Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

ComEd

April 20, 1999

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

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: Response to Request for Additional Information
Steam Generator Tube Rupture Analysis for Byron Station, Unit 2, and
Braidwood Station, Unit 2

References: (1) NRC letter, "Revised Steam Generator Tube Rupture Analysis – Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, dated January 28, 1998.

(2) NRC letter, "Revised Steam Generator Tube Rupture Analysis – Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, dated March 11, 1998.

(3) Letter from D. R. Helwig (ComEd) to U.S. NRC, "Revised Steam Generator Tube Rupture Analysis Dose Evaluations for Byron and Braidwood Stations Unit 2," dated April 13, 1998.

The safety evaluation for the revised Steam Generator Tube Rupture (SGTR) analysis for Byron Station, Unit 1, and Braidwood Station, Unit 1, was issued by the NRC in a letter dated January 28, 1998 (Reference 1), as supplemented by a letter dated March 11, 1998 (Reference 2). Reference 3 provided information to the NRC in order to complete the SGTR dose evaluations for Byron Station, Unit 2, and Braidwood Station, Unit 2.

In a telephone conference held between representatives of Commonwealth Edison (ComEd) Company and the NRC on December 3, 1998, additional information was requested regarding the "Revised Steam Generator Tube Rupture Analysis" for Byron Station, Unit 2 and Braidwood Station, Unit 2. In a follow-up telephone conference held between representatives of ComEd and the NRC on February 11, 1999, the additional information was discussed and the issues reflected in the NRC requests were resolved. The additional information and subsequent resolution of the issues discussed during the February 11, 1999, telephone conference are documented in the attachment.

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Should you have any questions concerning this letter, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Vice President – Regulatory Services

Attachment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station

Attachment

Response to Request for Additional Information
Revised Steam Generator Tube Rupture (SGTR) Analysis
Byron Station, Unit 2, and Braidwood Station, Unit 2

<u>Request 1</u>

Section 4 of Nuclear Fuel Services Report, NFSR-0114, Revision 0, "Revised Steam Generator Tube Rupture Analysis for Byron/Braidwood," dated November 1996, states that the methodology used in Commonwealth Edison (ComEd) Company Report, "Steam Generator Tube Rupture Analysis for Byron and Braidwood Plants, Revision 1," dated February 1990, to calculate offsite dose, is consistent with the methodology used in WCAP-10698-P-A, Supplement 1, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," dated March 1986. A review of the two methodologies identified that the WCAP offsite dose analysis considers steam release through the ruptured steam generator (SG) power operated relief valve (PORV) for the 2 to 8 hour period after the SGTR event for the purpose of depressurizing the reactor coolant system (RCS) to enable use of the Residual Heat Removal (RHR) system. These releases are substantial in quantity. The 1990 ComEd analysis does not consider this release. Explain this inconsistency.

Response

We acknowledge this difference in the two methodologies. When we previously stated that the ComEd methodology was "consistent" with the Westinghouse methodology, it was not meant to imply that the two methods were identical. During plant cooldown, the Byron Station and Braidwood Station Emergency Operating Procedures caution the operators not to release steam from the ruptured SG if there is a concern with SG overfill or with meeting 10 CFR 20 limits. Byron Station and Braidwood Station Emergency Procedure ES-3.3, "Post-SGTR Cooldown Using Steam Dump," has CAUTION statements that specifically state, "Steam should <u>NOT</u> be released from any ruptured SG if water may exist in its steamline," and "An offsite dose evaluation should be completed prior to using this procedure to ensure 10 CFR 20 limits are <u>NOT</u> exceeded." Since the ruptured SG will not be used to depressurize the plant, the ComEd methodology does not consider steam release from the ruptured SG, after the associated PORV has been isolated, in the offsite dose calculation.

In addition, the ComEd offsite dose calculation methodology does not consider the steam release from the intact SGs during the two to eight hour RCS cooldown period, because of the small contribution to the offsite dose. However, to be consistent with the methodology in WCAP 10698-P-A Supplement 1, ComEd will incorporate the steam release from the intact SG, during the two to eight hour RCS cooldown period, into the SGTR offsite dose calculation the next time the analysis is performed.

Request 2

Provide the values for steam release from the ruptured and intact SGs during an SGTR event, including the two to eight hour period during RCS cooldown.

Response

The amount of steam release following a SGTR event is provided in Table 1. It should be noted that the ruptured SG PORV is isolated at 1715 seconds after event initiation. Therefore, although RCS depressurization termination and the associated break flow occur at a later time (i.e., 3763 seconds), there is no steam release from the ruptured SG after 1715 seconds. Table 1 also gives the steam release values from the intact SGs from the point of RCS depressurization termination to two hours into the event (i.e., from 3763 seconds to 7200 seconds), and during the two to eight hour cooldown period.

Request 3

Table 7, "Sequence of Events," of report NFSR-0114 indicates that ruptured SG tube flow continues until 3763 seconds following the tube rupture event, at which point the RCS depressurization is terminated. However, Figure 15 of the same report appears to indicate that ruptured SG tube flow continues until approximately 4200 seconds into the event. Explain this apparent inconsistency.

Response

Table 7 of NFSR-0114 shows that RCS depressurization termination occurs 3763 seconds into the event at which time the differential pressure (i.e., primary side to secondary side) across the ruptured SG tube is equalized and break flow is terminated. Figure 15 of NFSR-0114 indicates that ruptured tube break flow initially goes to "0 lbm/sec" at 3763 seconds. However, due to RCS pressure perturbations while equalizing charging and letdown flow, ruptured tube break flow is momentarily re-initiated twice more before ending the scenario at 4200 seconds.

Request 4

It appears that the ComEd April 13, 1998, submittal contains incorrect primary coolant activity levels for the SGTR accident-initiated spike case. The ComEd primary coolant activity values appear to be approximately 4 times too large. The correct values should be provided.

Response

It is recognized that the initial primary coolant activity levels used for the SGTR accident-initiated spike case are higher (i.e., more conservative) than the steady state primary coolant activity levels allowed by Technical Specifications for continued operation, (i.e., 1 μCi/gm dose equivalent iodine-131 concentration). The values used for the SGTR accident-initiated spike case in the ComEd April 13, 1998, submittal, were taken from the Updated Final Safety Analysis Report (UFSAR) Table 12.2-2, "Reactor Coolant

Sources for Shielding Design." The SGTR accident-initiated spike case should assume equilibrium iodine concentration for continued full power operation in combination with an assumed accident-initiated iodine spike. The activity levels in UFSAR Table 12.2-2 are conservative with respect to equilibrium iodine concentration values for continued full power operation. The activity levels for the individual iodine isotopes, corresponding to the Technical Specification primary coolant activity limit of 1 μCi/gm dose equivalent iodine-131 concentration, will be used the next time the analysis is performed and associated UFSAR changes will be made.

Table 1

Steam Release Following a Steam Generator Tube Rupture Event

Event	Intact SG Steam Release (lbm)	Ruptured SG Steam Release (lbm)
Event Initiation to Ruptured SG Power Operated Relief Valve Isolation and Break Flow Termination (0 – 1715 seconds; PORV Isolated) (0 – 3763 seconds; Break Flow Terminated)	- - 1.51E5	9.19E4 - -
Break Flow Termination to 2 hours (3763 – 7200 seconds)	3.64E5	- -
Plant Depressurization - Time to put the RHR System In Service (2 – 8 hours)	1.18E6	- -