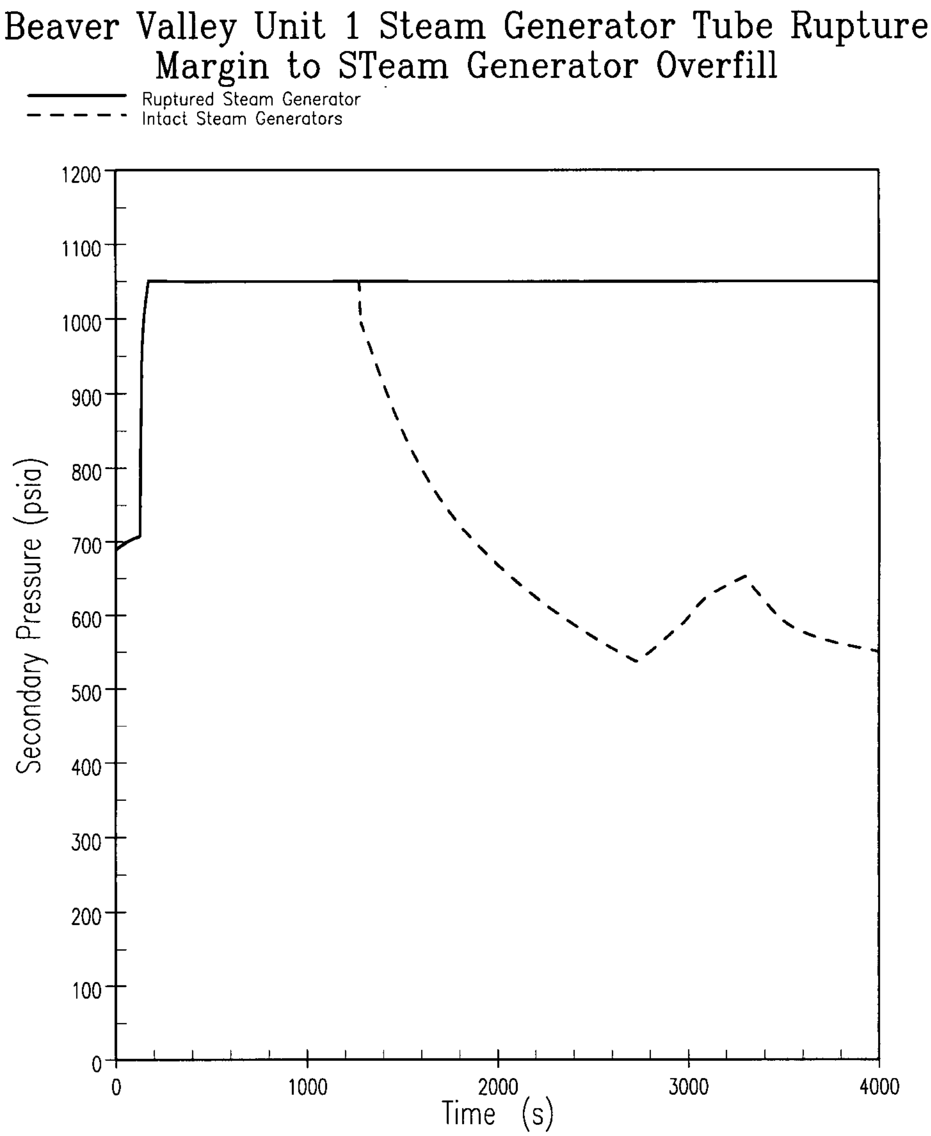


Figure 5-4
Intact Loop RCS Temperatures

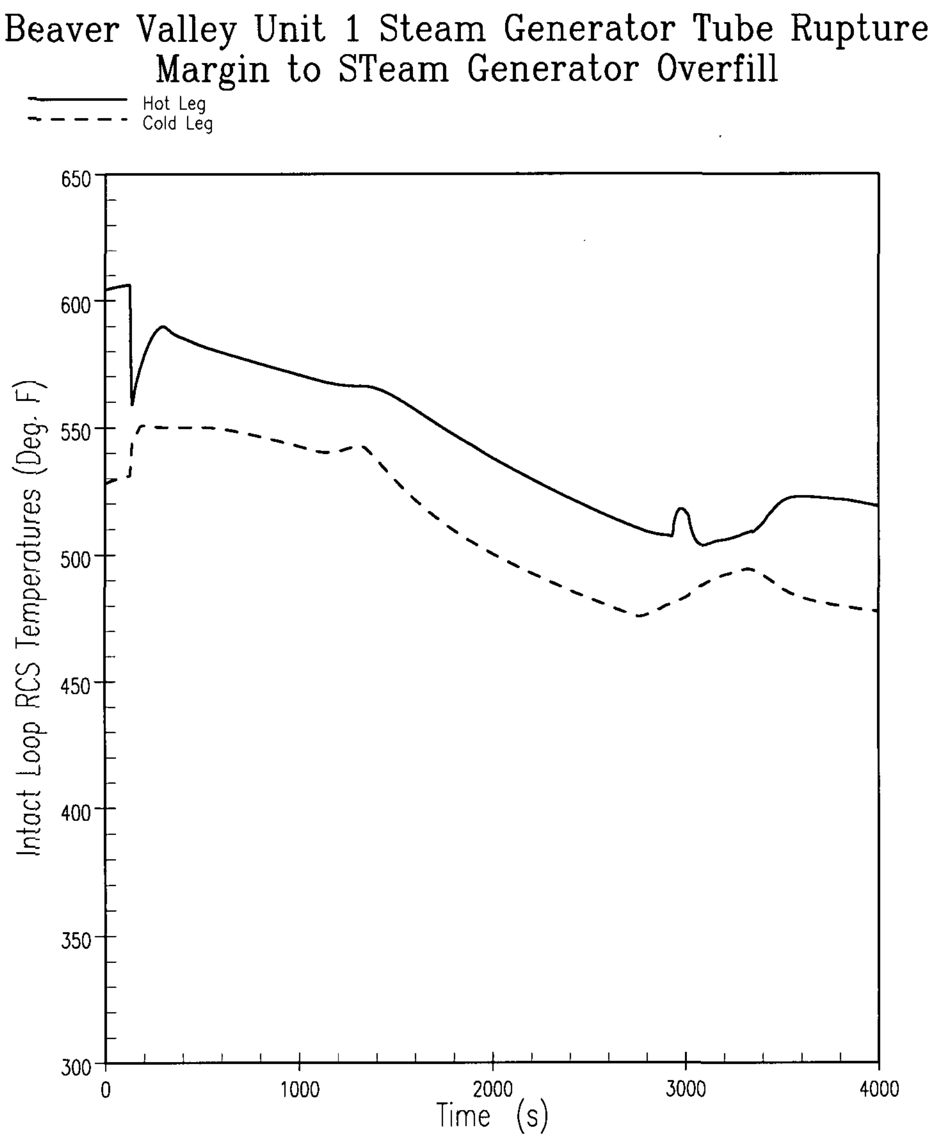


Figure 5-5
Primary-to-Secondary Break Flow

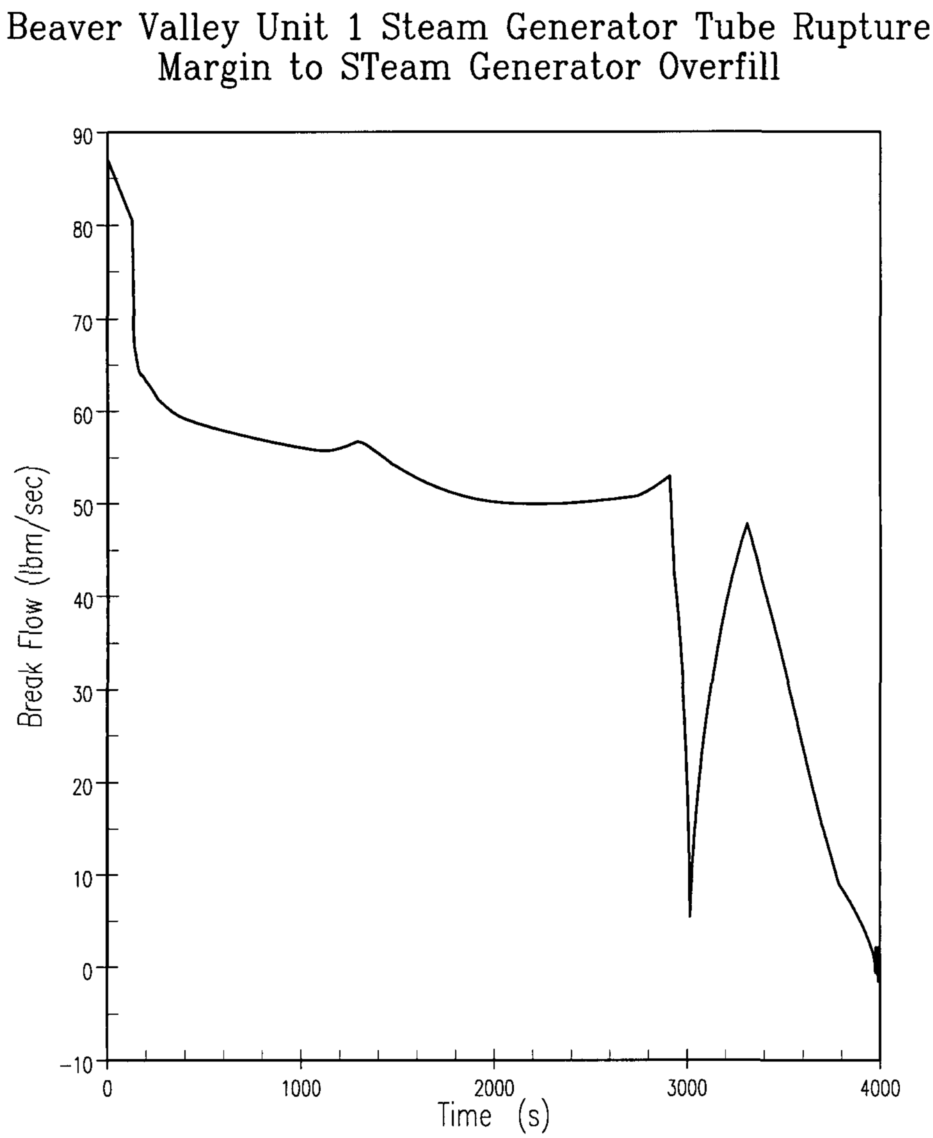
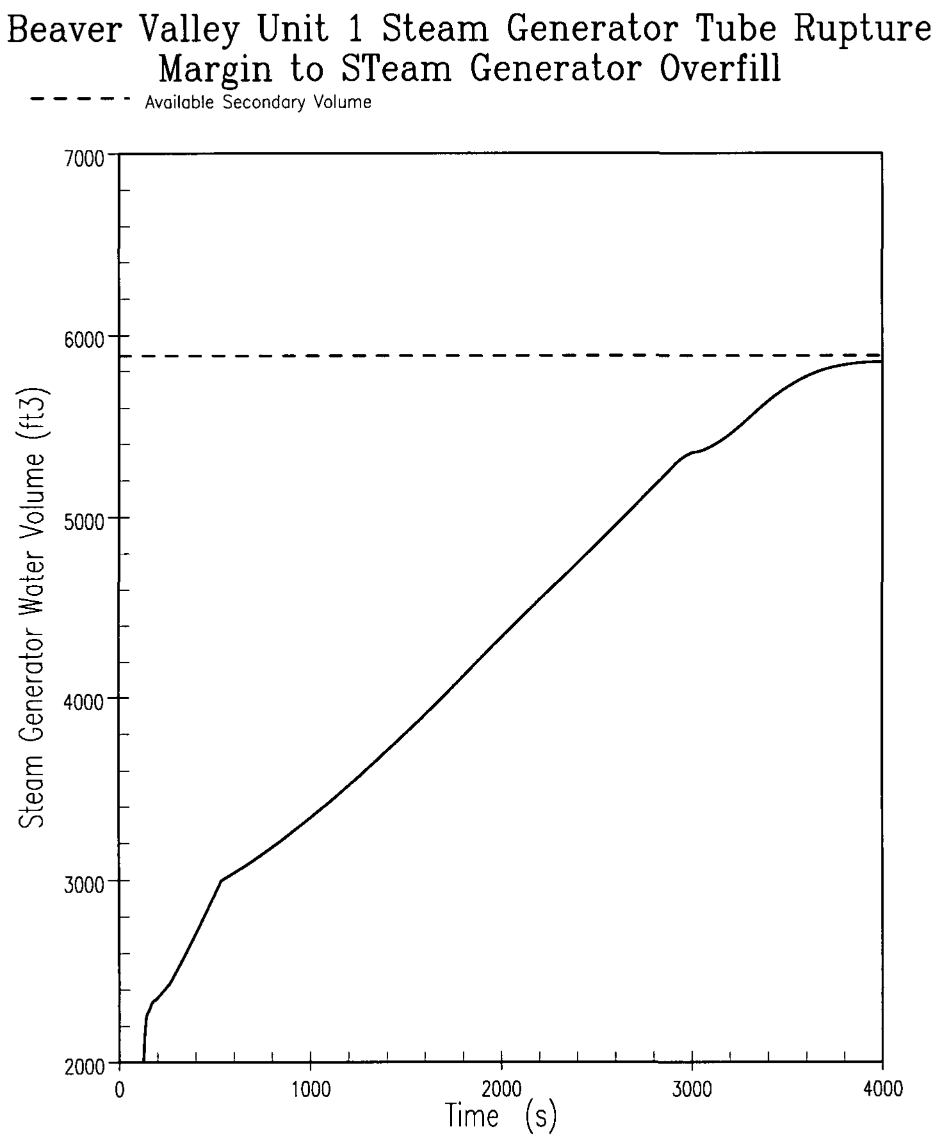


Figure 5-6
Ruptured Steam Generator Water Volume



6. Original Question:

The NRC staff requests that the licensee provide the technical justification as to why a transient break flow analysis was not performed for the SGTR and why the 30-minute release assumption is conservative for the analysis when termination of the event exceeds 30 minutes.

Updated Response *(with changes from previous response provided in bold type):*

The BVPS-1 SGTR licensing basis analysis methodology (Section 5.4 of Enclosure 2 of FENOC Letter L-05-069) consists of a thermal-hydraulic analysis to provide tube rupture data (e.g., break flow and steam releases) as input to the BVPS-1 SGTR radiological dose consequence analysis. This licensing basis methodology includes an assumption that the break flow and steam release from the ruptured steam generator are terminated at 30 minutes.

As described in the response to NRC RAI X.1 of Enclosure 2 of the July 8, 2005 (L-05-112) Response to RAIs on the EPU LAR, a condition report was written in the 1990's that documented that more than 30 minutes was required to terminate radioactive steam release from the ruptured steam generator. At that time through the corrective action process, the break flow termination time was revised to 51 minutes. Even though the break flow termination time increased, it was determined that the primary-to-secondary break flow based on the assumptions that terminate break flow in 30 minutes actually resulted in a higher primary-to-secondary break flow than the case that terminated break flow in 51 minutes. Consequently, it was concluded that the licensing basis methodology including the assumption of break flow termination at 30 minutes was conservative with respect to a transient break flow analysis that included a longer break flow termination time.

This approach established in the 1990's was retained for the BVPS-1 SGTR analysis for EPU conditions with the RSGs. A SGTR licensing basis methodology analysis was performed including the 30-minute isolation time to provide conservative break flow and steam release data as input to the SGTR radiological dose consequence analysis. The results of this SGTR licensing basis methodology analysis are presented in Section 5.4.1 of Enclosure 2 of the RSG LAR.

To develop operator action information for operator training as well as to confirm that the licensing basis methodology analysis continues to provide conservative estimates for the radiological dose consequences following a SGTR, a SGTR operational response (transient break flow) analysis was performed for BVPS-1 at EPU conditions with the RSGs. As noted in Section 5.4 of Enclosure 2 of the April 13, 2005 (L-05-069) RSG LAR, this operational response analysis demonstrates that the BVPS-1 SGTR licensing basis analysis methodology is conservative.

The SGTR radiological dose consequence analysis using the tube rupture data (e.g., break flow and steam releases) from the BVPS-1 SGTR operational response (transient break flow) analysis was provided in response to NRC RAI B.1 of Attachment A of the August 26, 2005 (L-05-137) Response to RAIs on the RSG LAR. This SGTR radiological dose consequence analysis demonstrates that the SGTR licensing basis methodology analysis with a break flow termination time of 30 minutes is more limiting than the operational response analysis with a break flow termination time of 3988 seconds (approximately 66.5 minutes), which is the break flow termination time calculated for the SGTR operational response analysis radiological dose consequence case at EPU conditions with RSGs.

The margin to overfill analysis from the operational response (transient break flow) analysis is provided in the response to Question 5. This margin to overfill analysis demonstrates that the primary-to-secondary break flow into the ruptured steam generator is terminated prior to overfilling the ruptured steam generator. The operational response analysis for the margin to overfill case at EPU conditions with RSGs shows a break flow termination time of ~~3578~~ **3976** seconds (approximately ~~60~~ **66** minutes). The operator action times associated with this analysis have been provided as input to operator training. Additional information pertaining to operator actions and operator training has been provided in Enclosure 3 of the October 7, 2005 (L-05-154) EPU LAR Supplemental Information.

This information along with the more detailed information in the referenced RAIs provide the technical justification as to why the 30-minute SGTR licensing basis methodology analysis is retained for the BVPS-1 SGTR and why the 30-minute release assumption is conservative for the radiological dose consequence analysis when termination of the event exceeds 30 minutes. The technical justification includes a SGTR operational response (transient break flow) analysis as described in this response.

Note that the operator response times for the BVPS-1 SGTR operational response analysis have been validated as part of the emergency operating procedure (EOP) simulator validation process. One operator action time included in the SGTR operational response analysis case to provide thermal-hydraulic tube rupture data for use in radiological dose consequence analysis is the "local isolation of a failed-open atmospheric dump valve on the ruptured SG within 6.5 minutes after the valve fails open." The atmospheric dump valve (ADV) is assumed to fail open when the main steam line isolation valve for the ruptured SG is closed. This operator action time has been validated and is reflected in the BVPS Corrective Action Program. In order to afford additional operator response time for this local operator action, additional analysis has been performed to allow up to a 10-minute operator action time. The analysis results for 10-minute operator action time still support the conclusion that the BVPS-1 SGTR licensing basis methodology with a break flow termination time of 30 minutes is more limiting with respect to radiological dose consequences than the operational response analysis. The results of the revised SGTR radiological dose consequence analysis and the validation of the 10-minute operator response time were reviewed at the November 29, 2005 NRC audit of the EPU/RSG radiological dose consequence analyses. This change to the operator action time to locally isolate a failed open ADV on the ruptured SG for the SGTR operational response analysis dose case does not impact the SGTR operational response analysis margin to overfill case.

Attachment 6 to L-07-042

Changes to FENOC Letter L-06-003

Information from FENOC Letter L-06-003, Additional Information in Support of License Amendment Request Nos. 302 and 173, has been updated and provided in this attachment. A summary of the affected changes to the applicable sections of the Licensing Report is provided below:

- Enclosure 2, Table 2-1 "Comparison of BVPS-1 Operator Action Times in EPU UFSAR Safety Analysis" has been updated to include the operator action times used in the SGTR re-analysis for initiation of cooldown, inside and outside the control room.

(Ref. FENOC Letter L-06-003, Additional Information in Support of License Amendment Request Nos. 302 and 173, dated January 25, 2006.)

NOTE: The changes have been formatted such that deletions are shown with a strike-through and insertions are shown double-underlined for ease of comparison with previously docketed information.

Table 2-1 (Continued)
Comparison of BVPS-1 Operator Action Times - EPU UFSAR Safety Analysis

UFSAR Safety Analysis	Operator Action	Operator Action Time Used in Current Power Analysis	Operator Action Time Used in EPU Analysis	Action Time (from EOP Validation) [Method]	Action Time Completed in Time Available (YES/NO)
Steam Generator Tube Rupture (SGTR)					
SGTR Overfill Analysis [See Note 1]	1. Isolate auxiliary feedwater flow to the ruptured SG	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reactor trip: - 6.8 minutes **from inside the main control room.** - **14 minutes (*see note below)**	4.1 minutes [Simulator] **[Simulator]**	YES **YES**
	2. Isolate steam flow (close MSIV) from the ruptured SG	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reactor trip: 16.7 minutes	8.1 minutes [Simulator]	YES
	3. Initiate cooldown from the intact SGs via the main steam system after MSIV closure	There is no current power analysis for actions inside the main control room or LOFTTR2 SG overfill operational analysis	Within the following times after the MSIV is closed: 1. For actions from inside the main control room: 2.4 minutes [See note 3] Cooldown initiation within 19.1 minutes of the Reactor trip.	[See Note 3] Cooldown initiation was validated at 11.7 minutes of the Reactor trip. [Simulator]	YES
			2. For actions from outside the main control room: ~~10~~ 15 minutes	6 minutes [Walkthrough]	YES
	4. Initiate RCS depressuriztion (open pressurizer PORV) after completion of the cooldown	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reaching the end of cooldown target temperature: 3.0 minutes	58 seconds [Simulator]	YES
	5. Terminate SI (isolate the high head safety injection flow path) after completion of RCS depressurization	There is no current power LOFTTR2 SG overfill operational analysis	Within the following times after reaching the end of RCS depressurization target pressure: 4.9 minutes	1.8 minutes [Simulator]	YES

(*) NOTE: Terminate AFW flow by alternate means (i.e., local operation of AFW MOVs or by tripping AFW pump(s).

Attachment 7 to L-07-042

Changes to FENOC Letter L-06-157

Information from FENOC Letter L-06-157, Supplemental Information – SGTR Analysis Update, has been updated and provided in this attachment. A summary of the affected changes to the applicable sections of the Licensing Report is provided below:

- Responses to RAI X.5 and X.6 were updated to address the procedures for mitigation of the SGTR event.

(Ref. FENOC Letter L-06-157, Supplemental Information – SGTR Analysis Update, dated November 30, 2006.)

NOTE: The changes have been formatted such that deletions are shown with a strike-through and insertions are shown double-underlined for ease of comparison with previously docketed information.

Updated Responses to RAI X.5 and X.6

L-06-157 Enclosure 1

X.3 (Applicable to RSG & EPU)

Discuss the limiting single failure assumed in the case concerning offsite dose. Compare the assumed single failure with a stuck open ADV in the failed SG after it is automatically opened following the event.

Response:

The limiting single failure in the thermal and hydraulic for dose analysis is a failed open Atmospheric Dump Valves (ADV) on the ruptured steam generator that is assumed to fail open at the time of ruptured steam generator isolation. This single failure was assumed in both the BVPS-1 supplemental steam generator tube rupture thermal and hydraulics for dose analysis and in the BVPS-2 licensing basis steam generator tube rupture thermal and hydraulics for dose analysis.

The steam generator tube rupture methodology used in the BVPS-1 and 2 EPU analyses determined that the limiting single failure for radiological consequences is a failure of the ADV (i.e., fail open) on the ruptured steam generator at the time of ruptured steam generator isolation. The scenario of a failure of an ADV at accident initiation was not considered.

X.4 (Applicable to RSG & EPU)

Confirm that a concurrent LOOP is assumed in the SGTR analysis.

Response:

The steam generator tube rupture analyses performed for the BVPS EPU assume a LOOP concurrent with reactor trip.

X.5 (Applicable to RSG & EPU)

Confirm that the operator actions assumed in the SGTR analysis are consistent with the BVPS-1 and 2 EOPs.

Response:

The operator actions assumed in the SGTR analysis are consistent with the BVPS-1 and 2 EOPs. These operator actions and corresponding EOP steps are as follows:

- Isolate AFW flow to the ruptured SG. This action is accomplished in E-3, "Steam Generator Tube Rupture," Step 5 (BVPS-1 and 2) that checks ruptured SG level. *(See Note 1)

- Isolate steam flow (close MSIV) from the ruptured SG. This action is accomplished in E-3, "Steam Generator Tube Rupture," Step 4 (BVPS-1 and 2) that isolates flow from the ruptured SG.

- Initiate cooldown from the intact SGs via the main steam system after MSIV closure. This action is accomplished in E-3, "Steam Generator Tube Rupture," (Step 8, BVPS-1; Step 7, BVPS-2) that initiates RCS cooldown. *(See Note 1)

- Initiate RCS depressurization (open pressurizer PORV) after completion of the cooldown. This action is accomplished in E-3, "Steam Generator Tube Rupture," Step 18 (BVPS-1 and 2) that depressurizes the RCS to minimize break flow and refills the pressurizer.
- Terminate SI (isolate the high head safety injection flow path) after completion of RCS depressurization. This action is accomplished in E-3, "Steam Generator Tube Rupture," Step 20 (BVPS-1 and 2) that checks if SI flow should be terminated.
- Isolate ruptured SG ADV (which is assumed to fail open after its MSIV has closed). If the ruptured SG atmospheric steam dump valve fails open after its MSIV is closed in E-3, then the left hand page item that checks if any SG pressure is dropping in an uncontrolled manner will initiate a transition to E-2, "Faulted Steam Generator Isolation." In E-2, Step 5 (BVPS-1 and 2) will isolate the atmospheric steam dump valve on the ruptured SG.
- Supplement PPDWST volume during the 8 hour cooldown to RHR initiation conditions. This action is accomplished by a left hand page item in E-3, "Steam Generator Tube Rupture," (BVPS-1 and 2). The operator monitors PPDWST level, and upon reaching the low level alarm, then makeup is initiated to the tank.

*Note 1:
Additionally ~~for Unit 2~~:
- **BVPS-1 and** BVPS-2, E-0, Reactor Trip or Safety Injection, Symptomatic Response / Unexpected Conditions No. 4, "Steam Generator Tube Rupture Criteria" and **for BVPS-2** No. 5, "Steam generator Tube Rupture and Loss of 480 VAC Emergency Bus.

- **BVPS-1 and** BVPS-2, E-3, Steam generator Tube Rupture, Symptomatic Response / Unexpected Conditions No. 6, "Steam Generator Tube Rupture Criteria" and **for BVPS-2** No. 7, "Steam generator Tube Rupture and Loss of 480 VAC Emergency Bus.

- BVPS-2, Attachment A-1.21, "Steam Generator Tube Rupture With Malfunctions."

X.6 (Applicable to RSG & EPU)

Describe EOP steps that would provide early control of AFW flow in feeding the ruptured SG to prevent SG overfill.

Response:

Isolation of AFW flow to a ruptured SG is desired when narrow range level reaches the indicating range to limit any release from the ruptured SG. For BVPS-1, during a SGTR, the operator transitions from E-0, "Reactor Trip or Safety Injection," to E-3, "Steam Generator Tube Rupture," at Step 16 in E-0. The step to isolate flow from the ruptured SG is the fifth step in E-3. However, as a preemptive action, the operator is permitted to isolate AFW flow to the ruptured SG. Preemptive actions are action steps in the EOPs that are performed early to stabilize plant parameters. Preemptive actions are only performed with the Shift Manager or the Unit Supervisor concurrence and after the Immediate Action Steps are performed.

For **BVPS-1 and** BVPS-2, E-0, Reactor Trip or Safety Injection, Symptomatic Response / Unexpected Conditions No. 4, "Steam Generator Tube Rupture Criteria" provides direction to the operator to isolate AFW to the ruptured SG if a ruptured SG is identified and SG level in the affected SG is greater than the required level.

For BVPS-2, E-0, Symptomatic Response / Unexpected Conditions No. 5, "Steam Generator Tube Rupture and Loss of 480 VAC Emergency Bus" provides the same direction, but addresses the loss of power to the AFW isolation valve. In this case the operator is directed to Attachment A-1.21, "Steam Generator Tube Rupture With Malfunctions" which provides direction to isolate AFW flow to the ruptured SG.

This same direction is provided **for BVPS-1 and BVPS-2** in E-3, Steam Generator Tube Rupture, Symptomatic Response / Unexpected Conditions No. 6, "Steam Generator Tube Rupture Criteria" and **BVPS-2** No. 7, "Steam Generator Tube Rupture and Loss of 480 VAC Emergency Bus".