FENOC

FirstEnergy Nuclear Operating Company

James H. Lash
Site Vice President

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

November 30, 2006
L-06-157

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
Supplemental Information – Steam Generator Tube Rupture (SGTR) Analysis Update

This submittal provides updated information relative to the Steam Generator Tube Rupture (SGTR) analysis for the Beaver Valley Power Station (BVPS) Unit No. 1 and 2 and the impact on the Extended Power Uprate (EPU).

The following attachments contained within this submittal are:

- Attachment 1 provides a summary of the updated SGTR analysis results for BVPS Unit No. 2.
- Attachment 2 identifies changes to previously docketed information relative to EPU and the BVPS Unit No. 2 SGTR analysis, and includes information to supplement previously provided information contained in References 1, 2, and 3.
- Attachment 3 contains the plans for updating the BVPS Unit No. 1 SGTR analysis.
- Attachment 4 summarizes the regulatory commitments contained within this submittal.

A001

If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager – FENOC Fleet Licensing, at (330) 315-7243.

Sincerely,

Richard A. Mende

for James H. Lash

Attachments:

1. Summary Results: Updated Steam Generator Tube Rupture (SGTR) Analysis for BVPS-2
2. Changes to Previously Docketed Information Relative to EPU and the BVPS-2 SGTR Analysis
3. Plans for Updating the Steam Generator Tube Rupture (SGTR) Analysis For BVPS-1
4. Commitment List

References:

1. FENOC Letter L-04-125, License Amendment Request Nos. 302 and 173, dated October 4, 2004.
2. 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.
3. FENOC Letter L-06-003, Additional Information in Support of License Amendment Request Nos. 302 and 173, dated January 25, 2006.

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)

Summary Results: Updated Steam Generator Tube Rupture (SGTR) Analysis for BVPS-2

The Atmospheric Steam Dump Valve (ASDV) capacities credited in the SGTR analysis did not include the effect of the upstream piping friction losses. Therefore, the actual valve capacities are effectively less when the friction loses are considered. This issue involving the BVPS-2 SGTR analysis was documented in the BVPS Corrective Action Program for resolution and the investigation was completed prior to EPU implementation for BVPS-2. A summary of the results of the BVPS-2 SGTR re-analysis using the corrected valve capacities, which include the piping friction losses, is provided herein.

Summary Results:

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 SGTR analysis for BVPS-2 has been updated to account for the reduced ASDV capacity. The analysis update addresses the impact on the following areas:

- Dose – the dose analysis previously performed (with higher ASDV capacities) remains bounding since the analysis assumes an ASDV on the ruptured SG fails wide open.
- Overfill – this analysis was updated and shows that the current dose analysis is maintained since no water relief through the ASDVs or Main Steam Safety Valves (MSSVs) occurs. With the reduced valve capacities, the water in the ruptured SG could partially fill the main steam line. However, the piping was analyzed and remains acceptable under the postulated event since partitioning of radioactive iodine is retained 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 8 hours, thus maintaining the original dose assumption.

During the investigation into quantifying the impact of the lower ASDV capacities, the following three (3) additional concerns were identified, and addressed as discussed below.

1) The BVPS-2 Emergency Operating Procedures (EOPs) did not contain steps to:

   a) Reset Safety Injection (SI) prior to stopping the Motor-Driven Auxiliary Feedwater (MDAFW) pumps, and

   b) Locally trip the Turbine-Driven Auxiliary Feedwater (TDAFW) pump should the control room action fail.

   Therefore, the design basis assumptions for this event were not correctly reflected in the SGTR mitigating EOPs for BVPS-2.

   To address this concern, the BVPS-2 EOP changes have been completed. Simulator and field validations have been conducted to demonstrate that the steps can be performed within the required times.

2) A more restrictive single failure scenario for the BVPS-2 SGTR event has been identified with respect to a SG overfill. The most limiting single failure has been determined to be a failure of the 'A' Train 480 VAC Substation 2-8 (Bus 2N).

For the purpose of SG overfill mitigation, Westinghouse generic WCAP-10698-P-A (Reference 1) and the BVPS-2 plant specific WCAP-12737 (Reference 2) determined the worst case failure to be a failure to open the ASDV on an intact SG. It was concluded that the effects of this failure bounded the effects of not being able to close the AFW discharge isolation valve on the line feeding the ruptured SG. However, it was not recognized at that time that at BVPS-2, a credible single failure of 480 VAC Substation 2-8 (Bus 2N) will de-energize both Motor Control Center MCC-2-E13 (which powers ASDVs 2SVS-PCV101B and C, as well as the AFW discharge isolation valves 2FWE-HCV100A, C and E) and MCC-2-E5 (which powers 2SVS-PCV101A). Therefore, a postulated failure of Bus 2N will disable both functions simultaneously. Such a failure affects both the ability to isolate AFW flow to the ruptured SG and the ability to open either the ASDVs or the RHR Valve (depending on the 480V bus that is lost). It has been determined that a failure of the 480 VAC "A" Train (Orange Bus) is bounding for this event.

To address this concern, the most limiting single failure of the 480VAC Train "A" (Orange Bus) was accounted for in the SGTR analysis update. This requires that the TDAFW pump be tripped from the control room and resetting SI prior to stopping the MDAFW pumps in order to support the required analytical time to isolate AFW flow to the ruptured SG within 2 minutes. This action can only be accomplished by continuing to credit the non-Class 1E trip circuit. This issue has been addressed in accordance with RIS 2005-20 "Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, Information to Licensees regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability". As such, the resolution of the non-Class 1E trip circuit is being addressed within the Corrective Action Program.

3) The TDAFW pump can only be stopped from the Control Room by crediting the non-Class 1E switch.

The existing design basis analysis assumptions identified includes a concurrent Loss of Offsite Power (LOOP) with the SGTR and credits only safety class components. The design basis analysis also assumes the TDAFW pump would be stopped by shutting its steam supply isolation valves from the benchboard. However, such an action would not be available for the SGTR design basis event at BVPS-2. The steam supply to 2FWE-P22 cannot be isolated by shutting the steam supply solenoids (2MSS-SOV105A-F), even after resetting SI, because the benchboard control circuits are also interrupted by Reactor Coolant Pump Bus Undervoltage, such as would occur on the concurrent postulated LOOP.

The use of the benchboard trip pushbutton for tripping the TDAFW from the control room would be necessary to meet the required 2 minutes assumed in the analysis. However, the BVPS-2 trip pushbutton circuit is non-Class 1E.

To address this concern, a 10 CFR 50.59 Evaluation has been performed to address the analytical changes and the EOP updates relative to SGTR analysis for BVPS-2. The evaluation concluded that prior NRC approval was not required to implement the changes, and the documentation for BVPS-2 EPU has been supplemented to reflect the inclusion of the SGTR re-analysis. (See Attachment 2 for updated information relative to the BVPS-2 SGTR re-analysis.)

References:

1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987.
2. WCAP-12737, "LOFTTR2 Analysis for a Steam Generator Tube Rupture for Beaver Valley Power Station Unit 2," October 1990 (approved by NRC SER for BVPS-2, dated July 15, 1994)

Attachment 2 of L-06-157

Changes to Previously Docketed Information Relative to EPU and the BVPS-2 SGTR Analysis

Sections 5.4.2 and 5.4.3 of the Extended Power Uprate (EPU) License Amendment Request (LAR) Licensing Report (FENOC letter L-04-125 dated October 4, 2004) provides the analysis for the Steam Generator Tube Rupture (SGTR) event. The information relative to the BVPS-2 SGTR evaluation has been supplemented (see attached Enclosure 1 of this Attachment) and a summary of the changes to Sections 5.4.2 and 5.4.3 of the Licensing Report is provided below:

Section 5.4.2 "BVPS-2 Margin to Steam Generator Overfill Analysis":

- The BVPS-2 SGTR re-analysis for EPU conditions, including the most limiting single failure and operator action times to terminate AFW flow to the ruptured steam generator.
- Margin to overfill, as defined in Section 5.4.2.4 Acceptance Criteria and Results.
- Changes to Table 5.4.2-1 and Table 5.4.2-2 for operator action times for SGTR analysis and margin to overfill analysis.
- Figures 5.4.2-1 through 5.4.2-6 revised to reflect the BVPS-2 SGTR re-analysis and margin to overfill analysis.

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

- Changes to Section 5.4.3.2 under Operator Action Times to account for the increase in time to initiate RCS cooldown in the margin to overfill analysis.

In addition, information provided previously in support of the EPU License Amendment Request Nos. 302 and 173 is being supplemented to reflect the BVPS-2 SGTR re-analysis, and is included with this attachment as Enclosures 2 and 3. Documents which provided this information are as follows:

1. 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.
   - Response to RAI A.1, Table A.1-22 "SGTR Overfill Analysis" which identifies the BVPS-2 EPU values for operator action times for mitigation of the SGTR event.
   - Response to RAI X.2, which describes the most limiting single failure with respect to steam generator overfill for BVPS-2.
   - Responses to RAI X.5 and X.6, which discusses the EOPs for mitigation of the SGTR event.
2. FENOC Letter L-06-003, Additional Information in Support of License Amendment Request Nos. 302 and 173, dated January 25, 2006.
   - Enclosure 2, Table 2-1 "Comparison of BVPS-2 Operator Action Times in EPU UFSAR Safety Analysis" (operator action times used in EPU analysis for initiation of cooldown, inside and outside the control room).

The BVPS-2 SGTR re-analysis resulted in operator action times for initiating cooldown equivalent to the operator action times reported for BVPS-1 for this scenario.

Enclosure 1 of Attachment 2 to L-06-157

Information from Section 5.4.2 "BVPS-2 Margin to Steam Generator Overfill Analysis" and Section 5.4.3 "BVPS-2 Thermal and Hydraulic Analysis for Offsite Radiological Consequences" of the EPU Licensing Report (Ref. FENOC Letter L-04-125, License Amendment Request Nos. 302 and 173, dated October 4, 2004)

5.4.2 BVPS-2 Margin to Steam Generator Overfill Analysis

In support of the EPU Project for BVPS-2, an analysis for a design basis steam generator tube rupture (SGTR) event has been performed to demonstrate that the potential consequences are acceptable. The analysis supports a full power average temperature (T_{avg}) operating window of 566.2° to 580.0°F as well as a main feedwater temperature window of 400° to 455°F. The analysis discussed herein assumes that up to 22% of the steam generator tubes are plugged and the unit is operating at a NSSS power level of 2910 MWt.

The major hazard associated with a 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. Therefore, an analysis must be performed to assure that the offsite radiation doses resulting from a SGTR are within the allowable guidelines. One of the major concerns for a SGTR is the possibility of steam generator water relief since this could potentially result in a significant increase in the offsite radiation doses. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The analysis confirmed that the ruptured steam generator does not overfill (see criteria in Section 5.4.2.4) and water relief through the MSSVs does not occur. A thermal and hydraulic analysis is also performed to determine the input for use in calculating the offsite radiation doses, assuming the limiting single failure relative to offsite doses without water relief (See Section 5.4.3).

Plant response to the SGTR event is modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the ruptured steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a SGTR based on the BVPS-2 Emergency Operating Procedures, which are based on the Westinghouse Owners Group Emergency Response Guidelines.

The LOFTTR2 analyses are performed for the time period from the SGTR until the primary and secondary pressures are equalized (break flow termination). In the margin to overfill analysis presented in this section, the water volume in the secondary side of the ruptured steam generator is calculated as a function of time to demonstrate that overfill does not occur. The thermal and hydraulic analysis to develop input for the radiological consequences is presented in Section 5.4.3.

Introduction

The 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.

The analysis outlined herein was performed to determine the margin to steam generator overfill for a design basis SGTR event for BVPS-2. The analysis was performed using the LOFTTR2 program and the methodology developed in Reference 1, 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.

5.4.2.2 Input Parameters and Assumptions

The margin to overfill analyses model the plant operating at the lower end of the T_{avg} window since a lower operating temperature results in a higher mass flow rate through the broken tube and less steam released from the ruptured steam generator. The analysis assumes that the plant is operating with the feedwater temperature at the low end of the temperature window, since this results in a higher mass of water in the steam generator at the start of the event, which limits the amount of break flow and auxiliary feedwater (AFW) that can accumulate in the ruptured steam generator. Maximum (22%) tube plugging is assumed in the limiting thermal-hydraulic analysis to determine the margin to overfill since this reduces the heat transfer to the ruptured steam generator, minimizing the amount of mass released from the steam generator due to steaming. The reduced heat transfer also prolongs the cooldown period, leading to delayed break flow termination. Although maximum tube plugging results in a lower initial water mass in the ruptured steam generator, which increases the available margin to overfill at the start of the event, this is not more limiting than the lower heat transfer effects described above. This has been confirmed via sensitivities performed for the BVPS-2 EPU Project.

Design Basis Accident

The 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. It was also assumed that a loss-of-offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The potentially limiting single failures with respect to margin to steam generator overfill for a SGTR are outlined below.

1. Mechanical Failure of the Auxiliary Feedwater (AFW) Flow Control Valve:~~ Failure~~

   The AFW control valves are normally open and are used to control inventory in the intact steam generators and terminate feedwater flow to the ruptured steam generator. A failure of the ruptured steam generator control valve would require the operator to perform additional actions to stop the associated AFW pump in order to terminate AFW flow to the ruptured steam generator. For the AFW control valve failure evaluation, it was assumed that ~~120~~180 seconds (~~2~~3 minutes) of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump. The additional ~~2~~3 minutes were added to the assumed time for AFW isolation without this failure. The additional AFW provided to the ruptured steam generator during the additional 3~~2~~ minutes reduced the margins to overfill by way of additional mass added and by a reduction in steaming from the ruptured steam generator. ~~For BVPS-2 the~~

~~single failure of the AFW control valve was found to be less limiting than a single failure of the atmospheric relief valves, in previous sensitivity studies.~~

2. Mechanical Failure of a Residual Heat Release Valve (RHRV) or an Atmospheric Steam Dump Valve (ASDV) on an Intact Steam Generator:~~s Power Supply Failures~~

Since offsite power is assumed to be lost at reactor trip for the SGTR analysis, the atmospheric steam dump valves (ASDVs) and/or residual heat release valve (RHRV) are relied upon to cool the reactor coolant system. A failure that affects the availability of these valves for the cooldown can reduce the available steam release capability increasing the time required for the cooldown and/or delaying the start of the cooldown. This in turn results in increased break flow and a reduction in the margin to overfill. Two failures that impact the cooldown are postulated:

a. A single mechanical failure ~~of the orange power supply can result in the loss~~ of the control room operation of one of the ASDVs. For this failure ~~the orange bus failure~~, the RHRV can be operated from the main control room (MCR) after the gate valve upstream of the RHRV, connected to the ruptured SG, is closed manually, while one ASDV can be operated ~~locally with a hand pump~~from the control room allowing the cooldown to be completed with the combined capacity of the two valves (one ASDV and the RHRV). ~~In this case initiation of the cooldown is delayed to allow for the local actions required for initiation of the cooldown.~~

b. A single mechanical failure of ~~the purple bus power supply results in a loss of power to~~ the plant residual heat release valve (RHRV). For ~~a~~ the mechanical failure of the RHRV~~purple bus failure (failure of the RHRV)~~, two ASDVs can be operated from the MCR without delay since local access to valves is not required.

3. Failure of the 480 Volt Orange Bus (Train "A"):

The single failure of the 480 Volt Orange Bus renders multiple components inoperable as described below:

a. The failure of the 480 Volt Orange Bus causes the failure of the AFW control valve. A failure of the ruptured steam generator control valve would require the operator to perform additional actions to stop the associated AFW pump in order to terminate AFW flow to the ruptured steam generator. For the AFW control valve failure evaluation, it was assumed that 120 seconds (2 minutes) of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump. The additional 2 minutes were added to the assumed time for AFW isolation without this failure. The additional AFW provided to the ruptured steam generator during the additional 2 minutes reduced the margins to overfill by way of additional mass added and by a reduction in steaming from the ruptured steam generator.

b. The failure of the 480 Volt Orange Bus will also cause the failure of the control room operation of the ASDVs. In this case initiation of the cooldown is delayed to allow for the local actions required for the initiation of the cooldown. After the operators isolate the ruptured steam generator, the cooldown is started with local operation of the two ASDVs and manually with the RHRV after the gate valve upstream of the RHRV, connected to the ruptured steam generator, is closed locally. The cooldown is initiated 10 minutes following ruptured steam generator isolation with both the ASDVs and RHRV.

~~The single failures described above bound a single mechanical failure of an ASDV or RHRV.~~

The analyses performed for the BVPS-2 EPU Project specifically evaluated the~~se~~ single ~~cooldown capability~~ failures provided above~~. However, since the difference in the margin between the two scenarios is small, neither scenario can be identified as clearly limiting~~. For BVPS-2 EPU Project the limiting single failure was found to be the failure of the 480 Volt Orange Bus. The analyses presented in this report consider the failure of the ~~RHRV~~ 480 Volt Orange Bus since it resulted in the smallest margin to overfill.

Conservative Assumptions

Plant responses until break flow termination are calculated using the LOFTTR2 computer code. The conservative conditions and assumptions which are found in Reference 1 are also used in the LOFTTR2 analysis to determine margin to steam generator overfill for BVPS-2 with the exception of the following differences.

1. Reactor Coolant System Flow Asymmetry

   The margin to overfill analysis considered a primary flow asymmetry based on operation at the EPU power with the lowest Tavg, the highest steam generator tube plugging, and the lowest feedwater temperature. The steam generator tube plugging is modeled considering loop-to-loop flow asymmetry. The loop asymmetry evaluation assumes that the loop with the ruptured steam generator is at the maximum steam generator plugging level and has the lowest loop flow. These assumptions contribute to the ruptured steam generator operating at a lower temperature and producing less power, which in turn provides a conservatively higher initial secondary water mass.

2. Reactor Trip and Turbine Runback

   A turbine runback can either be initiated automatically or the operator can manually reduce the turbine load to attempt to prevent a reactor trip on overtemperature-ΔT. This turbine runback would result in an increase in secondary water mass compared to the mass at full power. Although turbine runback is simulated in this analysis, credit is not taken for delaying reactor trip. Until reactor trip and the assumed loss of offsite power, the main feedwater control system is assumed to maintain a constant steam generator water level. Therefore, until reactor trip, the break flow does not reduce the margin to overfill. An earlier reactor trip will result in a higher RCS pressure at the time of reactor trip, and a higher post-trip break flow rate. For this analysis, the times of automatic turbine runback initiation and reactor trip on overtemperature-ΔT were determined by modeling the BVPS-2 control and protection system. The effect of turbine runback was conservatively simulated by increasing the secondary mass by the differential in the mass corresponding to operation at 100% power and at the power at reactor trip (modeling the plants turbine runback program), and performing the analysis at 100% power.

3. Steam Generator Secondary Mass

A higher initial secondary water mass in the ruptured steam generator was determined by Reference 1 to be conservative for overfill. As noted above, the flow asymmetry is skewed to the ruptured loop to provide the highest initial steam generator water mass in the ruptured generator. Also, turbine runback was assumed to be initiated and was simulated by artificially increasing the initial steam generator water mass. The initial steam generator total fluid mass that is assumed in the analysis is 10% above the nominal full power fluid mass attributable to the flow asymmetry, plus mass to simulate the effect of turbine runback.

4. AFW System Operation

For this analysis, the maximum AFW flow rate of 310 gpm to the ruptured steam generator is assumed to be initiated immediately after reactor trip with a 15-second startup delay credited.

Operator Action Times

In the event of a SGTR, the operator is required to take actions to stabilize the plant and terminate the primary-to-secondary leakage. The operator actions for SGTR recovery are provided in the BVPS-2 EOP E-3, and major actions are explicitly modeled in this analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory and termination of SI to stop primary-to-secondary leakage. These operator actions are described below.

1. Isolate AFW if necessary and identify the ruptured steam generator.

High secondary side activity, as indicated by the air ejector discharge radiation monitor, steam generator blowdown sample radiation monitor, or main steamline radiation monitor, typically will provide the first indication of a SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level, high radiation from a steam generator water sample, or a high radiation indication. For a SGTR that results in a reactor trip at high power as assumed in this analysis, the steam generator water level as indicated on the narrow range will decrease significantly for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary-to-secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of a SGTR event and also identifies the ruptured steam generator. In the BVPS-2 EOP for steam generator tube rupture, the operator is directed to maintain the level in the ruptured steam generator between 12 and 50% on the narrow range instrument. In the conservative scenario analyzed to minimize the margin to steam generator overfill, isolation of AFW flow to the affected SG can occur even before it is identified as a ruptured SG, based on the high SG level. Based on plant simulator studies modeling this scenario, it is assumed that AFW flow to the ruptured steam generator would be isolated when level in the steam generator reached 27.5% narrow range level (i.e., mid-point between just in range (5%) and 50%) or at 5.5 minutes from reactor trip, whichever is longer. For the single failure of the 480 volt orange bus, an additional 2 minutes of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump(s).

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator by (1) minimizing the accumulation of feedwater flow and (2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage. As noted above it is assumed that AFW flow to the ruptured steam generator would be isolated when level in the steam generator reached 27.5% narrow range level or at 5.5 minutes from reactor trip, whichever is longer. For the single failure of the 480 volt orange bus, an additional 2 minutes of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump(s). Complete isolation of steam flow from the ruptured steam generator (MSIV and ASDV closure) is verified when the narrow range level reaches 27.5% in the ruptured steam generator or at 15 minutes after reactor trip, whichever is longer.

3. Cooldown the Reactor Coolant System (RCS) using the intact steam generators.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This establishes adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. If offsite power is lost, the RCS would be cooled using the ASDVs on the intact steam generators, and residual heat relief valve (RHRV) after the gate valve upstream of the RHRV connected to the ruptured SG is closed. However, in the analysis performed for BVPS-2, the plant is cooled by the following actions: (1) with local operation of the two ASDVs, and (2) manually with the RHRV, after the gate valve upstream of the RHRV that is connected to the ruptured steam generator is closed locally. In this case, the cooldown is initiated 10 minutes following ruptured steam generator isolation with both the ASDVs and RHRV. ~~using the two intact steam generator ASDVs due to failure of the RHRV power supply (purple bus power supply), or through the RHRV and a single ASDV, in the event of a failure of the orange bus power supply.~~

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, SI flow will tend to increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be established. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until the RCS and the ruptured steam generator pressures equalize, an "excess" amount of inventory is needed so that the pressurizer level remains on span. The "excess" amount required depends on the RCS pressure and reduces to zero when the RCS pressure equals the pressure in the ruptured steam generator.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. Since offsite power is assumed to be lost at the time of reactor trip, the RCPs are not running and thus normal pressurizer spray is not available. Therefore, the depressurization is modeled using a pressurizer power operated relief valve (PORV).

5. Terminate SI to stop primary-to-secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory so that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary-to-secondary leakage. Primary-to-secondary leakage will continue after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown flow, and pressurizer heaters will then be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator.

Since these major recovery actions are modeled in the SGTR analysis, it is necessary to establish the time required to perform these actions. Although the intermediate steps between the major actions are not explicitly modeled, it is necessary to account for the time required to perform the steps. Note that the total time required to complete the recovery operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (i.e., RCS cooldown) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times to identify and isolate the ruptured steam generator, to initiate RCS cooldown, to initiate RCS depressurization, and to perform safety injection termination are developed for the design basis analysis consistent with the methodology in Reference 1. BVPS-2 has determined the corresponding operator action times to perform these operations. The operator actions and the corresponding operator action times used for the BVPS-2 analysis are listed in Table 5.4.2-1.

5.4.2.3 Description of Analyses and Evaluations

The LOFTTR2 analysis results for the BVPS-2 margin to overfill analysis with operation at the NSSS power of 2910 MWt are described below. The sequence of events for this transient is presented in Table 5.4.2-2.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 5.4.2-1. The RCS pressure also decreases as shown in Figure 5.4.2-2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary-to-secondary leakage, automatic reactor trip occurs on an overtemperature delta-T reactor trip signal at approximately 112 seconds.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss-of-offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator ASDVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 5.4.2-3. As a result of the assumed loss-of-offsite power, main feedwater flow is assumed to be terminated and AFW flow is assumed to be automatically initiated following reactor trip.

The RCS pressure and pressurizer level continue to decrease after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal at approximately 122 seconds. The SI flow increases the reactor coolant inventory and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figure 5.4.2-4); however, the temperature differential subsequently increases as the reactor coolant pumps coast down and natural circulation flow develops. The cold leg temperature trends toward the steam generator temperature as the fluid residence time in the tube region increases. The RCS temperatures continue to slowly decrease due to the continued addition of the auxiliary feedwater to the steam generators until operator actions are initiated to cool down the RCS.

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, auxiliary feedwater flow to the ruptured steam generator is assumed to be identified and isolated when the narrow range level reaches 27.5% on the ruptured steam generator or at 5.5 minutes after reactor trip, whichever is longer. For the BVPS-2 analysis, the time to reach 27.5% is less than 5.5 minutes, and thus AFW flow to the ruptured steam generator is assumed to be isolated at 5.5 minutes. For the single failure of the 480 volt orange bus, an additional 2 minutes of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump(s) because the AFW control valve is failed. ~~Also, a~~As indicated previously, 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 less than 15 minutes, and thus the ruptured steam generator is assumed to be isolated at 15 minutes after reactor trip.

2. Cooldown the RCS to establish subcooling margin.

After isolation of the ruptured steam generator, a 2.4-minute operator action time is imposed prior to initiating the cooldown with operation of the ASDVs from the MCR. With the failure of the 480 volt orange bus failure, the ASDVs must be operated locally. Additionally, the RHRV is opened after the gate valve upstream of the RHRV, which is connected to the ruptured steam generator, is closed locally. After this time, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the ASDVs on both intact steam generators and the RHRV. ~~This assumes a failure that results in the RHRV not being available for the cooldown.~~ It is assumed that the ASDVs and RHRV are opened at approximately 1612~~132~~ seconds for the RCS cooldown. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 21°F for subcooling uncertainty. When these conditions are satisfied at 2234~~020~~ seconds, it is assumed that the operator closes the ASDVs and RHRV to terminate the cooldown. This cooldown is performed so that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure 5.4.2-3, and the effect of the cooldown on the RCS temperature is also shown in Figure 5.4.2-4. The pressurizer level and RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 5.4.2-1 and 5.4.2-2.

~~This case was determined to result in slightly less margin compared to the case with the failure of the power supply to the intact steam generator ASDVs. In that case, a 7-minute operator action time is imposed prior to initiation of the cooldown to allow for manual opening of a single ASDV to supplement the RHRV. The delay is compensated by the higher steam release capacity of the RHRV and single ASDV compared to the two ASDVs credited in the case discussed above. In the case with the cooldown initiation delayed by 7 minutes, the cooldown with the higher steam dump capacity was completed only a little over a minute later, at 2094 seconds.~~

3. Depressurize the RCS to restore inventory.

The RCS depressurization is performed to establish adequate coolant inventory prior to terminating SI flow. A 4-minute operator action time is included prior to the RCS depressurization. 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 2476~~262~~ 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 2552~~334~~ 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 (Figure 5.4.2-2) reduces the break flow as shown in Figure 5.4.2-5 and increases SI flow to refill the pressurizer, as shown in Figure 5.4.2-1.

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 2732~~514~~ seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 5.4.2-2.

The intact SG ASDVs, which were used for the cooldown, also automatically open (at about 2806~~618~~ seconds) to dump steam to maintain the prescribed RCS temperature so 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 primary-to-secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

5.4.2.4 Acceptance Criteria and Results

The analysis is performed to demonstrate that the secondary side of the ruptured steam generator and steam piping up to the MSSVs does not completely fill with water. The available secondary side volume of a single BVPS-2 steam generator is 5730 ft^3. ~~Margin to overfill is demonstrated provided the transient calculated steam generator secondary side water volume is less than 5730 ft^3.~~ Additional margin is demonstrated considering the smallest water volume in the piping up to the MSSV being 314 ft^3. Margin to ~~MSSV flooding~~overfill ~~-~~is demonstrated provided the transient calculated secondary side water volume is less than 6044 ft^3.

The primary-to-secondary break flow rate throughout the recovery operations is presented in Figure 5.4.2-5. The water volume in the ruptured steam generator is presented as a function of time in Figure 5.4.2-6. The secondary side volume of a single BVPS-2 steam generator, up to the outlet nozzle, is 5730 ft^3. The peak ruptured steam generator water volume of 6023~~5656~~ ft^3 is indicated in Figure 5.4.2-6 showing that there is 21~~74~~ ft^3 of margin to overfill ~~and 388 ft^3 of additional margin to MSSV flooding~~. ~~(The orange bus power supply failure case resulted in an additional 5 ft^3 of margin, compared to the purple bus power supply failure.)~~ Therefore, it is concluded that steam generator overfill will not occur for a design basis SGTR for BVPS-2 at the NSSS power of 2910 MWt~~, and additional margin exists considering the water volume available prior to MSSV flooding~~.

5.4.2.5 Conclusions

It is concluded that steam generator overfill will not occur for a design basis SGTR for BVPS-2 at the NSSS power of 2910 MWt when operator actions are performed consistent with the times in Table 5.4.2-1. ~~Additional margin also exists considering the available water volume prior to MSSV flooding~~.

The results and conclusions of the analysis performed for SGTR margin to steam generator overfill 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.

5.4.2.6 References

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

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

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

Table 5.4.2-1 BVPS-2 Operator Action Times For Design Basis SGTR Analysis	
Action	**Time**
Isolate auxiliary feedwater flow to ruptured SG	5.5 minutes from reactor trip[(1)(2)]
Isolate steam flow from ruptured SG	15 minutes from reactor trip
Operator action time to initiate cooldown	2.4 minutes from time of steamline isolation[(3~~1~~)]
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 minutes from end of cooldown
Depressurization	Calculated by LOFTTR2
Operator action time to terminate SI following depressurization	3 minutes from end of depressurization or time to satisfy termination criteria
Pressure equalization	Calculated by LOFTTR2

Note:

(1) For the case with a mechnical failure of the AFW control valve, an additional 3 minutes is assumed prior to terminating AFW by stopping the associated AFW pump.

(2) For the case with the failure of the 480 volt orange bus, an additional 2 minutes is assumed prior to terminating AFW by stopping the associated AFW pump.

(3) ~~(1)~~For the case with the failure of the ~~power supply~~480 volt orange bus~~ to the~~, ~~intact steam generator ASDVs~~ a ~~7~~10-minute operator action time is imposed prior to initiation of the cooldown to allow for manual opening of the~~a single~~ ASDVs to supplement the RHRV.

~~(2)~~

Table 5.4.2-2 BVPS-2 Sequence of Events for Margin to Overfill Analysis	
Event	**Time**
Tube Rupture (sec)	0
Reactor Trip (Overtemperature Delta-T) – (sec)	112
SI Actuation (Low Pressurizer Pressure) – (sec)	122
AFW Initiated – (sec)	127
Isolate AFW – (sec)	~~442~~562
Isolate Ruptured SG – (sec)	1012
Initiate Cooldown with Intact SGs – (sec)	~~1132~~1612
Terminate Cooldown – (sec)	~~2020~~2234
Initiate Depressurization with Pzr PORV – (sec)	~~2262~~2476
Terminate Depressurization – (sec)	~~2334~~2552
Stop SI Flow – (sec)	~~2514~~2732
~~Steam Relief to Maintain Subcooling~~Break Flow < 0 – (sec)	~~2618~~3200
~~Break Flow < 0 – (sec)~~	~~2934~~
Ruptured SG Fills – (sec)	NA
Approximate margin to Overfill~~MSSV flooding~~ (ft^3)	~~74~~21
~~Approximate margin to MSSV flooding (ft^3)~~	~~388~~

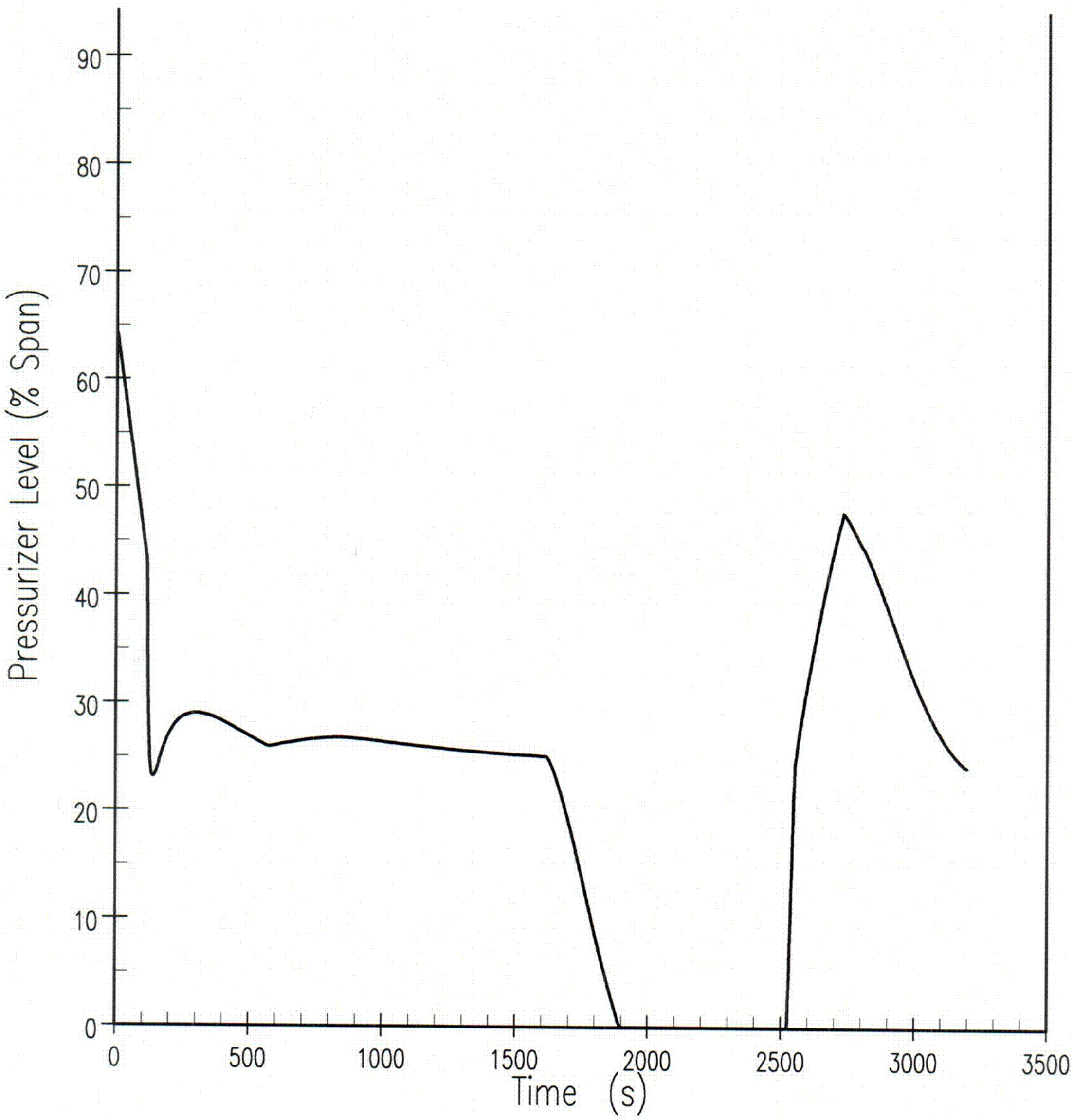

Figure 5.4.2-1
BVPS-2 Margin to Overfill Analysis
Pressurizer Level

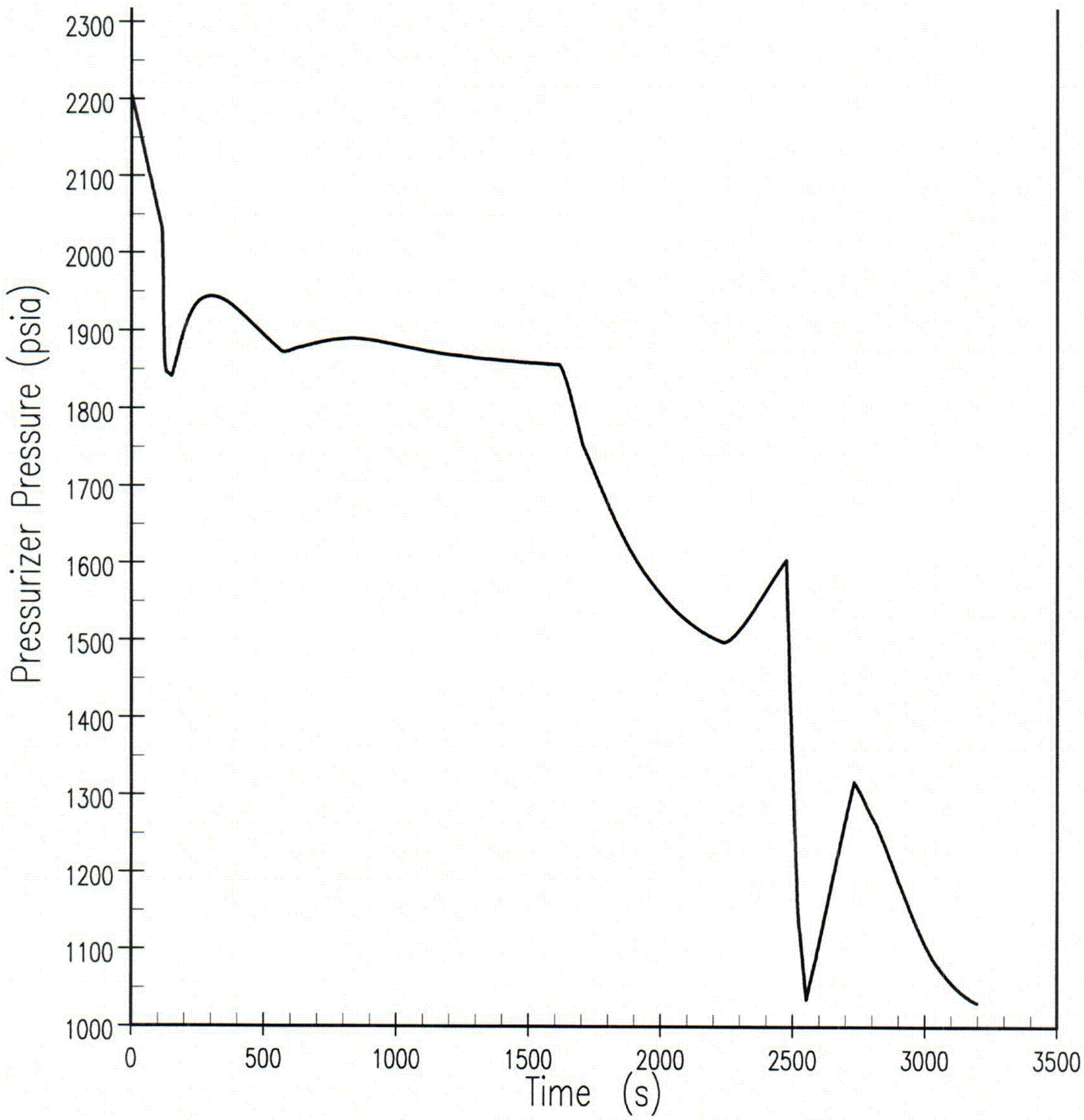

Figure 5.4.2-2
BVPS-2 Margin to Overfill Analysis
RCS Pressure

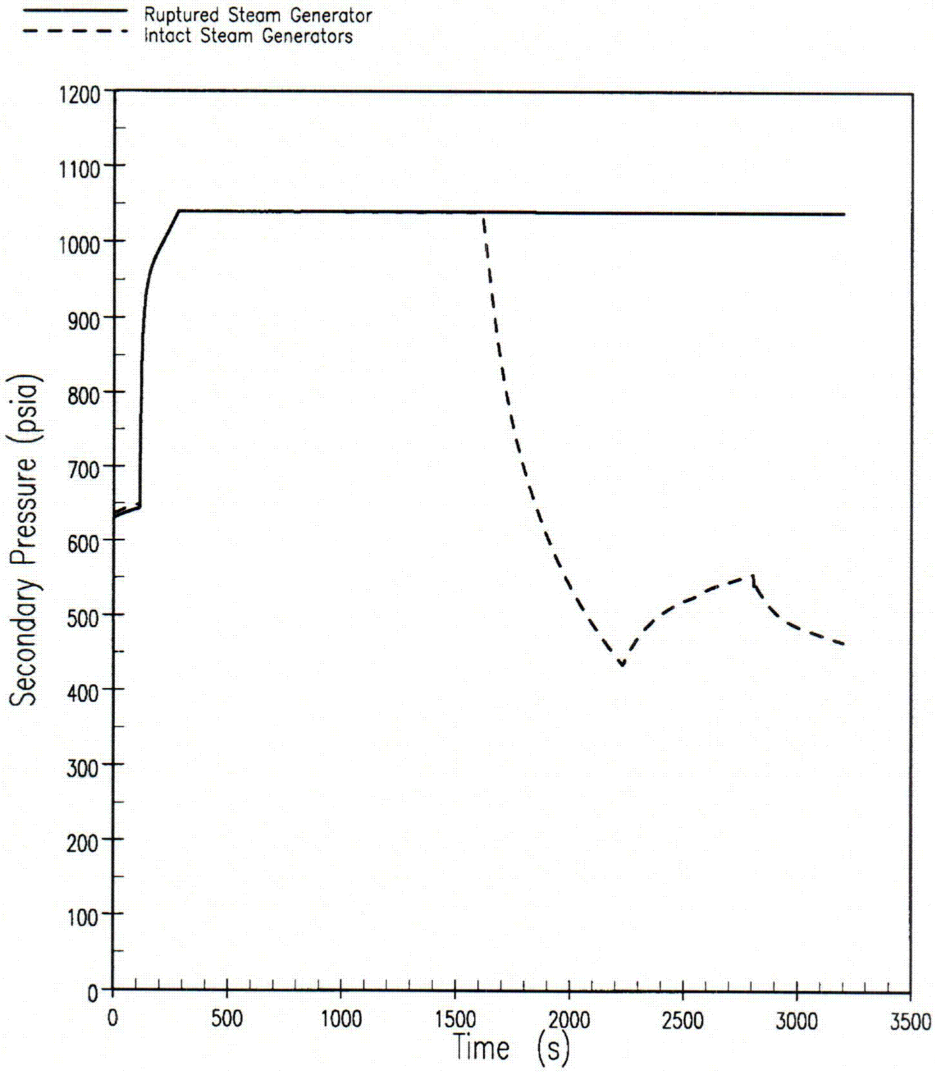

Figure 5.4.2-3
BVPS-2 Margin to Overfill Analysis
Secondary Pressure

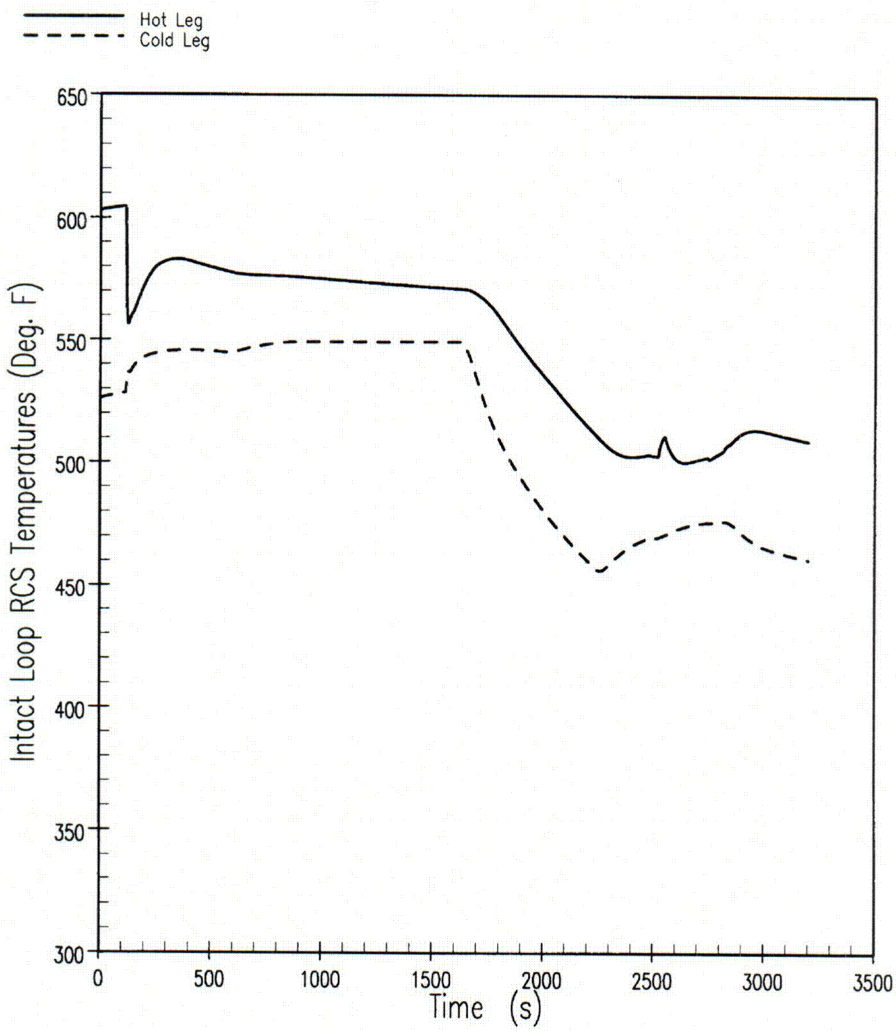

Figure 5.4.2-4
BVPS-2 Margin to Overfill Analysis
Intact Loop Hot & Cold Leg Temperatures

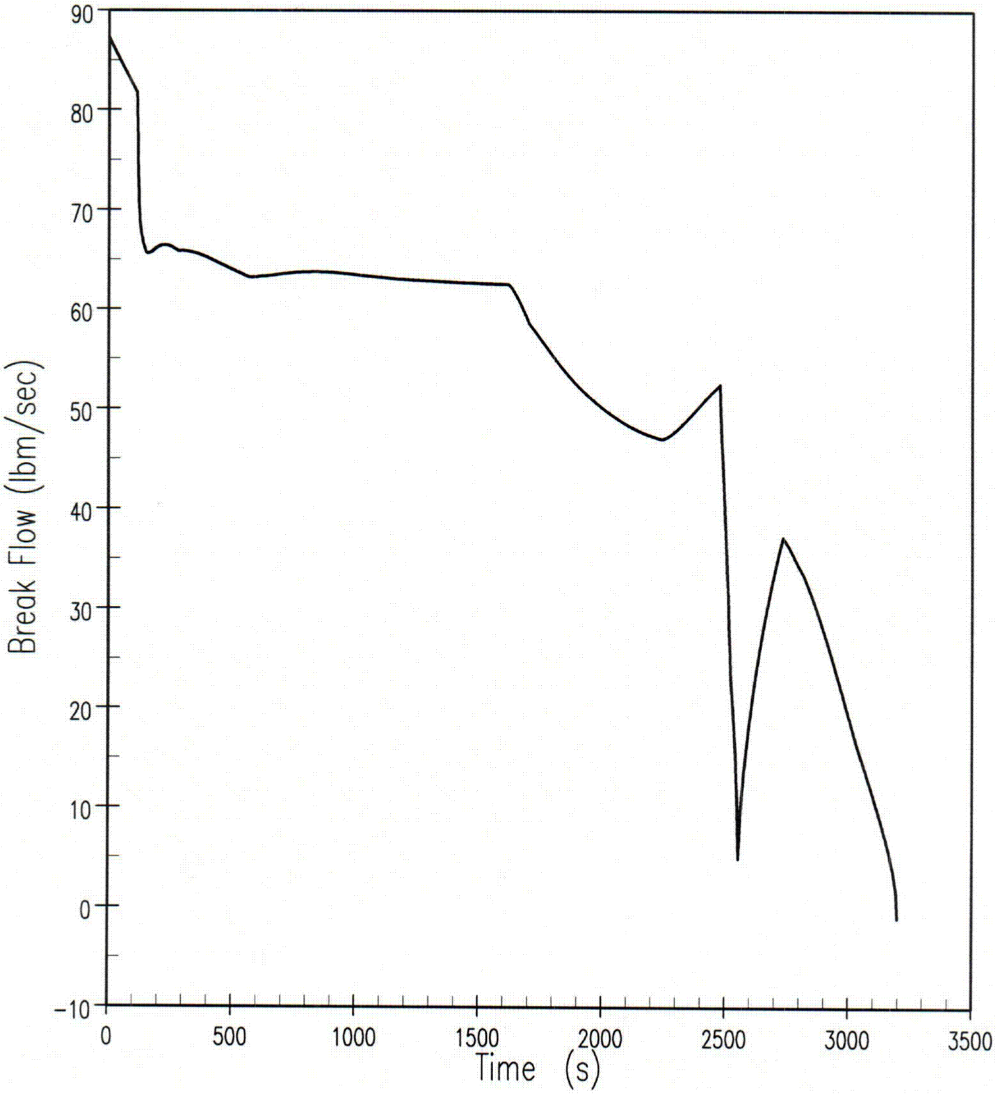

Figure 5.4.2-5
BVPS-2 Margin to Overfill Analysis
Primary-to-Secondary Break Flow

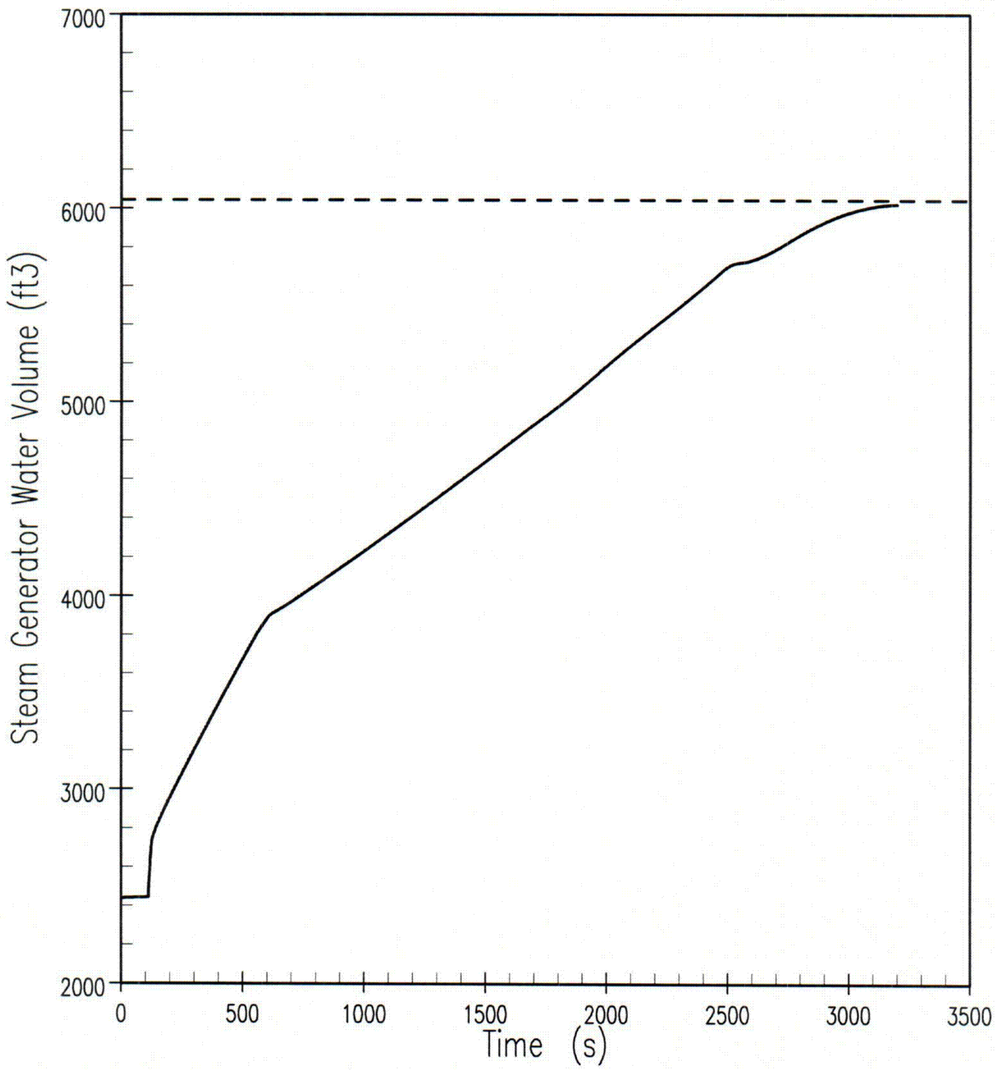

Figure 5.4.2-6
BVPS-2 Margin to Overfill Analysis
Ruptured SG Water Volume

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 minutes after the failure. Thus, it is assumed that the ruptured steam generator ASDV is isolated at 10 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, an~~the~~ additional delay time of 2 minutes ~~(Table 5.4.2-1)~~ is assumed for the operator action time to initiate the RCS cooldown. Note, the margin to overfill analysis operator actions as presented in Table 5.4.2-1 includes a 2.4 minute delay for the operation to initiate the RCS cooldown. The offsite dose analysis presented in this report assumes a higher capacity ASDV to fail open on the ruptured steam generator. The high capacity ASDV flowrate does not account for the friction loss to the ASDVs. A sensitivity to the reduced ASDV flowrates, resulting in an increase in the RCS cooldown duration, was performed using the 2.4 minute delay to initiate the RCS cooldown. The sensitivity showed that the use of the higher capacity ASDVs remains conservative and bounding with respect to the reduced ASDV flowrate (including friction losses) and the small increase in the time to initiate the RCS cooldown.

Information from 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.

- Response to RAI A.1
- Response to RAI X.2
- Response to RAI X.5
- Response to RAI X.6

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	~~7~~10	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~~2.4	Operation of ASDVs in 9 minutes	No failure case, different failure combinations analyzed
Initiate RCS depressurization, after cooldown, minutes	4.9	4	2.5	
Initiate SI Termination, after depressurization, minutes	3	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 @ 2500 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	1856	1856	
SI Flow Rate vs. Pressure (Max Safeguards), gpm vs. psig, maximum	See Table A.1-22B	See Table A.1-22B	See Table A.1-22A	
SI Full Flow Delay (without offsite power), seconds, nominal	10	10	0	

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

X.2 (Applicable to RSG & EPU)

Discuss the limiting single failure assumed in the case concerning SG overfill. Compare the assumed single failure with: 1) failure of an atmospheric dump valve (ADV) in the intact SG which causes slower RCS cooldown and increased cumulative leak flow from the RCS to the SGs, and 2) failure of AFW flow control which causes a more severe SG overfill transient.

Response:

Three single failure scenarios were considered in the BVPS-1 supplemental steam generator tube rupture analysis for margin to steam generator overfill analysis: 1) a single failure of the instrument air supply requiring local operation of the intact steam generator atmospheric relief valves (ARVs) for cooldown, 2) a single failure of a steam generator ARV to open on demand at the time of RCS cooldown, and 3) a single failure of the auxiliary feedwater isolation valves to close on demand. The limiting single failure scenario in the BVPS-1 supplemental steam generator tube rupture margin to steam generator overfill analysis is the failure of an ARV to open on demand at the time of cooldown initiation.

The potentially limiting single failures with respect to margin to steam generator overfill for a SGTR are outlined below.

1. Mechanical Failure of the Auxiliary Feedwater (AFW) Flow Control Valve

   The AFW control valves are normally open and are used to control inventory in the intact steam generators and terminate feedwater flow to the ruptured steam generator. A failure of the ruptured steam generator control valve would require the operator to perform additional actions to stop the associated AFW pump in order to terminate AFW flow to the ruptured steam generator. For the AFW control valve failure evaluation, it was assumed that 180 seconds (3 minutes) of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump. The additional 3 minutes were added to the assumed time for AFW isolation without this failure. The additional AFW provided to the ruptured steam generator during the additional 3 minutes reduced the margins to overfill by way of additional mass added and by a reduction in steaming from the ruptured steam generator.

2. Mechanical Failure of a Residual Heat Release Valve (RHRV) or an Intact Steam Generator Atmospheric Steam Dump Valve (ASDV)

   Since offsite power is assumed to be lost at reactor trip for the SGTR analysis, the atmospheric steam dump valves (ASDVs) and/or residual heat release valve (RHRV) are relied upon to cool the reactor coolant system. A failure that affects the availability of these valves for the cooldown can reduce the available steam release capability increasing the time required for the cooldown and/or delaying the start of the cooldown. This in turn results in increased break flow and a reduction in the margin to overfill. Two failures that impact the cooldown are postulated:

a. A single mechanical failure of the control room operation of one of the ASDVs. For this failure, the RHRV can be operated from the main control room (MCR) after the gate valve upstream of the RHRV, connected to the ruptured SG is closed manually, while one ASDV can be operated from the control room allowing the cooldown to be completed with the combined capacity of the two valves (one ASDV and the RHRV).

b. A single mechanical failure of the plant residual heat release valve (RHRV). For the mechnical failure of the RHRV, two ASDVs can be operated from the MCR without delay since local access to valves is not required.

3. Failure of the 480 Volt Orange Bus

The single failure of the 480 Volt Orange Bus renders multiple components inoperable as described below:

a. The failure of the 480 Orange Bus causes the failure of the AFW control valve. A failure of the ruptured steam generator control valve would require the operator to perform additional actions to stop the associated AFW pump in order to terminate AFW flow to the ruptured steam generator. For the AFW control valve failure evaluation, it was assumed that 120 seconds (2 minutes) of operator action time would be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump. The additional 2 minutes were added to the assumed time for AFW isolation without this failure. The additional AFW provided to the ruptured steam generator during the additional 2 minutes reduced the margins to overfill by way of additional mass added and by a reduction in steaming from the ruptured steam generator.

b. The failure of the 480 Orange Bus will also cause the failure of the control room operation of the ASDVs. In this case initiation of the cooldown is delayed to allow for the local actions required for the initiation of the cooldown. After the operators isolate the ruptured steam generator, the cooldown is started with local operation of the two ASDVs and manually with the RHRV after the gate valve upstream of the RHRV, connected to the ruptured steam generator, is closed locally. The cooldown is initiated 10 minutes following ruptured steam generator isolation with both the ASDVs and RHRV.

The analyses performed for the BVPS-2 EPU specifically evaluated the single failures provided above. For BVPS-2 EPU, the limiting single failure was found to be the failure of the 480 Volt Orange Bus. The analyses presented in this report consider the failure of the 480 Volt Orange Bus since it resulted in the smallest margin to overfill.

~~For BVPS-2 licensing basis steam generator tube rupture analysis, two single failure cases were analyzed: 1) Orange Bus power supply failure which results in a loss of control room operation of the ARV and, 2) Purple Bus power supply failure resulting in loss of the plant residual heat release valve (RHRV). The EPU analysis indicates that a loss of either power bus results in a similar margin to steam generator overfill. Recent evaluations performed in support of the EPU examining a failure of the auxiliary feedwater isolation valves to close on demand determined that the time required to isolate the auxiliary feedwater valves was reasonable when crediting local operator action.~~

~~As noted above, the loss of cooldown capacity is the limiting single failure with respect to margin to steam generator overfill for both BVPS-1 and 2.~~

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-2, E-0, Reactor Trip or Safety Injection, Symptomatic Response / Unexpected Conditions No. 4, "Steam Generator Tube Rupture Criteria" and No. 5, "Steam generator Tube Rupture and Loss of 480 VAC Emergency Bus.

- BVPS-2, E-3, Steam generator Tube Rupture, Symptomatic Response / Unexpected Conditions No. 6, "Steam Generator Tube Rupture Criteria" and 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 ~~15~~ 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-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.

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 in E-3, Steam Generator Tube Rupture, Symptomatic Response / Unexpected Conditions No. 6, "Steam Generator Tube Rupture Criteria" and No. 7, "Steam Generator Tube Rupture and Loss of 480 VAC Emergency Bus".

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

Enclosure 2, Table 2-1 "Comparison of BVPS-2 Operator Action Times in EPU UFSAR Safety Analysis" (page 7 of 8)

Table 2-1 (Continued)
Comparison of BVPS-2 Operator Action Times in 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 [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	Within 9.1 minutes after reactor trip	Within the following times after reactor trip: 5.5 minutes	5 minutes [Talk-through]	YES
	2. Isolate steam flow (close MSIV) from the ruptured SG	Within 9.1 minutes after reactor trip	Within the following times after reactor trip: 15.0 minutes	10 minutes [Talk-through]	YES
	3. Initiate cooldown from the intact SGs via the main steam system after MSIV closure	Within 9 minutes after the MSIV is closed for action from outside main control room	Within the following times after the MSIV is closed: 1. For actions from inside the main control room: ~~2.0~~2.4 minutes	2 minutes [Talk-through]	YES
			2. For actions from outside the main control room: ~~7.0~~10 minutes	6 minutes [Talk-through] & [Walk-through]	YES
	4. Initiate RCS depressuriztion (open pressurizer PORV) after completion of the cooldown	Within 2.5 minutes after reaching the end of cooldown target temperature	Within the following times after reaching the end of cooldown target temperature: 4.0 minutes	1 minute [Talk-through]	YES
	5. Terminate SI (isolate the high head safety injection flow path) after completion of RCS depressurization	Within 1.25 minutes after reaching the end of RCS depressurization target pressure	Within the following times after reaching the end of RCS depressurization target pressure: 3.0 minutes	2 minutes [Talk-through]	YES

Attachment 3 of L-06-157

Plans for Updating the Steam Generator Tube Rupture (SGTR) Analysis For BVPS-1

The BVPS-1 Atmospheric Steam Dump Valves (ASDV) [PCV-1MS-101A, B, & C] and Residual Heat Removal Valve (RHR) [HCV-1MS-104] capacities used in Extended Power Uprate (EPU) analysis for BVPS-1 did not include the effect of the upstream piping friction. The actual valve capacities are less when such friction is considered.

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 cooldown, or to provide heat removal following accident conditions. A secondary function of the valves is to isolate steam release from a steam generator. This function is required for normal plant operation to maintain system integrity and is important to minimize offsite doses in the event of a steam generator tube rupture (SGTR). The SGTR analysis is being re-assessed to determine the impact of the reduced steam release capacity for BVPS-1. This issue is being addressed in accordance with RIS 2005-20 "Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, Information to Licensees regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability". As such, the resolution of this issue is being addressed within the Corrective Action Program.

The BVPS-1 SGTR re-analysis will be completed to confirm the current dose analysis results remain acceptable. The re-analysis will be completed by March 30, 2007. If changes are identified to previously docketed information relative to EPU as a result of this effort, FENOC will provide the necessary information within 60 days of completion of the BVPS-1 SGTR re-analysis. These actions are being tracked as commitments and are identified in Attachment 4 of this submittal.

Attachment 4 of L-06-157

Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by FENOC. They are described only as information and are not regulatory commitments. Please notify Mr. Gregory A. Dunn, Manager - Licensing, at 330-315-7243 of any questions regarding this document or associated regulatory commitments.

Commitments	Due Date
1. The BVPS-1 SGTR re-analysis will be completed to confirm the current dose analysis results remain acceptable.	March 30, 2007
2. If changes are identified to previously docketed information relative to EPU as a result of this effort, FENOC will provide the necessary information.	Within 60 days of completion of the BVPS-1 SGTR re-analysis.