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June 24, 1997

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Response to Request for Additional Information Regarding the Revised Steam Generator Tube Rupture Analysis

Byron Nuclear Power Station
Facility Operating License NPF-37 and NPF-66
NRC Docket Numbers: 50-454 and 50-455

Braidwood Nuclear Power Station
Facility Operating License NPF-72 and NPF-77
NRC Docket Numbers: 50-454 and 50-455
6 7

References:

1. J. Hosmer Letter to USNRC, "Steam Generator Tube Rupture Analysis for Byron and Braidwood Generating Stations", dated November 13, 1996.

2. G. Dick Letter to I. Johnson, Request for Additional Information Regarding the Revised Steam Generator Tube Rupture Analysis - Byron and Braidwood Stations, dated February 11, 1997.

3. J. Hosmer Letter to USNRC, Response to Request for Additional Information Regarding the Revised Steam Generator Tube Rupture Analysis - Byron and Braidwood Stations, dated March 20, 1997.

4. G. Dick Letter to USNRC, Request for Additional Information Regarding the Revised Steam Generator Tube Rupture Analysis - Byron and Braidwood Stations, dated May 20, 1997.

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On November 13, 1996, Commonwealth Edison Company (ComEd) submitted its revised steam generator tube rupture analysis for Byron and Braidwood Stations (Reference 1). On February 11, 1997, NRC issued a request for additional information (Reference 2). ComEd provided its response on March 20, 1997 (Reference 3). Another request for additional information was transmitted on May 20, 1997 (Reference 4). This document is ComEd's response to the May, 1997, request.

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Please direct any questions to this office.

Dale L. Shaw
for

John B. Hosmer
Vice President
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Attachment

cc: A. B. Beach - Regional Administrator, RIII
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Office of Nuclear Safety, IDNS

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REVISED STEAM GENERATOR TUBE RUPTURE ANALYSIS - BYRON STATION AND BRAIDWOOD STATION

NRC Question 1

The response to question A.1 in the licensee's March 20, 1997, submittal indicates that there is a requirement to choose a cycle Tave that is consistent with the analyzed Tave window. Where is the requirement (to choose a Tave greater than 575 °F) contained?

Response 1

The reload design process used at ComEd follows WCAP 9272 (Reference 1). Table 5.3 in WCAP 9272 shows a list of the general and specific accident analysis parameters which, if applicable, are evaluated for every reload cycle of each plant reloaded by Westinghouse. ComEd documents these parameters in the Reload Design Key Parameter Checklist (RDKPC). The RDKPC specifies the minimum and maximum values for Tave. A minimum Tave value of 575°F will be specified in the RDKPC after the Reference 3 topical is approved. The reload design Tave must fall between the maximum and minimum values for the reload design to be acceptable. The generation and revision of the RDKPC is controlled by procedure as part of the reload design process.

NRC Question 2

The response to question A.2 in the licensee's March 20, 1997, submittal indicates that there is significant conservatism in the initial conditions and analysis approach. Please provide a more detailed discussion of the conservatism in the initial conditions and analysis approach. Please include a specific discussion of the following issues:

- The limiting case for the old analysis is used for the new analysis. Justify why this is acceptable considering the plant now has new emergency operating procedures (EOPs), new steam generators, and a new analysis methods. Be sure to justify why the single failures chosen for the different cases remains bounding considering the changes in the procedures, plant configuration, and analysis method.
- Page 9 of the original submittal (November 13, 1996) states that the secondary mass and pressure are derived from the revised nominal Tave. Why is the use of the nominal Tave more appropriate than a conservative value?

Response 2

The initial conditions and analysis approach are generally the same between the 1990 Topical Report submittal and the 1996 submittal. The 1996 submittal identified any revisions to the initial conditions or assumptions from the approved SGTR Topical Report. These revisions have maintained adequate levels of conservatism in the SGTR analysis. As a method to demonstrate the amount of conservatism in the SGTR analysis, the March 20, 1997 response to NRC Question A.2 provided sensitivity studies for several key parameters, including any that had uncertainty changes as identified in the 1996 submittal. Additionally, the sensitivities to the input operator action times were also provided. Finally, a discussion of the change in the value used for turbine runback was presented and justification that adequate conservatism was maintained was provided.

Additional information on the specific clarifications requested in the details of this question are provided in this response. The single active failure chosen for the November 1996, submittal is discussed. A clarification of the initial steam generator mass and pressure assumption is also provided.

Single Active Failure

In the 1990 submittal (Reference 2), the following three single active failures were investigated: intact steam generator PORV failure, AFW flow control valve failure, and Main Steam Isolation Valve (MSIV) failure. It was determined in the Reference 2 submittal that the most limiting single active failure is the intact steam generator PORV failure. This single active failure is assumed in the 1996 submittal (Reference 3). The following discussion provides justification for this assumption.

If the single active failure is removed from the Reference 3 analysis, while keeping the rest of the input assumptions the same, the amount of Margin to Overfill (MTO) for the

Original Steam Generator (OSG) and for the Replacement Steam Generator (RSG) are 619 cubic feet and 356 cubic feet, respectively. As reported in the Reference 3 submittal, failure of an intact steam generator PORV reduces the amount of MTO to 293 cubic feet and 60 cubic feet for the OSG and RSG, respectively. The decreases in MTO due to intact steam generator PORV failure for the OSG and the RSG are 326 and 296 cubic feet, respectively (see Tables 1 and 2).

The failure of a MSIV has no impact on the steam generator tube rupture (SGTR) transient response because the condenser is assumed to be lost due to loss of offsite power. However, in accordance with the Emergency Operating Procedures (EOPs), operators must perform the contingency actions associated with the MSIV failure. These additional actions may delay the operator response times assumed in the analysis. The delay is estimated to be less than 4 minutes in WCAP 10698-P-A (Reference 4). This is a conservative estimate. Most of the valves, which must be closed as part of the contingency actions for a MSIV failure, can be closed in the control room. For the valves which must be closed locally, an operator can be sent out to do so while the rest of the crew continue on with the procedure. It should also be noted that, per the EOPs, the contingency actions for a MSIV failure take place after AFW isolation, which is the most critical step in mitigating a SGTR event. Therefore, the contingency actions can be performed well within 4 minutes from the standpoint of total procedure performance.

The steam generator fill rate after AFW isolation is conservatively estimated to be 1 cubic foot per second. With a MSIV failure, the delay due to performance of contingency is conservatively estimated to be 4 minutes. The decrease in MTO can then be estimated to be 240 cubic feet.

Failure of the AFW control valve in the open position is already considered in the Reference 2 and 3 analyses. With the loss of offsite power and loss of instrument air, the AFW control valves are assumed to be in the full open position and maximum AFW flow is delivered to the ruptured steam generator.

Failure of the AFW control valve in the closed position on a intact steam generator can increase flow to the ruptured steam generator while decreasing flow to the intact steam generators. This flow redistribution will impact the transient response and result in a decrease in MTO. The analysis performed to model the flow redistribution results in a MTO of 472 cubic feet and 217 cubic feet for the OSG and RSG, respectively. This represents a decreases in MTO of 147 and 139 cubic feet from the reference case for the OSG and RSG, respectively (see Tables 1 and 2).

The impact of intact steam generator PORV failure, MSIV failure, and AFW control valve failure has been investigated. Table 1 summarizes the results for the OSG and Table 2 summarizes the results for the RSG. The most limiting single active failure remains the intact steam generator PORV failure.

Table 1 – Single Active Failure (OSG)

Case	MTO (cubic feet)	Decrease from Reference Case (cubic feet)
Reference (no active failure)	619	-
Intact steam generator PORV failure	293	326
MSIV failure	379 *	240 *
AFW control valve failure	472	147

Table 2 – Single Active Failure (RSG)

Case	MTO (cubic feet)	Decrease from Reference Case (cubic feet)
Reference (no active failure)	356	-
Intact steam generator PORV failure	60	296
MSIV failure	116 *	240 *
AFW control valve failure	217	139

* From 4 minutes delay in operator responses assumed in analysis

Steam Generator Mass and Pressure

The secondary mass and pressure are derived from the revised nominal Tave of 575 °F. Uncertainties are then conservatively applied to the derived values. The analysis would be overly conservative with uncertainties applied to the steam generator mass and steam generator pressure at the analysis Tave of 567 °F, which already included uncertainty.

NRC Question 3

The response to question A.3 of the March 20, 1997, submittal indicates that the 1973 standard decay heat curve is used; however, the safety evaluation for the referenced topical report indicates that 120 percent of the 1971 ANS decay heat rate is to be used. Please describe why the 1973 standard decay heat curve is acceptable.

Response 3

The decay heat model used in the November 1996 submittal and the decay heat model used in the original 1990 submittal are the same. They both used 120% of the 1971 ANS decay heat model. In other words, the same RETRAN decay heat model was used.

The RETRAN manual for MOD 5 refers to this decay heat model as the 1973 decay heat model. However, the reference in the manual for this model is the ANS 1971 decay heat standard.

NRC Question 4

With regard to the choice of initial power level in the accident analysis, the discussion provided in response to question A.5 of the March 20, 1997, submittal and the discussion in the original submittal November 13, 1996, is not consistent with the discussion in the topical report and the safety evaluation for the topical report. Please provide a more rigorous justification for the initial power chosen. Include a complete discussion of the different effects contained in both the topical report and the submittals for this assumption. Include the effects for both the original and the replacement generators.

Response 4

The discussion provided in page 3-2 of the 1990 submittal (Reference 2) stated that increasing the core decay heat was found to nominally increase the MTO.
This finding was based on the results of two competing sets of effects involving steam generator **system responses** and the behavior of the **simplified steam generator model**.

The first set of effects deals with the system responses to higher decay heat and is described in the response to question A.5 of the March 20, 1997, submittal (Reference 5). The higher decay heat increases the transient response time to depressurize the RCS after reactor trip and, therefore, decreases the MTO. It also leads to a higher primary to secondary leak rate during the RCS cooldown and depressurization period, which decreases the MTO. A higher decay heat level does, however, increase the steam release through the steam generator relief valves after reactor trip causing an increase in the MTO. The net effect of the interaction of these system responses is a slight reduction in MTO with an **increase** in decay heat for both the original and replacement steam generators.

The second set of effects deals with a characteristic of the simplified steam generator modeling. The simplified steam generator model used in the RETRAN SGTR analysis is described in Section A-3 of the 1990 submittal (Reference 2). The steam generator is modeled as a single saturated volume for both the original and the replacement steam generators. It was noted in Section 3.2.2 of the Westinghouse Owners Group (WOG) analysis (Reference 4) that a simplified steam generator model at homogeneous, saturated conditions, predicts unrealistically slow secondary side temperature response and artificially lowers the steam generator pressure when the tube bundle region is being cooled by AFW. The artificially lowered steam generator pressure leads to increased break flow which causes a reduction in MTO. With a lower decay heat level, the primary side temperature is also lower which leads to an even larger reduction in MTO. The net effect is a slight reduction in MTO with a **decrease** in decay heat for both the original and replacement steam generators.

Since these effects are competing effects and both are only marginal in their impact, it is not possible to determine the most conservative conditions to assume without actually running cases. For the 1996 submittal, both the RSG and the OSG models were run at

100% and 102% power to determine the most conservative case. In both cases, the decay heat model was the approved RETRAN model of 120% of the 1971 ANS decay heat model. The results show that the MTO was reduced for the RSG 102% power case. For the OSG case, the MTO was increased at 102% power. Since the RSG is clearly the limiting case with regard to MTO, the RSG results represented the more conservative approach. Therefore, the power uncertainties of +2% were applied to the evaluations presented in the 1996 submittal.

NRC Question 5

Please explain why the initial steam generator volumes are different for the OTΔT and the low pressurizer pressure cases in Figure 7 of the November 13, 1996, submittal.

Response 5

There is a difference in initial SG volume for the OTΔT and the low pressurizer pressure cases due to the difference in reactor trip times. Since the reactor trip from OTΔT will occur sooner than the reactor trip on low pressurizer pressure, the amount of turbine runback will be smaller for the OTΔT case than the low pressurizer pressure case. In the SGTR analysis, turbine runback is modeled by assuming the SG mass at the runback power level. With a smaller amount of turbine runback (and subsequent higher power level), the initial SG mass (or liquid volume) is less for the OTΔT case.

NRC Question 6

The November 13, 1996, submittal indicates that the analysis assumptions are being revised because the EOPs are being modified. Please verify that the equipment and instrumentation relied on to identify and mitigate the tube rupture is all safety-related.

Response 6

The equipment and instrumentation used to identify and mitigate the SGTR event are safety-related. The revisions to the EOPs, referred to in the November 1996 submittal, did not change which components were used for identification or mitigation of the SGTR. The change was in the step sequence of the procedure. This change allows the operator to act sooner for certain mitigating actions, primarily, isolation of Auxiliary Feedwater.

References:

1. "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272, March, 1978.

2. "Steam Generator Tube Rupture Analysis for Byron and Braidwood Plants, Revision 1," ComEd Report, March 1990.

3. "Revised Steam Generator Tube Rupture Analysis for Byron/Braidwood," NFSR-0114, November 1996.

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

5. "Response to Request for Additional Information for the Steam generator Tube Rupture Analysis," letter from John B. Hosmer (ComEd) to NRC, March 20, 1997.

6. "Byron/Braidwood Stations Updated Final Safety Analysis Report," Revision 6.