UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

February 1, 2011

Mr. Michael J. Pacilio
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Road
Warrenville IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
VERIFICATION INSPECTION RELATED TO ANALYSIS OF STEAM GENERATOR TUBE RUPTURE EVENT MARGIN TO OVERFILL; 05000456/2011009; 05000457/2011009

Dear Mr. M. Pacilio:

On January 19, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed a verification inspection at your Braidwood Station, Units 1 and 2. This report documents the actions taken to verify if a condition related to the Analysis for Steam Generator Tube Rupture (SGTR) Event Margin to Overfill (MTO), previously identified at Byron Station, also exists at your Braidwood site. The results were discussed on January 19, 2011, with members of your staff.

The inspection examined activities conducted under your license, as they relate to safety and to compliance with the Commission’s rules and regulations, and with the conditions of your license. The inspector reviewed selected analyses, and records.

Based on the results of this inspection, the NRC verified the site had identified a concern with respect to the single failure assumptions taken in your analyses for a steam generator tube rupture event. Specifically, in several previous correspondences, you stated the worst single active failure assumed in the SGTR analysis involved a mechanical failure of a single steam generator power operated relief valve (PORV). This less conservative single failure assumption was not challenged and was subsequently approved by the agency. After further review, the NRC determined the assumption of a single PORV failure is not the most limiting single failure, in that, a failure of electrical components would result in a failure of two PORVs. The staff concluded failures of electrical components should have been postulated to comply with 10 CFR Part 50, Appendix A. The staff assessed this issue as it relates to a backfit and determined that the provisions of 10 CFR 50.109 (a)(4), were applicable, in that, a modification is necessary to bring a facility into compliance with the rules or orders of the Commission.

You are requested to respond to this letter with your assessment of the issue and a description of your intended actions to address the noncompliance, including a proposed schedule to complete those actions. Your actions should also include an assessment of the extent of condition of this issue. Specifically, you are requested to review other transients and accidents outlined in Chapter 15 of your Updated Final Safety Analysis Report and identify any similar discrepancies with respect to the inappropriate reliance or assumption of a single active failure. Identification of such issues should be communicated to the Regional Administrator and should be handled in accordance with your corrective action program.

You have 30 calendar days from the date of this letter to appeal the staff's determination. Such appeals will be considered to have merit only if they meet the criteria given in Part II of NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection." To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

You should provide a response within 30 days of the date of this inspection report, with your proposed actions or the basis for your appeal of the staff's determination, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Braidwood Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by A. T. Boland for/

Steven A. Reynolds, Director
Division of Reactor Safety

Docket Nos. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 05000456/2011009; 05000457/2011009
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000456; 05000457
License Nos:	NPF-72; NPF-77
Report No:	05000456/2011009; 05000457/2011009
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	January 13 – 19, 2011
Inspector:	J. Corujo-Sandín, Reactor Engineer, Mechanical
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2011009; 05000457/2011009; January 13 – 19, 2011; Braidwood Station, Units 1 and 2, verification inspection.

This report covers a verification inspection by regional inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. **NRC-Identified and Self-Revealed Findings**

No findings were identified.

B. **Licensee-Identified Violations**

No violations of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

4OA5 Other Activities

.1 Concerns with Licensee's Margin to Overfill (MTO) Analysis Related to Steam Generator Tube Rupture (SGTR) Event.

Introduction: The licensee determined that an unresolved item (URI) identified at Byron Station is directly applicable to Braidwood. The issue is related to the licensee's evaluation of potential failures of the steam generator power operated relief valves (SG PORVs) during a postulated steam generator tube rupture (SGTR) event. Specifically, the licensee's margin to overfill (MTO) analysis was based on the failure of a single SG PORV to open and did not consider the potential failure of two valves to open due to a common electrical system failure (most limiting single failure).

Background Information of the Technical Issue at the Byron Station: During the 2009 Byron Station's Component Design Bases Inspection (CDBI), documented in Inspection Report 05000454/2009007; 05000455/2009007, the inspectors identified a concern related to the appropriateness of the component failure assumed in a design-bases Steam Generator Tube Rupture event (i.e., SGTR concurrent with a Loss of Offsite Power (LOOP) and a single failure). Specifically, the inspectors noted that after a SGTR, the operators open the steam generator power operated relief valves (SG PORVs) associated with the intact steam generators to cooldown and depressurize the reactor coolant system. This operation would be time critical to prevent overfilling the ruptured steam generator and allowing liquid to enter the steam piping. The licensee's SGTR accident analysis was based on the single failure of one SG PORV to open when required; this was consistent with Updated Final Safety Analysis Report (UFSAR) Section 15.6.3 and Table 15.0-15. Failure of one SG PORV would enable operators to cooldown the reactor coolant system using the remaining two SG and associated PORVs.

As documented in the URI (05000454/2009007-03; 05000455/2009007-03), the inspectors noted the four electric/hydraulic SG PORVs (MS018A-D) at the Byron Station, are powered from two redundant 480V electrical busses (Bus 131X and Bus 132X for Unit 1, for example). Each bus provides power to two SG PORVs: Bus 131X provides power to MS018A and MS018D and Bus 132X provides power to MS018B and MS018C. Therefore, the failure of a single electrical power supply could result in the failure of two SG PORVs to operate. For example, if a rupture were to occur on steam generator B, the failure of motor control center (MCC) 131X2 (or Bus 131X or associated breakers) would result in the failure of MS018A and -D, leaving only MS018C available for cooldown (i.e., only steam generator C will be available for cooldown). Using current procedures, the operators would not be able to cooldown the reactor at the appropriate rate to prevent overfilling of the ruptured steam generator. This will result in a condition outside of the Byron licensee's accident analysis.

The inspectors noted that the Byron licensing basis for SGTR events were based on the generic Westinghouse analysis. The Westinghouse SGTR analysis (WCAP-10698) was based on a three-loop reference plant and the failure of a single SG PORV to open but did not specifically address electrical bus failures. In the single failure evaluation section, the WCAP stated, "common mode failures of all steam generator PORVs were not evaluated since electrical power and air supplies to the PORVs are largely plant specific…." The associated NRC evaluation (dated March 30, 1987) concluded that the WCAP analysis methodology was conservative, but pointed out that there may be major design differences between plants and required plant specific information. Section D.5 of the NRC evaluation required the following plant specific information, "A survey of plant primary and 'balance-of-plant' systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided."

In response to the NRC, the licensee provided the required plant specific information (Commonwealth Edison letter, dated April 25, 1990). This letter included Revision 1 of the SGTR analysis for the Byron and Braidwood plants. The analysis stated, in part, "The compatibility of the Byron/Braidwood systems with the WCAP-10698-P-A bounding plant analysis has been evaluated and no major design differences affecting the MTO exist. The same limiting single failures as identified in WCAP-10698-P-A and Supplement 1 of WCAP-10698-P-A were utilized in the analysis…." The NRC's evaluation of the Byron/Braidwood plant specific SGTR analysis (NRC letter dated April 23, 1992) included a statement that the licensee had responded satisfactorily to this confirmatory issue.

Following the Byron Station CDBI, the inspectors requested assistance from the Office of Nuclear Reactor Regulation (NRR) in providing a position of single failure in the SGTR accident analysis. The staff from NRR reviewed the issue and provided a response to Task Interface Agreement (TIA) 2010-002 by letter dated December 20, 2010, (ML103230177). In the response, NRR determined that the failure of a breaker to perform its safety function, regardless of how that failure occurs, is considered a single failure as defined by 10 CFR Part 50, Appendix A. The staff also concluded that the Byron Station licensing basis includes consideration of the most limiting single failure in the design of safety systems as defined in Appendix A to 10 CFR Part 50. The existing design does not conform to the single failure criteria defined in Appendix A to 10 CFR Part 50 and Section 3.1 of Braidwood/Byron Station UFSAR.

Applicability to the Braidwood Station: Because of the similarities in design and because several correspondences included the Braidwood Station, the inspectors evaluated if the issue identified at Byron Station could also be applicable to Braidwood Station. That is, the assumption of a single active failure of a SG PORV was not the limiting failure; and the licensee needed to consider the failure of an electrical source (resulting in the failure of two SG PORVs) to meet the definition of a single failure as defined by 10 CFR Part 50, Appendix A. As a result, this verification inspection was performed. During this inspection, the inspector noted the following:

- The Braidwood licensee had initiated AR 00897366, which documented the SGTR MTO analysis of record for Braidwood Station also assumed only a single active failure of a SG PORV. Due to the complexity of establishing the appropriate design and licensing bases for this issue, the Braidwood licensee needed the final determination on the NRC's licensing bases/regulatory position relating to the Byron Station URI in order to proceed with their actions.

- The four electric/hydraulic SG PORVs (MS018A-D) at the Braidwood Station are powered from two redundant 480V electrical busses (Bus 131X and Bus 132X for Unit 1, for example). Each bus provides power to two SG PORVs: Bus 131X provides power to MS018A and MS018D; Bus 132X provides power to MS018B and MS018C. Therefore, the failure of a single electrical power supply could result in the failure of two SG PORVs to operate.

- The inspectors noted that the Braidwood licensing basis for SGTR events were also based on the generic Westinghouse analysis, WCAP-10698. In addition, the correspondences discussed above which were used to address this analysis for the Byron Station also pertained to the Braidwood Station. Therefore, the NRC's letter dated April 23, 1992, which stated the licensee had responded satisfactorily to this confirmatory issue was also applicable for the Braidwood Station.

- The licensee initiated AR 01155373 on December 22, 2010, after being informed of the results the TIA. In the AR, the licensee acknowledged receiving the conclusions of the TIA, regarding the need for Byron Station to consider both active and passive failures in the SGTR MTO analysis. The AR stated that Byron's analysis was directly applicable to Braidwood. The AR also discussed some of the compensatory actions the licensee had initiated at Braidwood Station. The licensee revised Braidwood Chemistry Procedures 1/2BwCSR 3.4.16.2-1 to direct personnel to evaluate the SGTR Analysis if reactor coolant Dose Equivalent Iodine-131 is greater than 0.5 microcuries per gram (1/2 of the Technical Specification limit). In addition, a standing order was issued to all licensed operators, stating that the administrative limit for entry into LCO 3.4.16 is now 0.083 microcuries per gram. These new limits derive from engineering change (EC) 375055, which concluded that if the RCS Iodine activity were limited to 0.083 microcuries per gram and a SGTR with overfill were to occur, the resulting dose would be within the regulatory limits.

<u>Disposition of Technical Issue</u>: Based on this inspection and on the conclusions documented in TIA 2010-002, the inspector concluded the use of a single failure (including passive and active failures of electrical systems) should have been assumed in the SGTR event analysis for the Braidwood Station. However, the inspectors identified several inconsistencies within the NRC's correspondences to the Braidwood licensee during the original evaluation of a SGTR event.

Specifically:

- In the request for additional information dated April 19, 1984, from B. J. Youngblood, (Chief, Licensing Branch) to Mr. Farrar (licensee), the NRC stated, "Include in the analysis of the SGTR accident the most limiting **single active failure** *[emphasis added]*. If the most limiting **single active failure** is failure of an atmospheric relief valve to close, operator action to close the block valve may be assumed if justified."

- In a letter from K. Ainger (licensee) to H. Denton (then Director of NRR) dated January 21, 1987, the licensee stated that it was demonstrated that the operator can perform the required SGTR recovery actions and "… given this configuration without overfill and assuming the **worst single failure** *[emphasis added]*, the evaluation demonstrated offsite radiation doses to be within the allowable dose guidelines…."

- In a letter from Schuster (licensee) to Dr. Murley (then Director of NRR), dated April 25, 1990, the licensee transmitted their SGTR analysis. In this document, the licensee misquoted WCAP10698-P-A by stating "the most limiting **single active failures** *[emphasis added]*." *[Note: The WCAP does not use the term "active."]*

- In a letter and attached Safety Evaluation Report, dated April 23, 1992, stated the licensee's response satisfactorily addressed the overfill criteria.

- In a letter from John Hosmer (licensee) to the NRC dated November 13, 1996, the licensee transmitted a topical report in support of their replacement of the steam generators. The licensee stated that the new analysis met the requirements set forth in the April 23, 1992, SER. On page 15 of this analysis, the licensee stated that "the use of the same limiting single failures as identified in WCAP 10698-P-A and Supplement 1 of WCAP 10698-P-A are applicable for this analysis."

- In a letter dated May 20, 1997, from G. Dick of NRR to I. Johnson (licensee), the NRC requested the licensee to provide additional information - specifically requesting the licensee to justify why the **single failures** *[emphasis added]* chosen for the different cases remains bounding considering the changes in procedures, plant configuration, and analysis methods.

- In a letter dated June 24, 1997, from Hosmer (licensee) to the NRC, the licensee addressed the Question 2 posed in the May 20, 1997 letter. In this letter, the licensee provided clarification of the single active failure by stating, "the following three **single active failure** *[emphasis added]* were investigated: intact steam generator PORV failure, AFW [auxiliary feedwater] flow control valve failure, and Main Steam Isolation Valve failure. It was determined in the Reference 2 submittal that the most limiting **single active failure** *[emphasis added]* is the intact steam generator PORV failure." The licensee also provided justification for this assumption.

- In a letter dated January 28, 1998, from G. Dick (NRR) to O. Kingsley (licensee), the NRC completed the review and concluded that with regard to the SGTR accident analysis, the staff finds the replacement of the steam generators at Byron and Braidwood acceptable. On page 5 of the SER, the NRC stated, "additionally, a number of single failures were evaluated to determine the most limiting failure."

In reviewing the correspondences, the inspectors concluded the NRC was not clear or consistent with communicating the need to assume passive failures of the electrical components even though passive failures were required to be evaluated under 10 CFR Part 50, Appendix A. Therefore, the current NRC staff position regarding the requirement to evaluate single passive failures of the electrical components is compliant with Appendix A but is different than the staff position previously communicated to the licensee. Therefore, the provisions of 10 CFR 50.109 are applicable. Specifically, 10 CFR 50.109 defines backfitting as "the modification of or addition to systems, structures, components, or design of a facility, any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position." After consultation with NRR and the Office of General Counsel, the inspectors determined that no backfit analysis is required under 10 CFR 50.109(a)(2) because the provisions of 10 CFR 50.109 (a)(4), were applicable, in that, a modification is necessary to bring a facility into compliance with the rules or orders of the Commission.

Regional management discussed these conclusions and the need to be in compliance, with the licensee. The licensee initiated corrective actions: (1) establishing an administrative limit for reactor coolant activity which is more limiting than the current technical specifications; (2) performing an analysis on the steam generator main steam line supports to ensure integrity if the steam generators were to overfill; (3) evaluating potential changes to the current emergency action level classification for a tube rupture event; and (4) revising procedures accordingly. In addition, the licensee plans to modify the power sources for the affected breakers.

This issue is considered unresolved pending the licensee's response to this inspection report (URI 05000456/2011009-01; 05000457/2011009-01).

4OA6 Management Meeting(s)

.1 Exit Meeting Summary

- On January 19, 2011, the Reactor Inspector presented the inspection results to Mr. G. Dudek (Acting Plant Manager), and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

G. Dudek, Training Manager, Acting Plant Manager
B. Finlay, Security Director
R. Gaston, Regulatory Assurance Manager
J. Gerrity, Exelon Regulatory Assurance
J. Muraida, Operations
J. Odeen, Project Management Manager
R. Randanovich, Exelon Nuclear Oversight
T. Schuster, Chemistry Manager
M. Smith, Engineering Director

Nuclear Regulatory Commission

M. Satorius, Region III, Regional Administrator
S. Reynolds, Director, Division of Reactor Safety
E. Duncan, Chief, Division of Reactor Projects, Branch 3
A.M. Stone, Chief, Division of Reactor Safety Engineering Branch 2
J. Corujo-Sandín, Reactor Engineer, Division of Reactor Safety Engineering Branch 2

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000456/2011009-01; 05000457/2011009-01;	URI	Concerns with Licensee's Margin to Overfill (MTO) Analysis Related to Steam Generator Tube Rupture (SGTR) Event

Closed and Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

CORRECTIVE ACTION PROGRAM DOCUMENTS

Number	**Description or Title**	**Date**
AR 00897366	Preliminary NRC Info on Single Failure for SGTR MTO	March 25, 2009
AR 01155373	Additional Information on Single Failure for SGTR MTO	December, 22, 2010

MISCELLANEOUS

Number	**Description or Title**	**Date or Revision**
Log number: 10-019	Standing Order: RCS Activity Administrative Controls	December 22, 2010

PROCEDURES

Number	**Description or Title**	**Revision**
1BwCSR 3.4.16.2-1	Unit 1 Reactor Coolant Dose Equivalent Iodine-131-Once per 14 Days or Due to Changing Reactor Power.	8
2BwCSR 3.4.16.2-1	Unit 2 Reactor Coolant Dose Equivalent Iodine-131-Once per 14 Days or Due to Changing Reactor Power.	9

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
CFR	Code of Federal Regulations
IR	Inspection Report
LOOP	Loss of Offsite Power
MTO	Margin to Overfill
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publicly Available Records System
PORV	Power Operated Relief Valve
SER	Safety Evaluation Report
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
TIA	Task Interface Agreement
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

discrepancies with respect to the inappropriate reliance or assumption of a single active failure. Identification of such issues should be communicated to the Regional Administrator and should be handled in accordance with your corrective action program.

You have 30 calendar days from the date of this letter to appeal the staff's determination. Such appeals will be considered to have merit only if they meet the criteria given in Part II of NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection." To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

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Sincerely,

/RA by A. T. Boland for/

Steven Reynolds, Director
Division of Reactor Safety

Docket Nos. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 05000456/2011009; 05000457/2011009 w/Attachment: Supplemental Information

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OFFICIAL RECORD COPY

Letter to Mr. Michael J. Pacilio from Mr. Steven A. Reynolds dated February 1, 2011.

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
VERIFICATION INSPECTION RELATED TO ANALYSIS OF STEAM GENERATOR TUBE RUPTURE EVENT MARGIN TO OVERFILL; 05000456/2011009; 05000457/2011009

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