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January 17, 1992

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attn: Document Control Desk

Subject: Byron Station Units 1 and 2
Braidwood Station Units 1 and 2
Response to Request for Additional Information
on Byron/Braidwood SGTR Analysis
TAC #M57080/63247 and M64026/64053
NRC Docket Nos. 50-454/455 and 50-456/457

Reference: (a) April 25, 1990 letter from T.K. Schuster to Dr. T. Murley transmitting Revision 1 of the plant specific Byron/Braidwood Steam Generator Tube Rupture Analysis.

(b) August 22, 1991 Teleconference between the NRC (NRR Project Managers for B/B and Human Factors Branch) and CECo (Nuclear Licensing and Nuclear Fuel Services).

(c) November 13, 1991 Teleconference between the NRC (NRR Project Manager for Braidwood and Human Factors Branch) and CECo (Nuclear Licensing, Nuclear Fuel Services, Byron/Braidwood Plant Operating Staffs and Production Training Department).

Dear Dr. Murley:

The letter of Reference (a) provided the most recent revision of a plant specific Steam Generator Tube Rupture analysis performed by Commonwealth Edison Company (CECo) Nuclear Fuel Services for Byron/Braidwood Stations. The revision superseded the previous Revision 0 document and was provided for NRC review and approval. During the review of the Revision 1 SGTR Report the NRC Human Factors Branch indicated there was additional information required to complete their review. The information was requested in the teleconference of Reference (b). The teleconference of Reference (c) further discussed the information necessary to satisfy the review needs of the NRC Staff. The information request and subsequent CECo responses are enclosed as Attachment A to this letter.

A001

Please note that a separate CECo response dated November 15, 1991 was provided for a RAI from the Radiation Protection Branch, relative to the Revision 1 SGTR Report of Reference (a).

Please direct any questions you may have concerning this matter to this office.

Respectfully,

Terence K Schuster

T.K. Schuster
Nuclear Licensing Administrator

cc: A. Hsia NRR
R. Pulsifer-NRR
W. Kropp-Byron
S. Dupont-Braidwood
B. Clayton-RIII

Attachment A

Response to NRC RAI for B/B SGTR Analysis

References:

1) "Steam Generator Tube Rupture Analysis for Byron and Braidwood Plants", Commonwealth Edison Nuclear Fuel Services, Revision 1, March 1990

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

NRC Request #1:

Provide the basis for the Operator Action times assumed in the B/B SGTR Analysis.
- Demonstrate times assumed in the over fill scenario analysis are accurate
- Suggestion - could provide additional column of data to existing table
- Ideally, testing of all licensed operators (80-100%) could be used as a validation method

Response #1:

The NRC Safety Evaluation Report (SER) for WCAP 10698-P-A states in (Section D1), that each utility "have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis". This paragraph also states that "demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention... ...and to demonstrate that the operator action times assumed in the analysis are realistic". The SER Conclusion (Enclosure 2) states"...we require that each plant referencing WCAP-10698 demonstrate, using a plant-specific simulator and its' typical control room staff, actions and times consistent with those assumed for the WCAP-10698 design basis analysis."

Nuclear Fuel Services (NFS) reviewed the CECo Production Training Department (PTD) operator training requirements for the design basis SGTR simulator scenario, and determined they can adequately evaluate the operator performance assumed in the NFS design basis SGTR analysis (Reference 1). Since PTD coordinates both the B/B operator initial and requalification license training, the CECo program will ensure that every operator is instructed on the reference assumptions in the NFS design basis SGTR analysis. Steam Generator Tube Rupture scenarios will continue to be a part of the licensed operator initial and requalification training. These scenarios will be used periodically to evaluate licensed operators' response as part of a larger simulator training program, in a manner consistent with the changing needs of such a program.

In order to evaluate the conservatism of the operator actions assumed in the SGTR analysis, NFS directed PTD to evaluate two randomly selected typical operating crews each for the mitigation of the design basis SGTR Overfill transient case (Reference 1) and the Offsite Dose Case (Reference 1), four different crews total. The operator action times measured and assumed in the SGTR Analysis (Reference 1) are summarized in Table 1 for the Overfill case and Table 2 for the Offsite Dose Case. These times are conservative with respect to the WCAP 10698 (Reference 2) and NFS validated these assumed times based upon the guidance given in the applicable NRC SER. The analysis and simulator sequence of events for the SG Overfill case, including the operator action time intervals, are presented in Table 3. These average times demonstrate the assumed operator action times in the design basis SGTR analysis are conservative with respect to the Byron and Braidwood operator simulator mitigation performance.

In order to evaluate the significance of operator action times and to establish appropriate acceptance criteria, NFS performed a sensitivity study which determined the average effective steam generator fill rate (in ft**3 per second) does not vary significantly after the ruptured steam generator is isolated. The isolation is the most critical mitigation step, since the combination of Auxiliary Feedwater and the greater break flow for the design basis SGT[illegible]uces the fastest steam generator fill rate. NFS determined that afte[illegible]ation, the final transient overfill results are independent of the individual transient phases (cooldown, depressurization, etc.), but are only dependent on the total mitigation time required. Therefore, even if an operator exceeds an individual assumed action time (excluding SG isolation), this is not critical to the design basis SGTR analysis overfill results as long as the total operator action time is within the total assumed time frame.

NFS evaluated measured operator action times which were not within the analysis assumptions for the Overfill case (see Table 1). These action times were establishing the 70 gpm charging and the final RCS depressurization. The average time to terminate the ECCS flow and establish the 70 gpm charging flow was 1.1 (rounded from 1.08) minutes each, which slightly exceeded the assumed 1 minute operator action time. In light of the aforementioned sensitivity study NFS determined this had an insignificant effect on the transient results and was acceptable.

Also, the average measured operator action time for the final RCS depressurization step was 5.0 minutes compared to the assumed 2.0 minutes. Discussions with the Production Training Department (PTD) determined the operators utilized the auxiliary pressurizer spray instead of the PORV as assumed in the analysis. The SGTR procedure EP-3 directs the operator to complete the final RCS depressurization, using the normal pressurizer sprays first. However, if these are unavailable due to the design basis loss of offsite power, the procedure instructs the operators to utilize the auxiliary spray if available, and the PORV if necessary. During the NFS design basis overfill transient analysis, the steam generator wide range indication is offscale high, and the operator is assumed to use the PORV to ensure overfill mitigation. The pressurizer PORVs would have been used if auxiliary spray was unavailable or ineffective.

As a result of recent experience gained on the Braidwood simulator the Table 3 "Measured Simulator Event Time" for the RCS cooldown interval (difference between RCS cooldown termination and initiation) could be approximately 17 minutes versus the present 9 minutes derived from the table. The difference is due to a more conservative assumption reducing the availability of non-safety related equipment. This results in a lower temperature/pressure condition in the ruptured SG, to which the RCS must be cooled down to. Therefore, the cooldown process takes longer. The increased Measured Simulator Event RCS cooldown time is still consistent with the assumed Analysis Event Time of approximately 16 minutes. The "Total" Measured Simulator Event Time remains lower than the "Total" Assumed Analysis Event Time since an increased RCS cooldown interval can be offset by conservatism in the Measured Simulator Event Time tabulation of Table 3.

Two acceptance criteria have been established for the licensed operator design basis SGTR simulator training scenario. The first acceptance criteria requires the operators to completely isolate the ruptured SG within 16 minutes after the tube rupture occurs. The second acceptance criteria requires the operators to complete the remaining mitigation actions such that the total cumulative operator action time does not exceed 37 minutes (inclusive of the SG isolation time). The start and completion of each operator action time interval (as listed in Table 1) are defined in the SGTR analysis (Reference 1). It should be noted that the 37 minute total cumulative operator action mitigation time does not include the simulator plant response time such as the RCS cooldown time or the two RCS depressurization times.

For all other SGTR scenarios less significant than the design basis case, it would be inappropriate to place such time restrictions on the licensed operator. The successful mitigation of the event prior to SG overfill is an adequate acceptance criteria for SGTRs of lesser break flow or of lesser severity than that assumed in the Revision 1 SGTR accident analysis. The operator will not be responsible for the above acceptance criteria for scenarios which exceed the severity of the design basis event.

In summary, CECo NFS has determined the planned training program assures that the licensed operators will be instructed, and periodically evaluated on the simulator to ensure that the operator action acceptance criteria are met for the design basis SGTR (Reference 1). Also, the measured action times obtained from four typical B/B operator crews demonstrate that the action times assumed in the Reference 1 design basis SGTR analysis are realistic and achievable. However, during the 1992 & 1993 Licensed Operator Continuing Training cycles a minimum of 80% of the B/B licensed operator simulator crews, comprised of active licensed shift personnel, will be evaluated on the design basis SGTR overfill scenario. PTD will monitor and document the results of the evaluations with emphasis on actual operator response times relative to the 16 minute Steam Generator isolation and 37 minute total mitigation times assumed in the analysis. This evaluation of licensed shift operating crews will be completed by June 30, 1993. Finally, sensitivity studies have determined that time to Steam Generator isolation and the total operator mitigation time are the key factors in successfully mitigating the design basis SGTR Overfill event, as opposed to other individual action times. In conclusion, CECo NFS feels the plant specific requirements and intent of the NRC SER have been successfully demonstrated to determine the CECo SGTR analysis is consistent and conservative with respect to the WCAP 10698-P-A approved methodology.

Table 1

SGTR Overfill Case Operator Action Times

Operator Action	Assumed Time (min)	Average Measured Time (Min)
SG Isolation	16	10.5 Note 1
Establish RCS Cooldown	9	6.9
Establish RCS Depress.	4	0.3
ECCS Flow Term.	1	1.1
70 gpm Charging	1	1.1
RCS Letdown	4	3.6
Establish Final RCS Depress.	2	5.0 Note 2
Total Time	37	28.5

Note 1: Time at which operators would have isolated the steam generator, but prevented by simulator instructor to maintain design basis sequence of events.

Note 2: Operators used auxiliary pressurizer spray instead of PORV as assumed in analysis.

Table 2

SGTR Offsite Dose Case Operator Action Times

Operator Action	Assumed Time (min)	Average Measured Time (Min)
SG Isolation	20	20 Note 1
RCS Cooldown	9	2.0
RCS Depress.	4	0.3
ECCS Flow Term.	3 Note 2	1.0
Total Time	36	23.3

Note 1: The operators would have isolated the steam generator earlier, but were prevented by the simulator instructor to maintain the design basis sequence of events.

Note 2: ECCS flow termination was not explicitly modelled in the analysis.

Table 3 SGTR Overfill Sequence of Events

& Assumed/Measured Operator Action Times

Sequence of Events	Assumed Operator Action Interval (min)	Assumed Analysis Event Time (sec)	Measured Operator Action Interval (min)	Measured Simulator Event Time (sec)
SGTR Initiates		0		0
Reactor Trip		337		359
Safety Injection		347		390
AFW Initiation		347		426
SG Isolation	16	960	10.5*	630*
RCS Cooldown Initiated	9	1500	6.9**	1398
RCS Cooldown Terminated		2479		1925
RCS Depressurization Initiated	4	2719	0.3	1945
RCS Depressurization Terminated		2803		2207
ECCS Flow Terminated	1	2863	1.1	2275
70 gpm Charging Initiated	1	2923	1.1	2340
RCS Letdown Initiated	4	3163	3.6	2555
2nd RCS Depressurization	2	3283	5.0	Note 1
RCS Pressure < SG Pressure		3308		2855
Total Time (min)	37	55.1 ***	28.5	47.6 ***

* Time at which operators would have isolated the steam generator, but prevented by simulator instructor until 986 seconds to maintain design basis sequence of events.

** Since isolation was withheld until 986 seconds, cooldown initiation interval was 1398-986 which equals 6.9 minutes.

*** The simulator model is not meant to exactly replicate the engineering analysis.

Note 1-The simulator measured time for the final RCS depressurzation included both the operator action interval and the depressurization time.

Response to NRC RAI for B/B SGTR Analysis

NRC Request #2

A. Do the B/B procedures allow for Steam Generator sampling as an alternative means for identification of the ruptured Steam Generator? How long would it take to obtain a sample? - These questions apply only to the Margin To Overfill Case.

Response #2

Part A

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 B/B SGTR report.

1) B/B 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) B/B SGTR analysis credits Steam Generator Narrow Range level indication as the primary identification of which Steam Generator has the ruptured tube. 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 B/B 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.

Part B

Under the conditions described in the B/B 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 B/B 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 Margin to Overfill design basis event.