Commonwealth Edison Company
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October 1, 1997

ComEd

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Additional Information Pertaining to the Technical Specification Amendment for the Reduction in Dose Equivalent Iodine
Byron and Braidwood Units 1 and 2
NRC Docket Numbers: 50-454, 455, 456 and 457

References:

1) J. Hosmer letter to the Nuclear Regulatory Commission dated August 21, 1997, transmitting Additional Information for the Reduction in Dose Equivalent Iodine

2) J. Hosmer letter to the Nuclear Regulatory Commission dated August 22, 1997, transmitting Additional Information for the Reduction in Dose Equivalent Iodine

3) G. Stanley letter to the Nuclear Regulatory Commission dated September 2, 1997, transmitting Technical Specification Amendment Request

4) Meeting between the Nuclear Regulatory Commission and the Commonwealth Edison Company dated September 4, 1997

5) Teleconference between the Nuclear Regulatory Commission and the Commonwealth Edison Company dated September 19, 1997

References 1 and 2 transmitted the dose calculations for the exclusion area boundary, low population zone and the control room as the result of a Main Steam Line Break at Byron Unit 1. Reference 3 transmitted the same calculations for Braidwood 1.

At the Referenced meeting, the Commonwealth Edison Company (ComEd) and the Nuclear Regulatory Commission met to discuss these calculations. Subsequent to that meeting, ComEd has reviewed these calculations and has issued a revision to provide clarification along with correcting some information. The attachment provides those revisions and specifically addresses:

- For letdown and Reactor Coolant System (RCS) conditions, the appropriate density should be that of compressed water not saturated liquid. This was appropriately applied in the calculations transmitted in References 1, 2 and 3. In the revised calculation, Byron changed the let down parameters from UFSAR Chapter 9 to Chapter 11 values.

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- UFSAR Tables 15.1-4, 15.1-4b are labeled incorrectly. Table 15.1-4 should be labeled, "Activity Releases to Atmosphere from Steamline Break Accident-Unit 1," applicable to Braidwood; and UFSAR Table15.1-4b should be labeled, "Activity Releases to Atmosphere From Steamline Break Accident - Unit 2," applicable to Byron and Braidwood.
- Referencing Tables 2.F.3 and 3.F.3, the Byron and Braidwood 0-2 hour weighted activity release was corrected.
- A discrepancy between various UFSAR tables referencing primary and secondary concentrations for Iodine-132 was resolved.
- The control room dose calculation has also been revised to include the 8 hour dose for both a Main Steam Line Break (MSLB) and Loss of Coolant Accident (LOCA). Providing the dose for the same time period allows a more direct comparison of resultant doses.
- Corrections to references and variable names.

These revisions did not change the allowable leak rate value or the conclusions reached in the original calculations submitted via References 1, 2, and 3.

Additionally, the following addresses the questions raised by the Staff during the referenced teleconference.

Question 1: Does the Byron value stated on page 23, of the August 21st submittal for predicted leakage account for tube support plate as well as circumferential crack indications?

Response 1: Yes

Question 2: At what time is it assumed that the faulted steam generator is isolated following an accident?

Response 2: Primary to secondary leakage in the faulted steam generator from a MSLB is considered to be isolated when the primary side is depressurized and there is no pressure differential across the steam generator tube wall. The primary side is initially depressurized by rapid cooling from the secondary side break flow and then through the use of the steam generator PORV's on the 3 intact loops until the RH system can be started (375 psia) at approximately 8 hours. Final depressurization and therefore isolation occurs at 40 hours into the event. The primary to secondary leakage is assumed to be constant at full differential pressure (2560 psid) for the entire event until isolation.

Question 3: Is the control room make-up flow rate the same for Byron and Braidwood?

Response 3: Yes, the outside air make-up flow rate is 6000 cfm.

Question 4: Regarding page 7 of Exhibit C of the Braidwood calculation (the September 2, 1997, submittal), the Staff questioned that Reference 7 (UFSAR Table 6.4-1) does not provide the dose in the control room following a LOCA.

Response 4: ComEd's review of Reference 7 indicates that UFSAR Table 6.4-1, "Expected Dose to Control Room Personnel at Braidwood Station Following a Loss of Coolant Accident (LOCA)" is appropriate. Additionally, the revised Braidwood calculation references the correct 30 day dose from Braidwood UFSAR Table 6.4-1.

Question 5:Was Reference 5 (BWR-DIT-97-278) of the September 2, 1997, submittal transmitted to the Staff?

Response 5: Reference 5 was not transmitted. BWR-DIT-97-278 is a document that transmits information from one organization to another and contains information that is available elsewhere.

If you have any questions, please contact this office.

Sincerely,

John B. Hosmer
Engineering Vice President

Attachments: A. Byron Off-Site Dose Calculation
B. Byron Control Room Dose Calculation
C. Braidwood Off-Site Dose Calculation
D. Braidwood Control Room Calculation

cc: A. Beach, Regional Administrator - RIII
G. Dick, Byron Project Manager - NRR
D. Lynch, Senior Project Manager - NRR
S. Burgess, Senior Resident Inspector - Byron
C. Phillips, Senior Resident Inspector - Braidwood
Office of Nuclear Safety

Attachment A

Byron Off-Site Dose Calculation