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U. S. Nuclear Regulatory Commission
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Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Supplemental Information Related to License Amendment Request Regarding Measurement Uncertainty Recapture Power Uprate

References:
1. Letter from Craig Lambert (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Measurement Uncertainty Recapture (MUR) Power Uprate," dated June 23, 2011
2. Letter from Kevin F. Borton (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated February 20, 2012 [ML12052A113]

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications to implement an increase in rated thermal power of approximately 1.63% based on increased feedwater flow measurement accuracy.

Subsequent to the June 23, 2011 Measurement Uncertainty Recapture (MUR) Power Uprate license amendment request (LAR) (Reference 1), EGC identified portions of that submittal where revisions to the Steam Generator Tube Rupture / Margin to Overfill (SGTR/MTO) Analysis were necessary. Following flow testing conducted on the Steam Generator (SG) power operated relief valves (PORVs), it was determined that the valves have less flow capacity than was assumed in the SGTR/MTO analysis. As a result of the lower flow capacity, the margin to overfill could be reduced.

For Unit 1 at Braidwood and Byron Stations, the minimum flow of 188 lbm/sec/valve assumed in Section II, “Margin to Steam Generator Overfill Analyses,” of the SGTR/MTO analysis (Reference 1, Attachment 5a, Section II.2.C, “Plant Input”) could not be achieved with the proposed trim modification (Reference 1, Attachment 5a, Section II.2.F, “Modifications to

Support MTO Single Failure Considerations"). The Unit 1 SGTR/MTO was reanalyzed with the lower SG PORV capacity value of 144.6 lbm/sec/valve based on the information obtained during the valve flow testing. As a result of the lower flow capacity, the margin to overfill for Unit 1 at Braidwood and Byron Stations has been reduced from the calculated 94 ft^3 provided in the previous Margin to Steam Generator Overfill Analyses (Reference 1, Attachment 5a, Section II.5, Results") to 54 ft^3. The maximum SG PORV capacity will be less than the 212.4 lbm/sec/valve assumed in the thermal-hydraulic analysis for radiological consequences portion of the SGTR/MTO analysis (Reference 1, Attachment 5a, Section III.2.D, "Plant Input"), therefore the mass and resultant radiological dose consequences for Braidwood and Byron Stations, Unit 1, documented in the MUR LAR SGTR/MTO Analysis (Reference 1, Attachment 5a, Sections III.5. B, "Mass Release Results" and Section IV.4, "Results") remain bounding.

For Unit 2 at Byron and Braidwood Stations, in order to meet the required minimum SG PORVs flow of 114.32 lbm/sec/valve assumed in the SGTR/MTO analysis (Reference 1, Attachment 5a, Section II.2.C, "Plant Input"), EGC will be replacing the trim on the Unit 2 SG PORVs, similar to the Unit 1 SG PORVs, to restore their capacity to the assumed value. Since replacing the trim will increase their overall capacity, a mechanical block will be installed to limit their full open capacity to less than or equal to the maximum value (133.89 lbm/sec/valve) assumed in the thermal-hydraulic analysis for the radiological consequences portion of the SGTR/MTO Analyses (Reference 1, Attachment 5a, Sections III.2. D, "Plant Input") for Braidwood and Byron Stations, Unit 2. These modifications to the Unit 2 SG PORVs will be installed in accordance with 10 CFR 50.59, "Changes, tests and experiments," and are encompassed by the commitment provided in Reference 1, Attachment 4, "Summary of Regulatory Commitments," to implement modifications to support the MTO Single Failure considerations prior to increasing power above the current licensed thermal power (3586.6 MWt).

These revisions result in changes to the statements made in the original submittal (Reference 1), Attachments 5 and 7, "Measurement Uncertainty Recapture Technical Evaluation," Proprietary and Non-Proprietary, respectively, and Attachment 5a, "Steam Generator Tube Rupture and Margin to Overfill Analysis Report." The revised pages are marked and included in Attachments 1 through 3, respectively. The information provided in Attachments 1 through 3, are non-proprietary. Two administrative changes are also included in Attachment 3; wording in Section II.2.C, "Plant Input," has been revised to correctly identify that the minimal purge volume modeling is done to "expedite" the delivery of Auxiliary Feedwater to "maximize" the steam release and Figure II-12, "SG Steam Releases – Unit 2 Margin to Overfill Analysis" has been revised to address a plotting error.

Due to the reduced SG PORV capacity on Braidwood and Byron Stations, Unit 1, EGC conducted a sensitivity study to validate the conclusion that the lower ANS 1979-2σ decay heat factor and minimum auxiliary feedwater enthalpy remain limiting with respect to margin to overfill. Attachment 4, Table A4-1, "Byron/Braidwood Unit 1: Results of Sensitivity Study on MTO With Reduced PORV Flow" provide updated values corresponding to the 12 cases provided in Reference 2, Table SBPB R5-1, "Byron/Braidwood Unit 1, Results of Sensitivity Study on MTO." The results of this sensitivity study validate that the ANS 1979-2σ decay heat factor and minimum auxiliary feedwater enthalpy remain limiting.

An additional, sensitivity study was conducted to determine if maximum safety injection (SI) enthalpy remained limiting. The results indicate that with the reduced SG PORV capacity, minimum SI enthalpy now results in the most limiting conditions. The minimum SI enthalpy cases, (i.e., Cases 5m through 8m on Table A4-1), have either equal to or less margin to overfill than the corresponding maximum SI enthalpy cases (i.e., Cases 1m through 4m). Therefore,

minimum SI enthalpy is more limiting than the previous cases provided where the maximum SI enthalpy was assumed.

EGC has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in Reference 1. The information provided in this supplement letter does not affect the previously stated bases in Reference 1 for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Leslie E. Holden at (630) 657-3316.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 8th day of August 2012.

Respectfully,

Kevin F. Borton
Manager, Licensing - Power Uprate

Attachments:

Attachment 1: Revised pages to MUR Power Uprate LAR, Attachment 5, "Measurement Uncertainty Recapture Technical Evaluation"

Attachment 2: Revised pages to MUR Power Uprate LAR, Attachment 7, "Measurement Uncertainty Recapture Technical Evaluation"

Attachment 3: Revised pages to MUR Power Uprate LAR, Attachment 5a, "Steam Generator Tube Rupture Analysis Report"

Attachment 4: Revised pages to MUR Power Uprate Response to NRC Request for Additional Information, dated February 20, 2012 [ML12052A113]

Braidwood and Byron Stations

Supplemental Information Related to License Amendment Request Regarding Measurement Uncertainty Recapture Power Uprate

August 8, 2012

ATTACHMENT 1

Revised Pages to MUR Power Uprate LAR, Attachment 5, "Measurement Uncertainty Recapture Technical Evaluation"

III.13 Steam Generator Tube Rupture - UFSAR 15.6.3

A reanalysis of the Steam Generator Tube Rupture (SGTR) event was performed as the Margin to Overfill (MTO) results in the current analysis of record (AOR) are unacceptably small and revisions to the analysis assumptions are required. A detailed discussion of the SGTR and MTO Analysis is provided in Attachment 5a, "Steam Generator Tube Rupture and Margin to Overfill Analysis Report." A summary of the revised analysis is provided below.

The analysis addressed three major areas:

1. SGTR Margin to Steam Generator Overfill,
2. SGTR Thermal and Hydraulic Analysis for Radiological Consequences, and
3. SGTR Radiological Consequences

III.13.1 Margin to Steam Generator Overfill

III.13.1.1 Margin to Steam Generator Overfill Analysis

Analyses were performed to determine the margin to SG overfill for a design basis SGTR event for the Byron and Braidwood units. The SGTR MTO accident analysis demonstrated that SG overfill does not occur.

The analyses were performed using the LOFTTR2 program and the methodology developed in Reference III.13-1, with modifications to address NSAL-07-11 (Reference III.13-2) consistent with WCAP 16948 P (Reference III.13-3), and using plant-specific parameters. The MTO analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt. Therefore, the analyzed RTP power bounds the MUR power uprate conditions.

III.13.1.2 MTO Analysis Single Failure Assumptions

A single failure analysis was conducted for the SGTR MTO event to determine the most limiting single failure. This analysis is summarized in Attachment 5a, Sections I.1.E and II.2.E. It was determined that the most limiting failure regarding SG MTO was the failure of an intact SG PORV to open. It should be noted that the assumptions in this scenario necessitated installation of the plant modifications discussed below.

III.13.1.3 MTO Modifications

Byron and Braidwood Stations will be implementing the following plant modifications to support the Steam Generator Margin to Overfill Reanalysis single failure assumptions:

1. Install safety related air accumulator tanks to support AFW flow control,
2. Increase the capacity of the SG Power Operated Relief Valves (PORVs) (Unit 1 only)
3. Modify SG PORV to achieve analysis flow rates (Unit 2 only),

4. Install Uninterruptible Power Supplies (UPS) on two of the four SG PORVs
5. Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

A description of these modifications is provided in Attachment 5a, Section II.2.F, "Modifications to Support MTO Single Failure Considerations." As noted above, these modifications will be installed and made operational prior to increasing power above the current licensed power level. The safety-related air accumulator tanks for AFW valve flow control, the modification to the Unit 2 SG PORVs, the UPS to the PORVs and the manual SI isolation valve are planned to be installed in accordance with 10 CFR 50.59; however, installation of the modification to increase the Unit 1 SG PORVs flow capacity requires NRC approval prior to installation as this modification results in more than a minimal increase in the accident dose.

The modification to install uninterruptible power supplies to the SG PORVs is prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a SGTR event. The NRC documented their position regarding these URIs in Reference III.13-4. The NRC verified that this same SGTR-related concern was also applicable to Braidwood Station as documented in Reference III.13-6. Byron Station responded to the NRC in Reference III.13-5; and Braidwood Station responded to the NRC in Reference III.13-7. In these letters, both Byron Station and Braidwood Station committed to installing the UPS modification to resolve the single failure concern. This modification places the SGTR analysis in compliance with NRC regulations and preserves the assumption in the SGTR analysis.

III.13.2 Thermal and Hydraulic Analysis for Radiological Consequences

The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in References III.13-1 and III.13-8, and using the plant-specific parameters. From these predictions, the RCS and SG water masses, the ruptured SG break flow, the fraction of this break flow that flashes directly to steam, and the steam releases from the ruptured and intact SGs through the MSSVs and PORVs are calculated for input to the dose analyses. The thermal-hydraulic analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt to generate this data. Therefore, the analyzed power bounds the MUR power uprate.

III.13.3 Radiological Consequences Analysis

The steam generator tube rupture radiological analyses are based upon the alternative source term (AST) as defined in Regulatory Guide (RG) 1.183, with acceptance criteria as specified in RG 1.183 for offsite doses and in 10 CFR 50.67 for the control room. The analyses involve the transfer of activity from the primary to the secondary side of the SGs and then to the environment. The RCS iodine and noble gas source terms are scaled to the Technical Specification Dose Equivalent Iodine-131 and Xenon-133 limits in the primary coolant, which removes the power dependence from the analysis. The various parameters from the thermal-hydraulic analyses are consistent with a core power of 3658.3 MWt, or 102% of 3586.6 MWt. The resulting doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and in the control room remain within the applicable limits as shown in Attachment 5a, Table IV-6; therefore, the results of the SGTR radiological analyses are acceptable under MUR power uprate conditions.

Braidwood and Byron Stations

Supplemental Information Related to License Amendment Request Regarding Measurement Uncertainty Recapture Power Uprate

August 8, 2012

ATTACHMENT 2

Revised Pages to MUR Power Uprate LAR, Attachment 7, “Measurement Uncertainty Recapture Technical Evaluation”

III.13 Steam Generator Tube Rupture - UFSAR 15.6.3

A reanalysis of the Steam Generator Tube Rupture (SGTR) event was performed as the Margin to Overfill (MTO) results in the current analysis of record (AOR) are unacceptably small and revisions to the analysis assumptions are required. A detailed discussion of the SGTR and MTO Analysis is provided in Attachment 5a, "Steam Generator Tube Rupture and Margin to Overfill Analysis Report." A summary of the revised analysis is provided below.

The analysis addressed three major areas:

1. SGTR Margin to Steam Generator Overfill,
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III.13.1 Margin to Steam Generator Overfill

III.13.1.1 Margin to Steam Generator Overfill Analysis

Analyses were performed to determine the margin to SG overfill for a design basis SGTR event for the Byron and Braidwood units. The SGTR MTO accident analysis demonstrated that SG overfill does not occur.

The analyses were performed using the LOFTTR2 program and the methodology developed in Reference III.13-1, with modifications to address NSAL-07-11 (Reference III.13-2) consistent with WCAP-16948-P (Reference III.13-3), and using plant-specific parameters. The MTO analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt. Therefore, the analyzed RTP power bounds the MUR power uprate conditions.

III.13.1.2 MTO Analysis Single Failure Assumptions

A single failure analysis was conducted for the SGTR MTO event to determine the most limiting single failure. This analysis is summarized in Attachment 5a, Sections I.1.E and II.2.E. It was determined that the most limiting failure regarding SG MTO was the failure of an intact SG PORV to open. It should be noted that the assumptions in this scenario necessitated installation of the plant modifications discussed below.

III.13.1.3 MTO Modifications

Byron and Braidwood Stations will be implementing the following plant modifications to support the Steam Generator Margin to Overfill Reanalysis single failure assumptions:

1. Install safety related air accumulator tanks to support AFW flow control,
2. Increase the capacity of the SG Power Operated Relief Valves (PORVs) (Unit 1 only)
3. Modify SG PORV to achieve analysis flow rates (Unit 2 only),

4. Install Uninterruptible Power Supplies (UPS) on two of the four SG PORVs
5. Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

A description of these modifications is provided in Attachment 5a, Section II.2.F, "Modifications to Support MTO Single Failure Considerations." As noted above, these modifications will be installed and made operational prior to increasing power above the current licensed power level. The safety-related air accumulator tanks for AFW valve flow control, the modification to the Unit 2 SG PORVs, the UPS to the PORVs and the manual SI isolation valve are planned to be installed in accordance with 10 CFR 50.59; however, installation of the modification to increase the Unit 1 SG PORVs flow capacity requires NRC approval prior to installation as this modification results in more than a minimal increase in the accident dose.

The modification to install uninterruptible power supplies to the SG PORVs is prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a SGTR event. The NRC documented their position regarding these URIs in Reference III.13-4. The NRC verified that this same SGTR-related concern was also applicable to Braidwood Station as documented in Reference III.13-6. Byron Station responded to the NRC in Reference III.13-5; and Braidwood Station responded to the NRC in Reference III.13-7. In these letters, both Byron Station and Braidwood Station committed to installing the UPS modification to resolve the single failure concern. This modification places the SGTR analysis in compliance with NRC regulations and preserves the assumption in the SGTR analysis.

III.13.2 Thermal and Hydraulic Analysis for Radiological Consequences

The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in References III.13-1 and III.13-8, and using the plant-specific parameters. From these predictions, the RCS and SG water masses, the ruptured SG break flow, the fraction of this break flow that flashes directly to steam, and the steam releases from the ruptured and intact SGs through the MSSVs and PORVs are calculated for input to the dose analyses. The thermal-hydraulic analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt to generate this data. Therefore, the analyzed power bounds the MUR power uprate.

III.13.3 Radiological Consequences Analysis

The steam generator tube rupture radiological analyses are based upon the alternative source term (AST) as defined in Regulatory Guide (RG) 1.183, with acceptance criteria as specified in RG 1.183 for offsite doses and in 10 CFR 50.67 for the control room. The analyses involve the transfer of activity from the primary to the secondary side of the SGs and then to the environment. The RCS iodine and noble gas source terms are scaled to the Technical Specification Dose Equivalent Iodine-131 and Xenon-133 limits in the primary coolant, which removes the power dependence from the analysis. The various parameters from the thermal-hydraulic analyses are consistent with a core power of 3658.3 MWt, or 102% of 3586.6 MWt. The resulting doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and in the control room remain within the applicable limits as shown in Attachment 5a, Table IV-6; therefore, the results of the SGTR radiological analyses are acceptable under MUR power uprate conditions.

Braidwood and Byron Stations

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ATTACHMENT 3

Revised Pages to MUR Power Uprate LAR, Attachment 5a, “Braidwood and Byron Stations Steam Generator Tube Rupture Analysis Report”

rates the SGs would be removing different amounts of energy from the primary system, resulting in different initial secondary fluid mass calculations for the different loops.

The flow asymmetry was modeled in the analyses. From Reference 5 the maximum loop-to-loop flow asymmetry for a "newer" plant is 7% of nominal loop flow. The corresponding loop-to-loop power (SG heat flux) asymmetry was determined and used in steady state secondary side mass calculations to determine the conservative initial secondary side fluid mass.

2. SG Secondary Mass

A higher initial secondary water mass in the ruptured SG was determined by Reference 1 to be conservative for overfill. The calculation of the initial secondary side water mass conservatively modeled an RCS flow asymmetry of 7%, as discussed above. This was applied in addition to the conservative increase in initial mass applied based on Reference 1 to model the increase in mass that would result from a turbine runback to a lower power, and the consideration of mass uncertainties. The runback is assumed to start at event initiation and continue until the time that the reactor trip setpoint is reached. The power resulting from the turbine runback is determined using an average runback rate of 10% power per minute, with the total power reduction limited to 30% (i.e., 3 minutes of runback).

3. Decay Heat and NSAL-07-11

NSAL-07-11 (Reference 3) identifies a potential non-conservative assumption regarding the direction of conservatism for decay heat in the WCAP-10698-P-A (Reference 1) methodology for evaluating margin to overfill. For the margin to overfill analyses, higher decay heat yields a benefit by increasing steam releases from the ruptured SG, but results in a penalty from a longer cooldown and a conservatively delayed break flow termination. Conversely, lower decay heat yields a penalty by reducing steam releases from the ruptured SG, but results in a benefit from a shorter cooldown and earlier break flow termination. Similar impacts were identified in WCAP-16948-P (Reference 4) for the AFW and safety injection (SI) flow enthalpies. The relative importance of these competing effects is plant-specific, and plant-specific analyses are required to determine the conservative assumption. Plant-specific sensitivities performed for Byron and Braidwood Units 1 and 2 showed the following to be conservative with respect to margin to overfill for the limiting cases:

- 1979-2σ American Nuclear Society (ANS) decay heat was conservative compared to the 1971+20% ANS decay heat model specified in WCAP-10698-P-A (Reference 1). For these analyses, the 1979 ANS decay heat model minus 2σ uncertainty was used.

- Minimum AFW enthalpy was conservative compared to the maximum AFW enthalpy specified by WCAP-10698-P-A (Reference 1). For these analyses, the minimum AFW enthalpy of 0.03 Btu/lbm was modeled.

- For Unit 1, minimum SI enthalpy was conservative compared to the maximum SI enthalpy modeled in WCAP-10698-P-A (Reference 1) for some cases analyzed, although the difference was minimal. For other cases, the competing impacts of modeling maximum and minimum SI enthalpy have no net effect on the calculated margin to overfill. For the limiting analysis presented, the minimum SI enthalpy of 3 Btu/lbm was modeled.

- For Unit 2, maximum SI enthalpy was conservative consistent with WCAP-10698-P-A (Reference 1). For these analyses, the maximum SI enthalpy of 80 Btu/lbm was modeled.

The analyses also incorporated the conclusion of WCAP-16948-P (Reference 4) that beginning of life (BOL) minimum reactivity feedback coefficients are conservative for margin to overfill analyses.

II.C.2 Plant Input

The following significant plant-specific input was used in the analyses.

1. SG Dimensions

   The SGTR flow is a function of the SG tube inside diameter. The SG tube inside diameters modeled in the analyses are:

   - Unit 1: 0.608 inches.
   - Unit 2: 0.664 inches.

   The available secondary side volume was used to calculate the margin to overfill. The available secondary side volumes modeled in the analyses are:

   - Unit 1: 5122 ft^3
   - Unit 2: 5955 ft^3

2. SG Power-Operated Relief Valve (PORV)

   It was assumed that a loss of offsite power occurs at reactor trip for the SGTR analyses, and thus the SG PORVs open to limit the secondary pressure. The PORV pressure setpoint is 1129.7 psia. The PORVs are relied upon to cool the RCS. A low value for the capacity of the PORVs was modeled since this results in a slower cooldown. The capacity modeled was not based solely on the nominal size of the valves, and addressed the concern identified in TB-07-6 (Reference 6). The PORV capacities modeled in the analyses are:

   - Unit 1: 144.6 lbm/sec/valve @ 1190 psia (with planned modified valve trim)
   - Unit 2: 114.32 lbm/sec/valve @ 1190 psia (with planned modified valve trim).

3. Pressurizer PORV Capacity

   It was assumed that a loss of offsite power occurs at reactor trip for the SGTR analyses, and thus the pressurizer PORV was relied upon to depressurize the RCS. The capacity of 210,000 lbm/hr at 2350 psia was used in the analyses for both units.

4. Auxiliary Feedwater

   It was assumed that the maximum AFW flow was delivered to the SGs following reactor trip and loss of offsite power with no delay. A minimal purge volume (1 ft^3) was modeled to expedite delivery of cold AFW to the SGs and minimize steam release. The following AFW flows are modeled in the analyses:

   - Unit 1: 180 gpm/SG for the first 40 seconds, 360 gpm/SG after 40 seconds.
   - Unit 2: 263 gpm/SG for the first 40 seconds, 450.22 gpm/SG after 40 seconds.

2. Failure of ruptured SG MSIV

   This scenario considered the failure of the MSIV on the ruptured SG to close when the operators isolated the ruptured SG. In this case the operators were required to isolate the MSIVs on the intact SGs prior to initiating the cooldown. This was assumed to delay initiation of the cooldown by an additional 2 minutes (i.e., 5 minutes elapsed from the attempt to close the ruptured MSIV as part of ruptured SG isolation until the cooldown is initiated). This delay resulted in increased break flow. The cooldown was then performed using all three of the PORVs on the intact SGs.

3. Failure of ruptured SG feedwater control valve (FWCV)

   This scenario considered the failure of the FWCV on the ruptured SG to automatically reduce main feedwater flow to offset the addition of break flow to the ruptured steam generator. Thus, the mass in the ruptured SG increased in relation to the intact SGs prior to reactor trip. While this additional mass would be expected to provide early identification and isolation of AFW flow to the ruptured SG following reactor trip, no reduction in the operator action time for AFW isolation was credited. The initial secondary SG water mass was not increased to account for the impact of turbine runback. This modeling is consistent with the FWCV failure presented in WCAP-10698-P-A (Reference 1). The cooldown was performed using all three of the PORVs on the intact SGs.

The intact SG PORV failure was determined to be the limiting single failure. The penalty from the delay in cooldown initiation that resulted from the MSIV failure was offset by the faster cooldown obtained by use of the PORVs on three intact SGs. The impact of continued steam leakage of 5 lbm/sec for 30 minutes from the ruptured SG via unisolated flow paths was investigated for the MSIV failure case and it was determined to result in increased margin to overfill since the mass released was greater than the additional break flow that resulted from the small reduction in the ruptured SG pressure. Compared to the intact SG PORV failure case, there was a benefit in the FWCV failure case since the initial ruptured SG mass penalty for turbine runback was not included in the FWCV failure scenario. The FWCV failure case also received a benefit of a shorter cooldown due to the availability of the third intact SG PORV for the cooldown.

In the determination of the limiting failure the margin to overfill at the time of SI termination for the different sensitivity runs was compared. The operator responses after SI termination are not dependent on the specific failure scenario and the time for the operators to take positive control for termination of break flow would be consistent between the different single failure scenarios.

II.2.F Modifications to Support MTO Single Failure Considerations

Byron and Braidwood Stations will be implementing plant modifications to support the Steam Generator Margin to Overfill Reanalysis assumptions. The four modifications are as follows:

- Install safety related air accumulator tanks to support AFW flow control
- Increase the capacity of the SG Power Operated Relief Valves (PORV's) (on Unit 1 only)
- Modify SG PORV to achieve analysis flow rates (Unit 2 only)
- Install Uninterruptible Power Supplies (UPS) on 2 of the 4 SG PORVs

- Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

Below is a brief description of each modification:

1. Install Safety Related Air Accumulator Tanks to Support Auxiliary Feedwater (AF) Flow Control

   This modification will install two instrument air accumulator tanks (one per train on each unit) to provide a safety related air supply for the Auxiliary Feedwater Flow Control Valves. The tanks will be capable of providing 30 minutes of air supply to the Auxiliary Feedwater Flow Control Valves (AF005's). The tanks will be safety related. In addition, the modification will install two check valves in series for each tank to separate the non safety portion of the instrument air system from the safety related air accumulator tanks and tubing. A relief valve will be installed on each tank to provide overpressure protection. The electronic controls associated with the flow control loop have been verified to be safety related. This modification is planned to be installed in accordance with 10 CFR 50.59.

2. Increase the Capacity of the Steam Generator Power Operated Relief Valves (PORVs) on Unit 1 Only

   Byron and Braidwood will be replacing the SG PORV valve trim to increase the capacity of the valve from the previously estimated value of approximately 420,000 lbs/hr to approximately 540,000 lbs/hr. By increasing the valve capacity the operators can cool down and depressurize the Reactor Coolant System (RCS) more rapidly which will equalize the pressure between the RCS and the secondary side of the steam generator which terminates the flow from the RCS to the secondary. This reduces the inflow to the secondary and therefore increases the margin to overfill. NRC approval of this modification, as it is included in the SGTR reanalysis, is required prior to installation as this change results in more than a minimal increase in the accident dose as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000.

3. Modify SG PORV to achieve analysis flow rates (Unit 2 only)

   In order to meet the required minimum SG PORVs flow assumed in Section II.C, Byron and Braidwood Stations will be replacing their trim on the Unit 2 SG PORVs, similar to that described above for the Unit 1 SG PORVs. To limit the SG PORV capacity to the assumed maximum value a mechanical stop will be installed.

4. Install Uninterruptible Power Supplies (UPS) on two of the four Steam Generator (SG) Power Operated Relief Valves

   The modification to install uninterruptible power supplies to the SG PORVs is prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a Steam Generator Tube Rupture (SGTR) event. The NRC documented their position regarding these URIs in Reference 14. The NRC verified that this same SGTR-related concern was also applicable to Braidwood Station as documented in Reference 16. Byron Station responded to the NRC in Reference 15; and Braidwood Station responded to the NRC in Reference 17. In these letters, both Byron Station and Braidwood Station committed to installing the UPS modification to resolve the single failure concern. This modification places the SGTR analysis in compliance with NRC regulations and preserves the assumptions in the SGTR analysis.

The modification will install two UPS units on each Unit at Byron and Braidwood Stations. At Byron and Braidwood Stations, there are four SG PORV's; two PORVs power from electrical Division 1, and the other two PORVs powered from electrical Division 2. One UPS will be installed on each electrical division's power supply to one of the SG PORVs it supplies. Currently, a potential single failure of the Division 1 (or 2) 480VAC Unit Substation exists that would disable the power supplies to two SG PORV's. Following the implementation of this modification, the normal power feed will continue to be a Motor Control Center (MCC) powered from a 480VAC Unit Substation; however, on a loss of power to the 480VAC Unit Substation, the UPS would provide a backup power supply to one of the two PORVs from that division. By installing one UPS to a PORV powered from each electrical division, a single failure of a 480VAC Unit Substation would only adversely impact one SG PORV. The other division would still have power supplied to both SG PORV's. Therefore, at least two SG PORV's will be available to support cool down and depressurization of the RCS during a SGTR event (this also assumes that the ruptured SG PORV is isolated and unavailable for cooldown). This modification is planned be installed in accordance with 10 CFR 50.59.

5. Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

As mentioned in Section I.1.E, Byron and Braidwood do not have BIT isolation valves, but the two parallel High Head Safety Injection motor operated valves (1/2SI8801A/B) provide the equivalent function. These valves provide the means to isolate the high head safety injection flow during the SI termination phase. If a single failure of one of these valves occurs, it would be necessary to secure the charging pumps in order to depressurize the reactor coolant system (RCS) to stop the break flow into the ruptured steam generator and to prevent overfilling the RCS. If the charging pumps were secured, injection flow to the reactor coolant pumps seals ceases. While reactor coolant pump seal design provides for loss of seal injection, it is not a desirable condition.

To address the potential single failure of an SI8801A/B valve to close, a modification will be implemented which installs a manual valve upstream of each SI8801A/B valve to provide for isolation of high head safety injection without the need to stop all charging pumps. During normal plant conditions these manual valves will be locked in the open position. Upon failure of an SI8801A/B valve to close when demanded, an operator will be dispatched to locally close the upstream manual isolation valve. As described above, this manual action is not replacing an automatic function but is simply an equivalent manual action for locally isolating high head safety injection flow into the RCS. This modification is planned to be installed in accordance with 10 CFR 50.59.

II.3 Description of Analyses

The LOFTTR2 analysis for the limiting Unit 1 and Unit 2 margin to overfill cases are described below. For both units the limiting case with respect to margin to SG overfill considered operation at the minimum operating temperature (580.0°F for Unit 1 and 575.0°F for Unit 2), with the minimum main feedwater temperature (433.0°F for Unit 1 and 435.0°F for Unit 2), the maximum SGTP level (5% for Unit 1 and 10% for Unit 2), and the failure of a PORV on an intact SG to open when the operator performed the RCS cooldown. The sequences of events for these transients are presented in Table II-3.

Although the lower T_{avg} is conservative for margin to overfill, this does not mean that a T_{avg} coastdown will be more limiting. Since the Overtemperature Delta-T (OTΔT) reactor trip nominal T_{avg} and nominal

delta-T values in the setpoint equation are not reset to reflect the lower operating T_{avg}, the reactor trip will be delayed. The benefits of delaying reactor trip versus the disadvantages of operating at a lower T_{avg} were incorporated in a case considered in the analysis. The Unit 2 case modeling the T_{avg} coastdown to 573.5°F showed the same margin to overfill as the case considering the Unit 2 minimum operating temperature of 575.0°F. It was concluded that the low T_{avg} case would be reported as the limiting case but that future analyses would continue to examine the impact of T_{avg} coastdown.

Following the tube rupture, water flowed from the primary into the secondary side of the ruptured SG since the primary pressure is greater than the SG pressure. In response to this loss of coolant, pressurizer level decreased as shown in Figure II-1 (Unit 1) and Figure II-7 (Unit 2). The RCS pressure also decreased as shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2) as the steam bubble in the pressurizer expanded. As the RCS pressure decreased due to the continued primary to secondary break flow automatic reactor trip occurred on an Overtemperature ΔT (OTΔT) trip signal.

After reactor trip, core power rapidly decreased to decay heat levels. The turbine stop valves closed and steam flow to the turbine was terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remained 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 caused the secondary side pressure to increase rapidly after reactor trip, as shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2), until the SG PORVs (and safety valves if their setpoints are reached) lifted to dissipate the energy, as shown in Figure II-6 (Unit 1) and Figure II-12 (Unit 2). As a result of the assumed loss of offsite power, main feedwater flow was assumed to be terminated and AFW flow was assumed to be automatically initiated following reactor trip.

The RCS pressure and pressurizer level continued to decrease after reactor trip as energy transfer to the secondary system shrank the primary coolant and the tube rupture break flow continued to deplete primary inventory. The decrease in RCS inventory resulted in a low pressurizer pressure SI signal. The SI flow increased the RCS inventory and the RCS pressure trended toward the equilibrium value where the SI flow rate would equal the break flow rate.

AFW flow to the ruptured SG was assumed to be isolated 9 minutes after the start of the event, and the ruptured SG MSIV was assumed to be closed at 18 minutes. The ruptured SG level was well above the level required for identification and isolation by these times.

After isolation of the ruptured SG, a 3-minute operator action time was assumed prior to initiating the cooldown. Due to the assumed failure of one of the intact SG PORVs to open only two intact SGs were credited for the cooldown. It was therefore assumed that the PORVs on two intact SGs were opened for the RCS cooldown at 21 minutes after the start of the event, as shown in Figure II-6 (Unit 1) and Figure II-12 (Unit 2). The cooldown was continued until the cooldown termination temperature obtained from EOPs was reached. When this condition was satisfied the operator closed the PORVs to terminate the cooldown. This cooldown ensured that there would be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured SG pressure. The reduction in the intact SG pressure required to accomplish the cooldown is shown in Figure II-1 (Unit 1) and Figure II-8 (Unit 2). The pressurizer level and RCS pressure also decreased during this cooldown process due to shrinkage of the RCS as shown in Figure II-1 and Figure II-2 (Unit 1) and Figure II-7 and Figure II-8 (Unit 2).

The PORVs on the intact SGs which were used for the cooldown also automatically opened as necessary to maintain the prescribed RCS temperature to ensure that subcooling was maintained. When the PORVs were opened, the increased energy transfer from the RCS to the secondary system also aided in the depressurization of the RCS to the ruptured SG pressure after the SI flow was terminated.

After termination of the cooldown, a 4-minute operator action time was assumed prior to the RCS depressurization. In these analyses, the RCS depressurization was terminated when the RCS pressure was reduced to less than the ruptured SG pressure and the pressurizer level was above the required value, since there was adequate subcooling margin and the high pressurizer level setpoint was not reached. The RCS depressurization is shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2). The depressurization reduced the break flow as shown in Figure II-3 (Unit 1) and Figure II-9 (Unit 2) and increased SI flow to refill the pressurizer, as shown in Figure II-1 (Unit 1) and Figure II-7 (Unit 2).

After termination of the depressurization, a 3-minute operator action time was assumed prior to SI termination. The SI flow was terminated at this time since the requirements for SI termination were satisfied. (RCS subcooling was greater than the required allowance for subcooling uncertainty, minimum AFW flow was available or at least one intact SG level was in the narrow range, the RCS pressure was stable or increasing, and the pressurizer level was greater than the required value.) After SI termination the RCS pressure began to decrease as shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2).

II.4 Acceptance Criteria

The analyses were performed to demonstrate that the secondary side of the ruptured SG did not completely fill with water. The available secondary side volume of a single SG is 5122 ft^3 for Unit 1 and 5955 ft^3 for Unit 2. Margin to overfill is demonstrated provided the transient calculated SG secondary side water volume is less than these values. No credit is taken for the volume of the nozzle or any steam piping.

II.5 Results

The primary to secondary break flow rate throughout the recovery operations is presented in Figure II-3 (Unit 1) and Figure II-9 (Unit 2). The ruptured SG fluid mass is shown in Figure II-4 (Unit 1) and Figure II-10 (Unit 2). The water volume in the ruptured SG is presented as a function of time in Figure II-5 (Unit 1) and Figure II-11 (Unit 2). The peak ruptured SG water volume for Unit 1 is 5068 ft^3 resulting in 54 ft^3 of margin to overfill. The peak ruptured SG water volume for Unit 2 is 5685 ft^3 resulting in 270 ft^3 of margin to overfill. Therefore, it is concluded that overfill of the ruptured SG will not occur for a design basis SGTR for Byron and Braidwood Units 1 and 2.

II.6 Conclusions

It is concluded that overfill of the ruptured SG will not occur for a design basis SGTR for Byron and Braidwood Units 1 and 2.

Table II-3 Sequence of Events for Limiting Margin to Overfill Analyses

	Unit 1	Unit 2
Event	**Time (seconds)**	**Time (seconds)**
Steam Generator Tube Rupture	0	0
Reactor Trip (OTΔT) and LOOP	200.4	139.3
AFW Initiated	201	140
SI Actuated	474	317
AFW Flow to Ruptured SG Isolated	540	540
Ruptured SG MSIV Closed	1080	1080
RCS Cooldown Initiated	1260	1260
RCS Cooldown Terminated	1838	1958
RCS Depressurization Initiated	2080	2200
RCS Depressurization Terminated	2178	2302
SI Terminated	2359	2482
Break Flow Terminated	3360	3258

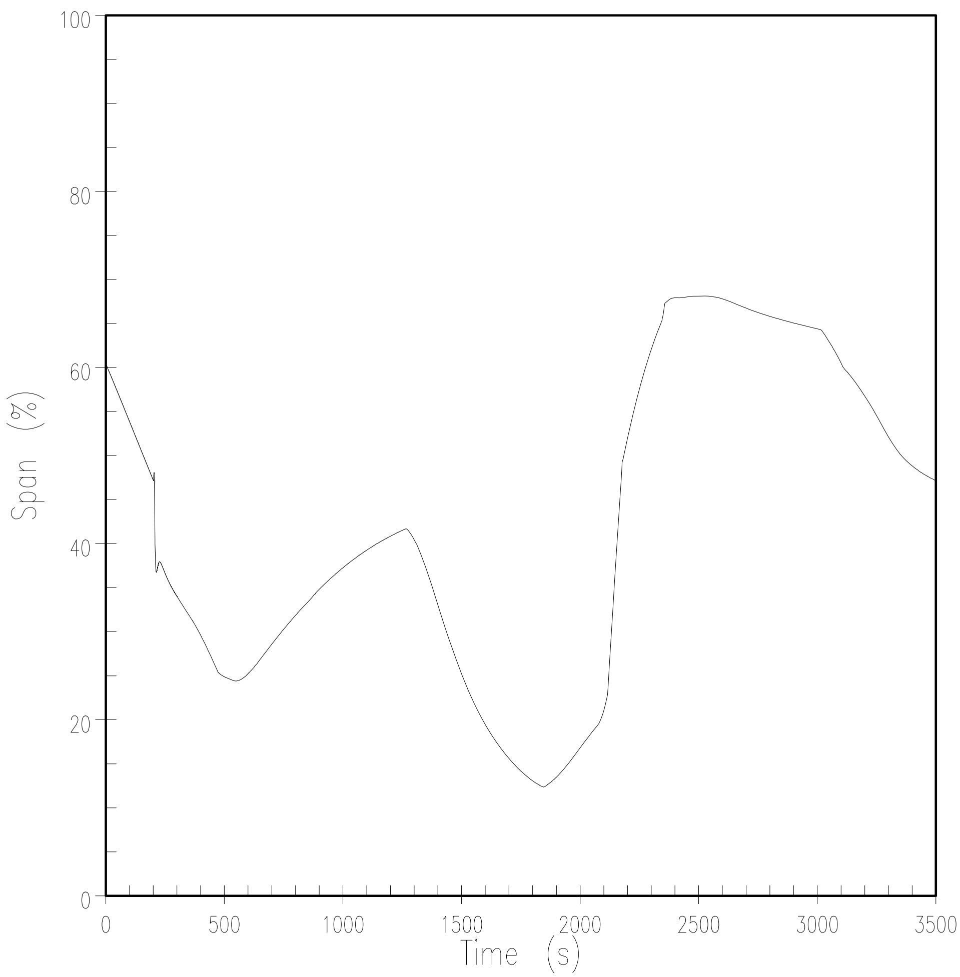

**Figure II-1 Pressurizer Level –
Unit 1 Margin to Overfill Analysis**

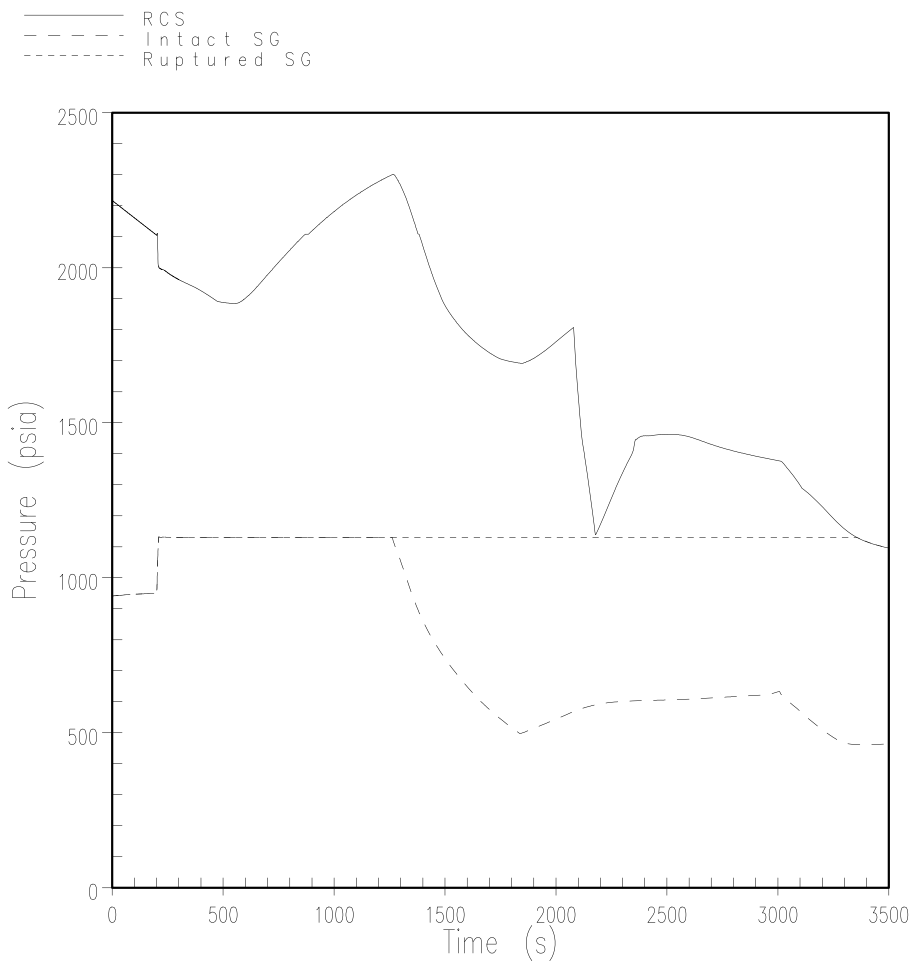

**Figure II-2 RCS and Secondary Pressures –
Unit 1 Margin to Overfill Analysis**

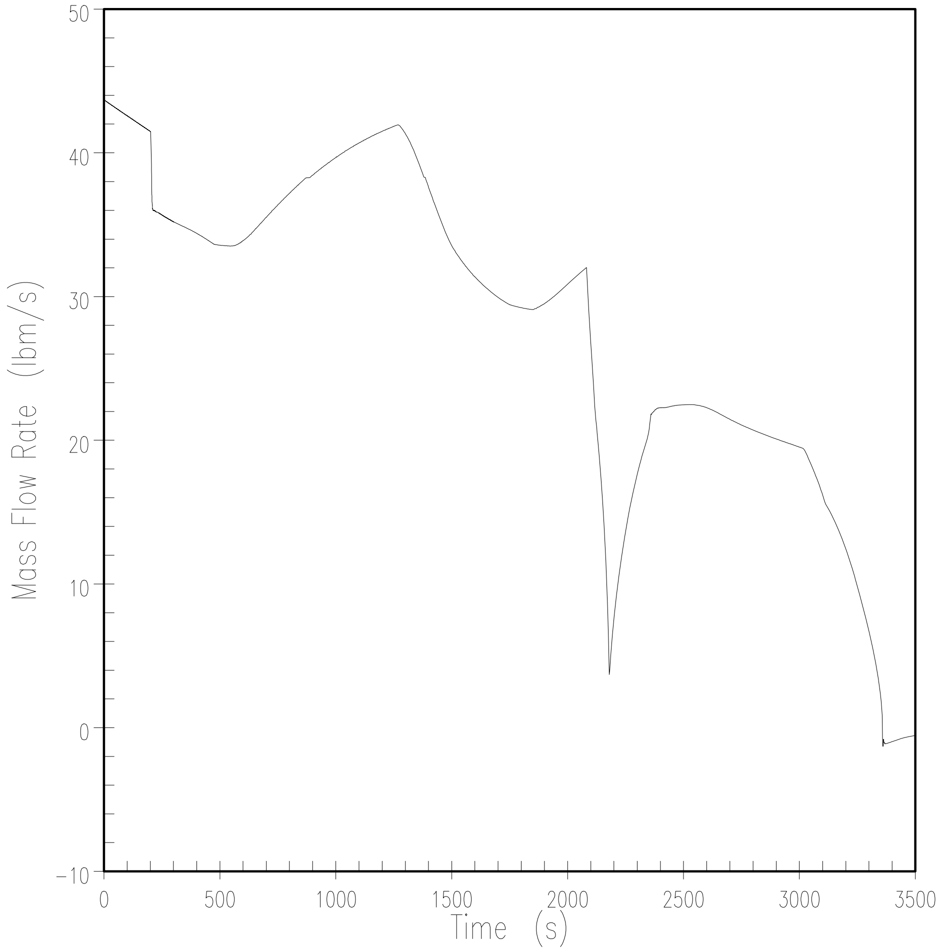

Figure II-3 Primary to Secondary Break Flow –
Unit 1 Margin to Overfill Analysis

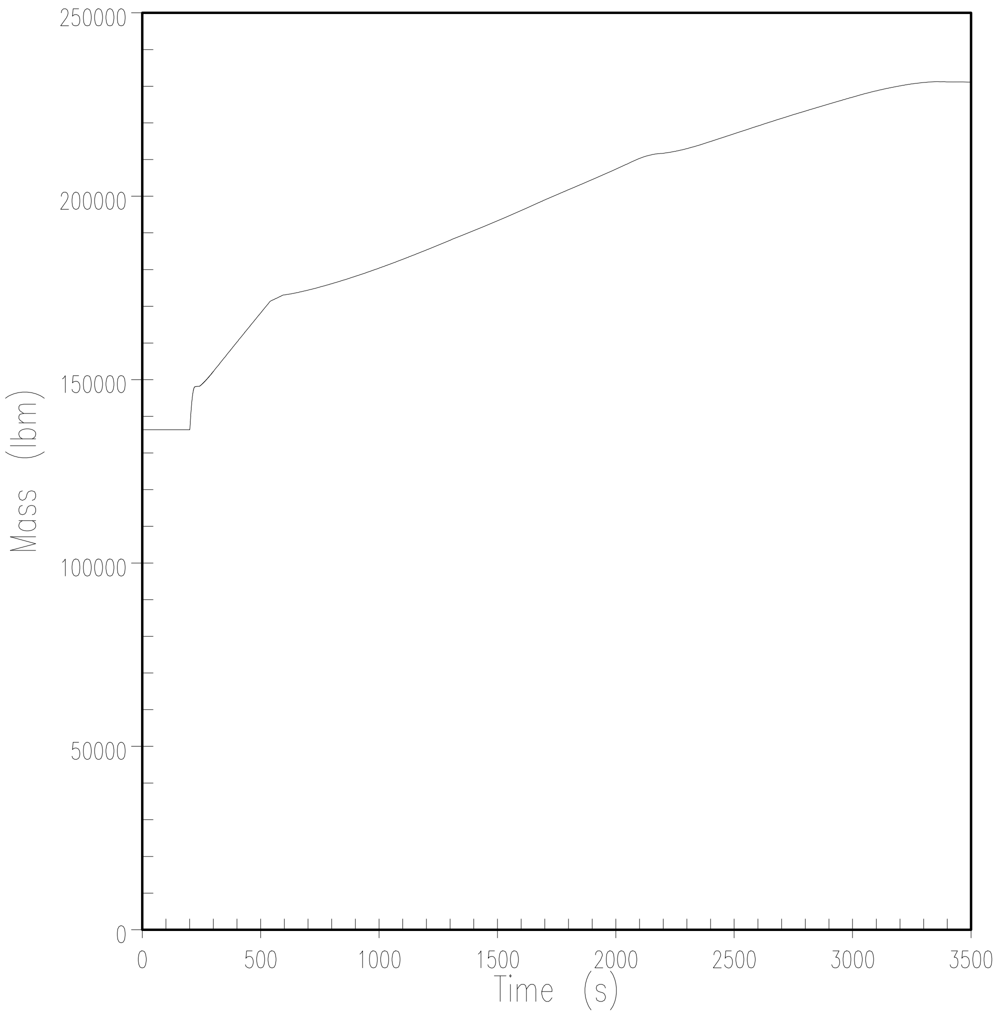

**Figure II-4 Ruptured SG Fluid Mass –
Unit 1 Margin to Overfill Analysis**

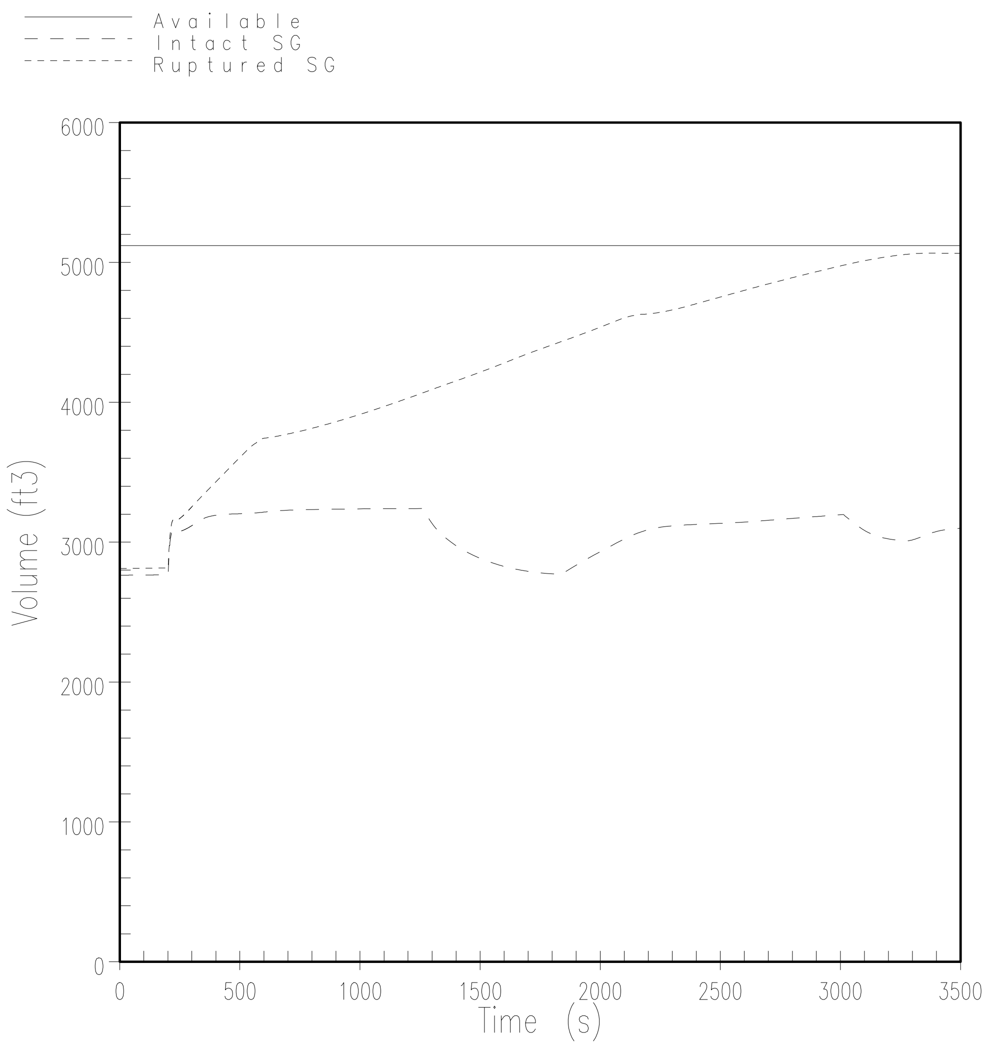

**Figure II-5 SG Water Volumes –
Unit 1 Margin to Overfill Analysis**

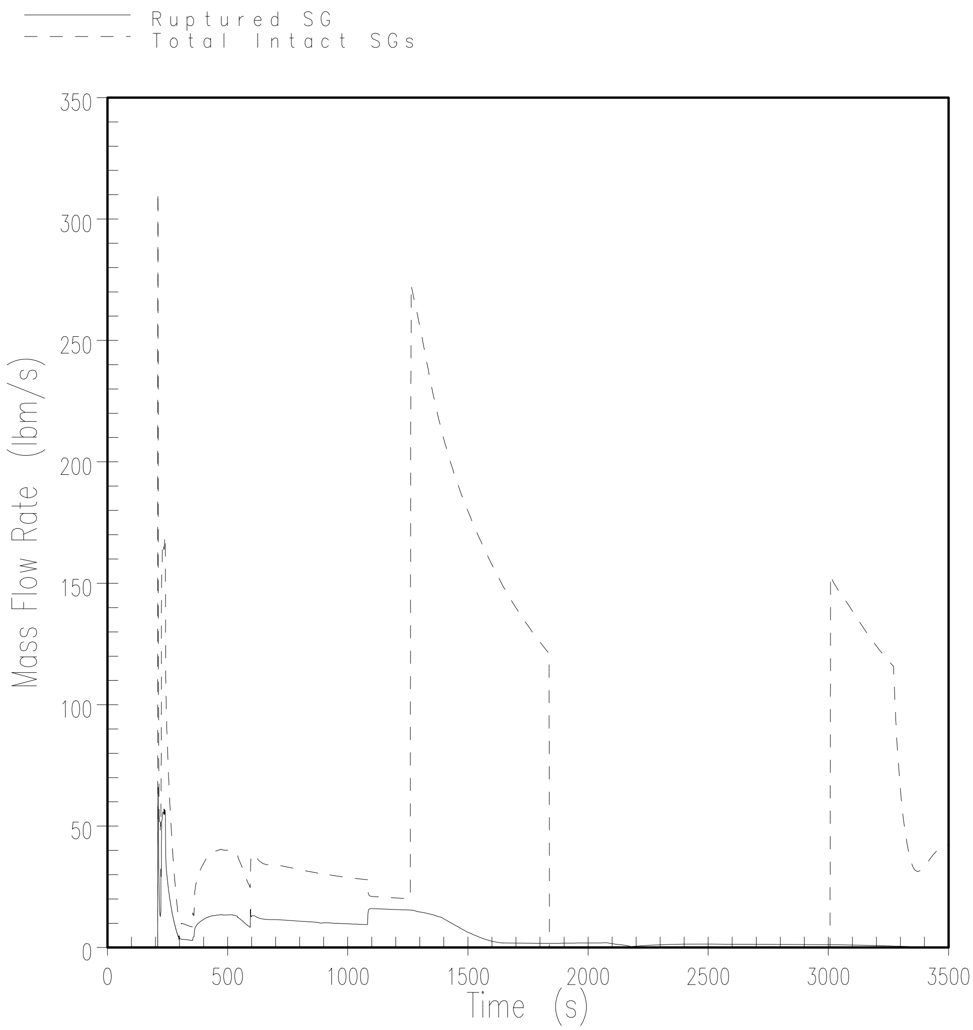

**Figure II-6 SG Steam Releases –
Unit 1 Margin to Overfill Analysis**

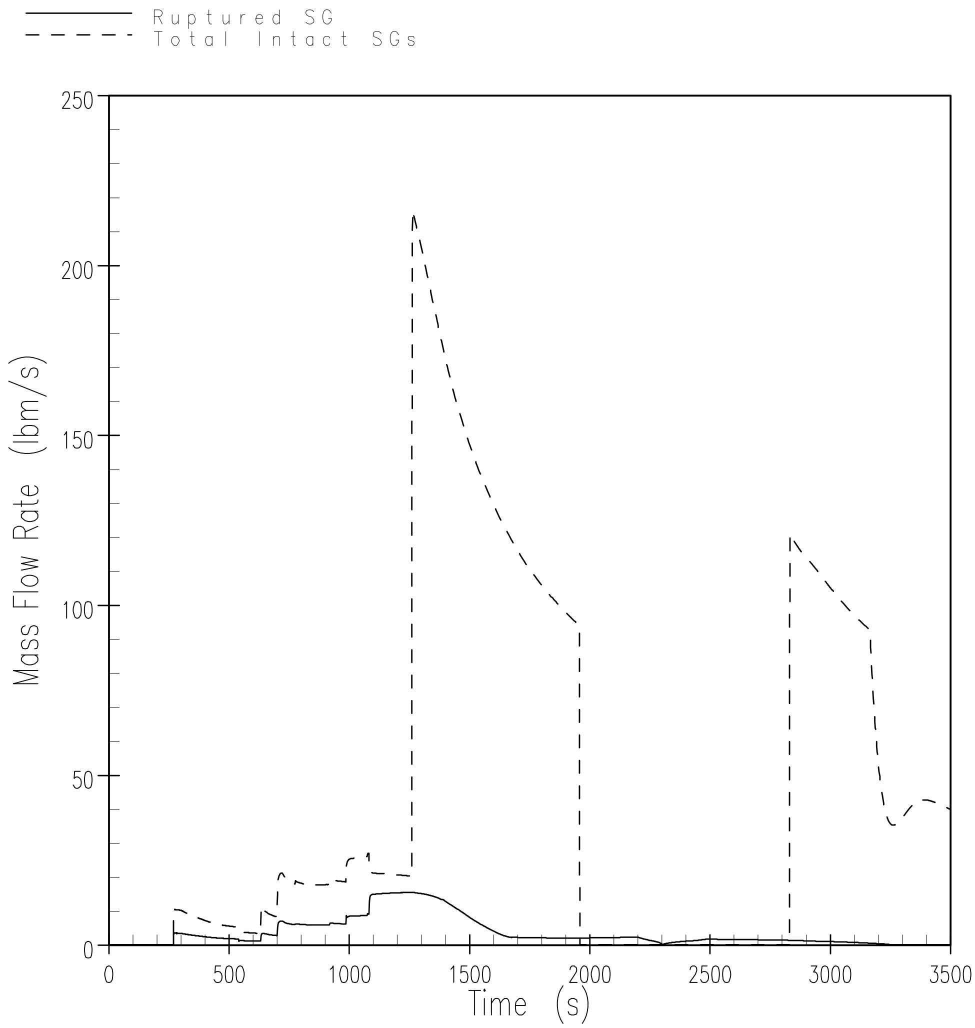

**Figure II-12 SG Steam Releases –
Unit 2 Margin to Overfill Analysis**

Most of the assumptions used for the margin to overfill analyses are also conservative for the radiological consequences analyses. The major differences in the assumptions that were used for the LOFTTR2 analyses for radiological consequences compared to those used in the margin to overfill analyses are discussed below.

1. SG Secondary Mass

   Plant-specific sensitivity studies determined that a maximum initial secondary water mass resulted in increased steam releases and flashed break flow. Therefore, the transient calculation considered the effects of turbine runback and RCS flow asymmetry in the same manner as the margin to overfill analyses. However, a lower secondary mass is conservative for the dose analyses. Additional cases were analyzed with minimum initial secondary water mass (without the turbine runback mass increase and with the maximum main feedwater temperature) and the results conservatively incorporated in the dose analyses. A lower secondary mass also results in a lower ruptured SG pressure when the ruptured SG is failed open and this was considered in the confirmation that the pressure did not fall below 320 psig noted in the operator action time discussion below.

2. Decay Heat and NSAL-07-11

   As noted in NSAL-07-11 (Reference 3) SGTR thermal and hydraulic analyses for input to radiological consequences analyses have no competing effects with respect to decay heat. Higher decay results in increased steam releases from the ruptured SG and a longer cooldown, leading to a later break flow termination. These effects are conservative for the SGTR radiological consequences calculation, and thus, lower decay heat was not considered. Similarly, the maximum AFW and SI enthalpies were used. The following changes were made to the related assumptions used in the margin to overfill analyses:

   - The 1971+20% ANS decay heat model specified by WCAP-10698-P-A (Reference 1) was used for these analyses.
   - Maximum AFW enthalpy is conservative consistent with WCAP-10698-P-A (Reference 1). For these analyses, the maximum AFW enthalpy of 91.12 Btu/lbm was modeled.
   - Maximum SI enthalpy is conservative consistent with WCAP-10698-P-A (Reference 1). For these analyses, the maximum SI enthalpy of 80 Btu/lbm was modeled for both units.

3. Flashing Fraction

   When calculating the fraction of break flow that flashes to steam, 100% of the break flow was 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 SG, the temperature of the combined flow will be less than the hot-leg temperature and the flashing fraction would be correspondingly lower. Thus, this assumption is conservative.

III.2.D Plant Input

The significant plant-specific input is the same as modeled in the margin to overfill analyses except for the changes listed below.

Braidwood and Byron Stations

Supplemental Information Related to License Amendment Request Regarding Measurement Uncertainty Recapture Power Uprate

August 8, 2012

ATTACHMENT 4

Revised Pages to MUR Power Uprate
Response to NRC Request for Additional Information
Dated February 20, 2012
[ML12052A113]

Table A4-1: Byron/Braidwood Unit 1: Results of Sensitivity Study on MTO With Reduced PORV Flow		
Case (*)	**Description****	**Impact on MTO***** (ft^3)
1m (9)	Low T_{avg}, 5% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy, max SI enthalpy	+0
2m (10)	Low T_{avg}, 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy, max SI enthalpy	+10
3m (11)	High T_{avg}, 5% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy, max SI enthalpy	+52
4m (12)	High T_{avg}, 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy max SI enthalpy	+55
5m****	Low T_{avg}, 5% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy, minimum SI enthalpy	Limiting
6m****	Low T_{avg}, 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy, minimum SI enthalpy	+10
7m****	High T_{avg}, 5% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy, minimum SI enthalpy	+50
8m****	High T_{avg}, 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy minimum SI enthalpy	+54
9m (5)	Low T_{avg}, 5% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy, max SI enthalpy	+48
10m (6)	Low T_{avg}, 0% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy, max SI enthalpy	+59
11m (7)	High T_{avg}, 5% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy, max SI enthalpy	+91
12m (8)	High T_{avg}, 0% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy, max SI enthalpy	+97
17m (1)	Low T_{avg}, 5% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy, max SI enthalpy	+304
18m (2)	Low T_{avg}, 0% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy, max SI enthalpy	+313
19m (3)	High T_{avg}, 5% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy, max SI enthalpy	+361
20m (4)	High T_{avg}, 0% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy, max SI enthalpy	+365

* Corresponding case from Table SBPB R5-1 (Reference A4-1).

** The low T_{avg} cases for the results presented in Table A4-1 were performed taking into consideration a low T_{avg} of 580°F. The low T_{avg} cases presented in Table SBPB R5-1 were performed taking into consideration a low T_{avg} of 575°F.

*** + indicates increase in MTO from the Limiting Case.

**** The minimum SI enthalpy cases, Cases 5m through 8m, have either equal to or less margin to overfill than the corresponding maximum SI enthalpy cases, Cases 1m through 4m. Therefore, minimum SI enthalpy is more limiting.

A4-1 Letter from Kevin F. Borton (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated February 20, 2012 [ML12052A113]