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ComEd

March 20, 1997

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Byron Nuclear Power Station, Units 1 and 2
Facility Operating Licenses NPF-37 and NPF-66
NRC Docket Numbers: 50-454 and 50-455

Braidwood Nuclear Power Station, Units 1 and 2
Facility Operating Licenses NPF-72 and NPF-77
NRC Docket Numbers: 50-456 and 50-457

Response to Request for Additional Information for the Steam Generator Tube Rupture Analysis

References:

1) J. Hosmer Letter to NRC transmitting Topical Report for the Revised Steam Generator Tube Rupture Topical Report for Byron and Braidwood Stations dated November 13, 1996.

2) NRC Request for Additional Information regarding the Revised Steam Generator Tube Rupture Analysis dated February 11, 1997.

3) Telecon between M. Lesniak, et al and G. Dick (NRC) on March 19, 1997.

In Reference 1 ComEd submitted a revised Steam Generator Tube Rupture (SGTR) Analysis Topical Report to the NRC. Reference 2 was an NRC Request for Additional Information (RAI) concerning the margin to overfill calculation and operator actions discussed in Reference 1. The answers to the specific questions raised in the RAI are contained in Attachment 1.

Per Reference 3 ComEd has included, as part of this response, the necessary information to perform the independent calculations for the maximum offsite radiological consequences from the postulated steam generator tube rupture event. The assumptions used and the iodine and noble gas releases calculated by ComEd are provided in the attachment.

A001/1

Please address any questions or comments to Marcia Lesniak, Nuclear Licensing Administrator, at 630-663-6484.

Sincerely,

John Hosmer

John B. Hosmer
Engineering Vice President

Attachments

cc: A. B. Beach, Regional Administrator - RIII
G. Dick, Byron/Braidwood Project Manage - NRR
C. Phillips, Senior Resident Inspector - Braidwood
S. Burgess, Senior Resident Inspector - Byron
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 A.1

The submittal states that, as part of the T-HOT reduction program, Byron and Braidwood have been analyzed for a Tavg window of 569.1 degrees Fahrenheit to 588.4 degrees Fahrenheit. The analysis submitted assumes 567.1 degrees Fahrenheit, 2 degrees Fahrenheit lower than the low end of the Tavg window. The Updated Final Safety Analysis Report states (pg. 15-09) that the average reactor coolant system temperature uncertainty is ±4.7 degrees Fahrenheit. Please explain why the choice of 567 degrees Fahrenheit bounds the operational range with instrument uncertainty included.

Response A.1

The analysis average Reactor Coolant System (RCS) temperature (Tavg) of 567 °F corresponds to a nominal Tavg of 575 °F with an uncertainty of -8 °F. This uncertainty is consistent with the Updated Final Safety Analysis Report (UFSAR) (pg. 15-09, revision 6, December 1996). After the approval and implementation of the Reference 1 analysis, the hot full power Tavg window for Byron and Braidwood will be limited to between 575 °F to 588.4 °F. For each reload, the design limits, including the Tavg window, are generated based on analyses used to support the design cycle. Therefore, the reload design process ensures the Tavg chosen for the design cycle is consistent with the analyzed Tavg window.

NRC Question A.2

The analysis concluded that the limiting MTO is 60 cubic feet. Given that this is probably less than a half a foot of water level, less than 30 seconds of auxiliary feedwater flow, less than a few seconds of main feedwater flow and less than 2 percent of the total steam generator volume, please provide greater justification that this very small margin gives reasonable assurance that the steam generators will not fill. An evaluation of the sensitivity of key parameters to the margin available or an estimation of the calculational uncertainty may be helpful in determining if the margin is adequate.

Response A.2

The Reference 1 analysis contains significant conservatism in initial conditions and analysis approach. The analysis, therefore, inherently provides additional margin beyond the calculated margin to overfill (MTO) of 62 ft^3. Table A.2 shows the sensitivity of several key parameters that affect the margin available. All cases were based on the Unit 1 replacement steam generators because they are limiting for MTO.

The first case removes uncertainties from the initial values of power, secondary pressure and Tavg. After these changes are made, the initial conditions are consistent with the design basis Steam Generator Tube Rupture (SGTR) analysis (Reference 2). The result is a gain in MTO of 137 ft^3.

The second case evaluates the impact of removing uncertainty from the initial value of primary pressure. The third case evaluates the impact of removing the uncertainty from the initial steam generator level. The gains in margin to overfill are 35 ft^3 and 159 ft^3, respectively.

The Reference 1 analysis assumes immediate auxiliary feedwater (AFW) flow upon reactor trip and loss of offsite power. However, there are delays to the start of the AFW flow associated with the emergency diesel generator starting and loading sequence. Table 8.3-5 of the Byron/Braidwood UFSAR listed a value of 45 seconds for such delay in delivering AFW flow from the 1A AFW pump. Case 4 evaluates the impact of delaying AFW flow from the 1A AFW pump by 30 seconds while the delay for the 1B AFW pump remains at 0 second. The result is a gain in MTO of 54 ft^3.

All of the above cases were run individually while maintaining all other parameters as is. A final case, incorporating all the changes made for cases 1 to 4, shows a net gain in margin to overfill of 339 ft^3, for a total MTO of 401 ft^3.

The parameters presented above are only a subset of the key assumptions. Other key assumptions which contain conservatism include:

- no credit taken for AFW flow throttling via AF013 motor operated valve,
- constant maximum AFW flow instead of SG pressure dependent flows, and
- friction limited break flow instead of choked flow.

The use of best estimate values for all of these assumptions would provide significant additional margin beyond the 339 ft^3 identified in this response.

In summary, the Reference 1 analysis contains conservatism in key assumptions and resulted in 62 ft^3 of MTO. Using best estimate values for a subset of key assumptions results in a gain of 339 ft^3 MTO for a total MTO of 401 ft^3, which is greater than 7% of the total steam generator volume. These results show that there is sufficient conservatism built into the analysis which provides assurance that the steam generators will not overfill following a SGTR event. Using best estimate assumptions regarding AFW would provide even greater MTO.

Table A.2: Sensitivity Study of Key Parameters

Case	Reference 1 Value	Best Estimate Value	Gain in Margin to Overfill
1	Tavg = 567 °F Power = 102% Secondary pressure = 841 psia	Tavg = 575 °F Power = 100 % Secondary pressure = 884 psia	137 ft^3
2	RCS pressure = 2207 psia	RCS pressure = 2250 psia	35 ft^3
3	Steam generator Level = 65 %	Steam generator level = 60 %	159 ft^3
4	AFW delay = 0 second, both pumps	AFW delay = 0 second for diesel driven pump AFW delay = 30 second for motor driven pump	54 ft^3
5	Composite Cases 1 - 4	Composite Cases 1 - 4	339 ft^3

NRC Question A.3

The original staff approved analysis was performed using RETRAN-02 MOD 3. The current analysis uses RETRAN-02 MOD 5.1. Please provide a reference for both the generic and site specific staff approval of MOD 5.1 for the application of steam generator tube rupture analysis.

Response A.3

Generic staff approval of MOD 5.1 was given in Reference 3, where the NRC found RETRAN-02 MOD 5.1 acceptable for referencing in licensing applications, subject to the same conditions and limitations as RETRAN-02 MOD 5.0. ComEd does not have site specific staff approval of MOD 5.1 for the application of SGTR analysis. The following discussions provide justification for such approval.

The RETRAN-02 MOD 5.0 computer code received an NRC SER in November, 1991 (Reference 4). The NRC found RETRAN-02 MOD 5.0 acceptable, subject to the limitations and restrictions contained in the original RETRAN-02 SER and the SERs for RETRAN-02 MOD 3 and MOD 4, and also subject to the following conditions regarding the features added in MOD 5.0:

1. With respect to each transient for which the general transport model is used, the user must justify the selected degree of mixing.

The Reference 1 steam generator tube rupture analysis does not use the general transport model.

2. When using the 1979 standard decay heat model, the user should be required to justify the associated parameter selection as presented in Section 2.2 of the SER.

The Reference 1 steam generator tube rupture analysis does not use the 1979 standard decay heat model. The 1973 standard decay heat model is used.

3. Each user should be cautioned that the reactivity components provided by the new edit feature are somewhat inexact and may be used as a qualitative indicator rather than a quantitative indicator of transient reactivity feedback.

This limitation is noted and the component reactivity information is used only qualitatively and is not used for input to other analyses.

The design basis SGTR analysis (Reference 2) included a review of the NRC SER for the intended application of using RETRAN-02 MOD 3 (Reference 8) for design analysis of a SGTR event. Since Reference 1 used the same modeling techniques and RETRAN options as Reference 2, the findings of Reference 2 remain applicable for Reference 1.

The SER review performed in Reference 2 found that the general and specific limitations cited in the RETRAN SER do not apply to the SGTR analysis, with the exception of the finding that RETRAN-02 was not generically qualified to model transients with the non-equilibrium pressurizer in a water solid or empty condition. This exception was noted because the offsite dose case performed for the Reference 2 analysis showed the pressurizer was empty for a brief period of time. For the Reference 1 analysis, the pressurizer did not go empty.

The RETRAN-02 MOD 5.1 code package was received from the Electric Power Software Center in June, 1992 and was installed on the ComEd HP 735 Workstation in accordance with approved installation procedures. The code then went through the software verification and validation process to ensure the code functioned as designed and the installation was done properly.

Additionally, a benchmark was performed comparing results between RETRAN-02 MOD 3 and RETRAN-02 MOD 5.1. The benchmark found no significant differences in results. It concluded that the previous SGTR model developed for MOD 3 can be used with MOD 5.1.

The NRC SER and ComEd software Quality Assurance program justify the use of RETRAN-02 MOD 5.1 for thermal-hydraulics analysis applications, such as the SGTR event.

NRC Question A.4

Provide additional discussion of why the amount of turbine runback is now assumed to be 70 percent rather than 60 percent in the previous analysis. Describe the effect of a turbine runback on the results of the analysis. Additionally, why is an immediate manual reactor trip not assumed if this provides more conservative results?

Response A.4

The justification is based on demonstrated operator performance. The expectation for the operator is to trip the reactor if a trip setpoint is exceeded. The observed operator performance at the ComEd simulator demonstrated this. In fact, the operators tripped the reactor without any turbine runback. Therefore, it is justified to use the conservative assumptions of turbine runback to 70% power, which is consistent with the Westinghouse Owner's Group (WOG) analysis (Reference 5).

The effect of a turbine runback is a reduction in the MTO. As the primary pressure decreases after a steam generator tube rupture, the overtemperature ΔT setpoint is approached. A turbine runback is initiated to attempt to reduce power to prevent a reactor trip on overtemperature ΔT. As the turbine runback proceeds, the turbine load is reduced. The secondary water mass increases as turbine load is reduced due to the feedwater control system's response to maintain programmed steam generator level. This process results in greater water mass in the ruptured steam generator and less MTO at the time of reactor trip.

The observed operator response time to isolate AFW is from initiation of the SGTR event. This response time is composed of two parts: time from SGTR to reactor trip and time from reactor trip to AFW isolation. If a manual reactor trip is assumed, the time of AFW isolation via operator response can be taken from the time of reactor trip. This has been shown to be less than 5 minutes (see Table B.4). Comparatively, the analysis time between reactor trip and auxiliary feedwater isolation is greater than 7 minutes, which results in more AFW going into the ruptured steam generator and a reduction in the MTO. Further, if an immediate manual reactor trip is assumed, the turbine runback assumption can be eliminated. The Reference 1 analysis assumes 5% power runback for the reactor trip on overtemperature ΔT case. Since the amount of auxiliary feedwater going into the ruptured steam generator is less and the amount of turbine runback is less, an immediate manual reactor trip case is less conservative than the reactor trip on overtemperature ΔT case and, therefore, was not assumed.

NRC Question A.5

Provide greater detail explaining why the assumption of 102 percent initial reactor power before the steam generator tube rupture (SGTR) is more conservative than the previously assumed value of 100 percent power.

Response A.5

A higher initial power leads to higher decay heat. As discussed in Reference 5, higher decay heat increases offsite dose release and decreases MTO since RCS depressurization after reactor trip is slower and primary-to-secondary leak rate is higher during the RCS cooldown and depressurization period. A higher decay heat level does increase the MTO by increasing steam release through the steam generator relief valves after the reactor trip. However, the net effect of higher decay heat is a slight reduction in MTO. Since higher initial power leads to higher decay heat, it is conservative to use 102% initial reactor power level.

NRC Question B.1

As noted in Question A.2, the margin to overfill is small. What assurance will the licensee provide to the staff that demonstration runs on the overfill scenario discussed in the submittal dated November 13, 1996, will be completed for 100 percent of Byron and Braidwood operators?

Response B.1

Reference 6 discussed previous commitments relative to demonstration runs for Byron and Braidwood station operators regarding the demonstration of operator action times for the current design basis SGTR overfill analysis (Reference 2) In brief, ComEd committed to ensuring by a specified date that "...a minimum of 80% of the Byron/Braidwood licensed operator simulator crews, comprised of active licensed shift personnel, will be evaluated on the design basis SGTR overfill scenario." The NRC staff accepted this commitment and documented their acceptance in an SER (Reference 7).

As stated in the November 13, 1996 submittal (Reference 1), necessary revisions to the station Emergency Operating Procedures (EOPs) are being implemented in early 1997. These revised procedures are integral to the reduced operator action times and will be included as part of the ongoing Licensed Operator Continuing Training cycles. Consistent with the previous commitment for demonstration runs for the design basis SGTR overfill scenario, ComEd will ensure that a minimum of 80% of the Byron/Braidwood licensed operator simulator crews, comprised of active licensed shift personnel, will be evaluated by December 31, 1997 to validate that operator action times ensure that steam generator (SG) overfill would not occur following a tube rupture event. All licensed shift personnel will be instructed in the revised EOPs when they are issued for use.

NRC Question B.2

The licensee committed to evaluate a minimum of 80 percent of the Byron and Braidwood licensed operator crews in the design basis SGTR overfill scenario, and in fact evaluated all 12 similar crews at each site. Please discuss the rationale for selecting the two licensed reactor operator crews from Byron and Braidwood for the revised SGTR overfill scenario.

Response B.2

Following the Ginna SGTR event on January 25, 1982, the SGTR subgroup of the Westinghouse Owners Group (WOG) submitted WCAP-10698, 'SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill", dated December 1984 to the NRC. A Byron and Braidwood plant specific analysis was subsequently provided to the NRC (Reference 2). As a result of a NRC Request for Additional Information (RAI), ComEd was requested to provide the basis for the operator action times assumed in the Byron and Braidwood analysis.

In response to the RAI, ComEd submitted Reference 6. In this submittal, ComEd indicated in part, "...In order to evaluate the conservatism of the operator actions assumed in the SGTR analysis, NFS directed PTD (*Production Training Department*) to evaluate two randomly selected typical operating crews each for the mitigation of the design basis SGTR." The measured action times obtained from these two typical Byron and Braidwood operator crews demonstrated that the action times assumed in the analysis (Reference 2) are realistic and achievable. The staff reviewed this response and concluded (Reference 7) that the response was acceptable.

Consistent with the ComEd response discussed above (Reference 6), ComEd selected four random crews (two from both Byron and Braidwood) to demonstrated that the action times assumed in Reference 1 are also realistic and achievable.

Additionally, a minimum of 80% of Byron/Braidwood licensed operator simulator crews, comprised of active licensed shift personnel, will be evaluated on the design basis SGTR overfill scenario by December 31, 1997. Actual operator action times obtained during the evaluation will be used to validate the analysis assumptions for operator actions stated in Reference 1 which ensures that SG overfill would not occur following a tube rupture event.

NRC Request B.3

Provide the results of a sensitivity study that would evaluate the significance of observed average response times that exceed the revised analysis response times discussed in Tables 3 and 4 of the November 16, 1996, submittal.

Response B.3

The results of a sensitivity study which evaluates the significance of observed average response times that exceed the revised analysis response times for the MTO case are included as Table B.3. The replacement steam generator case is used as the base case since it results in the limiting MTO.

Case 1 evaluates the impact of increasing operator response time to initiate RCS cooldown by 4 seconds. The impact is a reduction in MTO of 1 ft^3.

Case 2 evaluates the impact of increasing operator response time to establish charging by 1 second. The impact is a reduction in MTO of less than 1 ft^3.

Case 3 evaluates the impact of increasing operator response time to establish letdown by 1 second. The impact is a reduction in MTO of less than 1 ft^3.

A final case which incorporates all the increases in operator response times from cases 1 to 3 results in a reduction in MTO of 1.5 ft^3.

As shown in Figures 7 and 13 of Reference 1, the average rate of increase in ruptured steam generator liquid volume (or rate of reduction in MTO) does not vary significantly after the ruptured steam generator is isolated. With all other input assumptions held constant, the final transient overfill results are not sensitive to the operator response times for the individual transient phases after ruptured steam generator isolation, but are only dependent on the total operator response time required for completion of these steps. Therefore, even if an operator exceeds an individual assumed response time after the steam generator isolation step, it is not critical to the design basis SGTR analysis overall results as long as the total operator response time ensures that a SG overfill would not occur following a tube rupture event.

As shown in Figures 7 and 13 of Reference 1, the isolation of the ruptured steam generator is the most critical mitigation step for maintaining MTO since the combination of AFW flow and break flow produces the fastest steam generator fill rate. Table B.4 shows that crew #1 met the response time to isolate AFW flow by more than 250 seconds but exceeded the total response time for the remaining steps by 40 seconds. When the actual response times for crew #1 are used in place of the Reference 1 analysis times, the resultant MTO is 663 ft^3. This result demonstrates the significance of operator response to isolate AFW flow. The Reference 1 analysis AFW isolation response time assumption is very conservative since it exceeds the observed average

operator response time by more than 200 seconds and exceeds any individual crew response time by about 100 seconds.

The observed average response times exceeding the revised analysis response times has no impact on the offsite dose results. As stated in Section 3.2.3 of Reference 1, only the release during the time the ruptured steam generator power-operated relief valve (PORV) fails open is reported. Since the initiation of RCS cooldown occurs after the ruptured steam generator PORV is isolated, the reported results are not impacted. Section 3.2.3 of Reference 1 also stated that the amount of release during the RCS cooldown period is insignificant and therefore not included in the reported results. An increase in operator response time to initiate RCS cooldown by 4 seconds does not change this statement.

In summary, the impact of averaged operator response time exceeding revised analysis response time on MTO and offsite dose cases is minor. All Reference 1 conclusions remain valid. The critical operator response time requirement to isolate the ruptured steam generator is conservative when compared to operator performance. Therefore, the operator response times used in the MTO case are conservative and ensure that MTO is available after a tube rupture event.

Table B.3: Sensitivity of Operator Response Times Exceeding Analysis Response Times

Case	Reference 1 Analysis Response Time	Averaged Operator Response Time	Reduction in Margin to Overfill
1	Initiate RCS cooldown 1080 seconds	Initiate RCS cooldown 1084 seconds	1 ft^3
2	Establish charging 120 seconds	Establish charging 121 seconds	< 1 ft^3
3	Establish letdown 180 seconds	Establish letdown 181 seconds	< 1 ft^3
4	Composite Cases 1 - 3	Composite Cases 1 - 3	1.5 ft^3

NRC Request B.4

Provide the actual times for the Byron and Braidwood licensed operator crews that participated in demonstration runs for the overfill scenario discussed in the November 13, 1996, submittal.

Response B.4

The actual times for the Byron and Braidwood licensed operator crews that participated in the demonstration runs for the overfill scenario discussed in the Reference 1 submittal are provided in Table B.4.

Table B.4: Observed Operator Response Times

Response/Event	Crew # 1 (sec)	Crew # 2 (sec)	Crew # 3 (sec)	Crew # 4 (sec)	Averaged Time (sec)	Analysis Time (sec)
Reactor trip[1]	60	103	144	241	137	-
Isolate AFW to ruptured steam generator[1]	398	305	453	565	430	660
Initiate RCS cooldown	975	1212	1037	1113	1084	1080
Initiate RCS depressurization	135	122	100	37	99	120
Terminate ECCS flow (except 1 Centrifugal Charging Pump)	206	77	85	81	112	120
Establish charging	96	128	145	115	121	120
Establish letdown	205	120	193	20[illegible]	181	180
Reopen pressurizer PORV	283	150	237	2[illegible]9	237	240
Total response time excluding isolation of ruptured steam generator	1900	1809	1797	[illegible]831	1834	1860

[1] Time is from onset of steam generator tube rupture. Reactor trip time is not an operator response time used in the analysis. It is included as part of the response to NRC Question A.4.

NRC Question B.5

Provide information regarding the time anticipated to obtain steam generator sample results. Include discussion of the effect on performance of subsequent EOP steps and any resultant effect regarding the SGTR event.

Response B.5

This information was previously requested as part of the NRC staff review for the previously approved SGTR Analysis and was responded to in Reference 6. The previous response has been reviewed and determined to be accurate in all aspects to the current Byron and Braidwood EOPs and sampling procedures. This information continues to support the SER conclusion stated in Reference 7. ComEd's previous response is provided below. The staff assessment of operator action times is satisfied.

> "Under the conditions described in the Byron/Braidwood SGTR report, the maximum time for Radiation Chemistry technicians to respond and reply to the Operator's request for a secondary side sampling of all four steam generators is 3 to 4 hours after the request was made. This response time was determined in a very conservative manner. Sampling of a single suspected steam generator may be accomplished in less than 1 (one) hour.
>
> Secondary side sampling is not credited in the Byron/Braidwood SGTR analysis because sampling is only needed when steam generator narrow range level indications are not adequate. Indications from each steam generator's individual narrow range level instrument are reliable since they are class IE, Safety-Related displays readily visible on the main control panel. Tube ruptures where narrow range level indication is not sufficient to identify the steam generator with the ruptured tube must be small leaks and would not pose an overfill concern. Narrow range level indications alone give positive and clear identification of the ruptured steam generator for the MTO design basis event."
>
> "The Byron and Braidwood operating emergency procedure EP-0 directs the Operator to identify the ruptured steam generator from sequentially taken indications. The decision to enter the Steam Generator Tube Rupture procedure (EP-3) is contingent on indications, taken in sequence, first from radiation monitoring equipment, secondly from steam generator differential level indications and last from sampling of the steam generator secondary inventory for radioactivity.

Each of these identification methods can be addressed with respect to the assumptions contained in the Byron/Braidwood SGTR report.

1) Byron/Braidwood SGTR analysis does not credit use of the Steam Jet Air Ejector and S/G Blowdown radiation monitors because they are not Safety-Related. These monitors would typically be a very good, first indication for operators of primary to secondary leakage but are not credited under the SGTR design basis assumptions. The Main Steam Line radiation monitors, which are Safety-Related Technical Specification monitors, are also available for first indication for operators. One monitor per steamline is required to be operable by Technical Specifications while there is typically 2 monitors functional per steamline. Either the Main Steamline or S/G Blowdown radiation monitors may be used to identify which steam generator is ruptured.

2) Byron/Braidwood SGTR analysis credits Steam Generator Narrow Range level indication as the primary identification of which steam generator has the ruptured tube. S/G level indication is Safety-Related. Level provides direct indication of the parameters of interest. For tube ruptures of lesser primary to secondary leakage, identification of the ruptured steam generator by the differential steam generator level would become more difficult. However, lower leakages pose far less of an overfill concern. Tube ruptures of a lower primary to secondary leakage would progress at a much lower rate and the Operator would have sufficient time to respond. Procedures instruct Operators to maintain the steam generator levels between 4% and 50% narrow range by throttling auxiliary feedwater flow as needed. Under these long term conditions, any appreciable primary to secondary leakage would be detected by differing levels in the four steam generators long before overfill conditions could be reached.

3) Sampling is not credited in the Byron/Braidwood SGTR analysis because sampling is only needed when level indications are not adequate. Tube ruptures where level indication is not sufficient to identify the steam generator with the ruptured tube must be small leaks and would not pose an overfill concern. Level indications alone give positive and clear identification of the ruptured steam generator for the design basis event."

References:

1. "Revised Steam Generator Tube Rupture Analysis for Byron/Braidwood," NFSR-0114, November, 1996. Submitted via letter from John B. Hosmer (ComEd) to NRC dated November 13, 1996.

2. "Steam Generator Tube Rupture Analysis for Byron and Braidwood Plants, Revision 1," ComEd Report, March, 1990. Submitted via letter from T. K. Schuster (ComEd) to T. K. Murley (NRC), April 25, 1990.

3. "Acceptance for Referencing of the RETRAN-02 MOD 5.1 Code," letter from Martin J. Virgilio (NRC) to C. R. Lehmann (Pennsylvania Power and Light Co.), April 12, 1994.

4. "Acceptance for use of RETRAN-02 MOD 5.0," letter from Ashok C. Thadani (NRC) to W. James Boatwright (Texas Utilities Electric Company), November 1, 1991.

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

6. "Response to Request for Additional Information on Byron/Braidwood SGTR Analysis," letter from T .K. Schuster (ComEd) to T. E. Murley (NRC), January 17, 1992.

7. "Byron, Units 1 and 2, and Braidwood, Units 1 and 2 - Steam Generator Tube Rupture Analysis," SER letter from R. M. Pulsifer (NRC) to T. J. Kovach (ComEd), April 23, 1992.

8. "Acceptance for referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 regarding RETRAN 02/MOD 003 and MOD 004", letter from A.C. Thadani (NRC) to R. Funa (GPU Nuclear), October 19, 1988.

ATTACHMENT

Tables 1 and 2 in Reference 7 presented the assumptions and cumulative noble gas releases used by the staff to independently calculate the maximum offsite radiological consequences from the postulated SGTR event. To facilitate such calculations for the Reference 1 submittal, the assumptions used and iodine and noble gas releases calculated by ComEd in the Reference 1 submittal are provided in Tables C.1 and C.2, respectively.

Table C.1 - Assumptions Used for the Calculation of the Radiological Consequences Following a Postulated Steam Generator Tube Rupture Accident

Item	OSG	RSG
1. Core Power Level, Mwt	3479.22	3479.22
2. Steam Generator Leakage Prior to Event, gpm	1.0	1.0
3. Reactor Coolant Activity 3.1 Pre-Accident Spike 3.2 Accident Initiated Spike	 60 μCi/gm D.E. I_{131} UFSAR Table 15.0-8, w/ factor of 500 increase in release rate	 60 μCi/gm D.E. I_{131} UFSAR Table 15.0-8, w/ factor of 500 increase in release rate
4. Secondary Systems Initial Activity	0.1 μCi/gm D.E. I_{131}	0.1 μCi/gm D.E. I_{131}
5. RCS mass, lbm	477,740	538,361
6. SG Initial Mass, lbm	86,805	110,795
7. Offsite Power Availability	lost at time of trip	lost at time of trip
8. Total Mass Released from Ruptured SG, lbm	91,913	95,539
9. Total Mass Released from Intact SG, lbm	150,837	160,863
10. Iodine Partition Factor for Iodines Mixed in Secondary Coolant	0.01	0.01
11. 0 - 2 hour relative concentration χ/Q, sec/m^3 at EAB a. Byron site b. Braidwood site	 5.7×10^{-4} 7.7×10^{-4}	 5.7×10^{-4} 7.7×10^{-4}
12. 0 - 8 hour relative concentration χ/Q, sec/m^3 at LPZ a. Byron site b. Braidwood site	 1.7×10^{-5} 7.1×10^{-5}	 1.7×10^{-5} 7.1×10^{-5}

Table C.2 - SGTR Cumulative Iodine and Noble Gas Releases

Radionuclide	OSG Preaccident Spike (Ci)	OSG Post Accident Spike (Ci)	RSG Preaccident Spike (Ci)	RSG Post Accident Spike (Ci)
I - 131	44.46	34.38	33.19	25.78
I - 132	14.62	48.61	10.76	36.16
I - 133	70.50	74.75	52.55	56.00
I - 134	8.39	73.17	6.02	53.82
I - 135	37.95	65.34	28.19	48.87
Kr - 83m	0.00	1.68	0.00	1.42
Kr - 85	506.90	358.11	440.77	311.39
Kr - 85m	116.79	86.76	101.13	75.05
Kr - 87	61.36	50.89	52.59	43.51
Kr - 88	200.64	153.46	173.26	132.34
Kr - 89	0.00	4.21	0.00	3.26
Xe - 131m	0.00	77.32	0.00	67.23
Xe - 133	16172.80	11439.20	14060.70	9944.90
Xe - 133m	178.08	763.14	154.79	663.33
Xe - 135	356.62	257.81	309.44	223.60
Xe - 135m	6.54	13.39	5.34	10.95
Xe - 138	24.28	30.87	19.94	25.35