Commonwealth Edison
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Downers Grove, Illinois 60515

December 19, 1990

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

Attn: Document Control Desk

Subject: Braidwood Station Units 1 and 2
Application for Amendment to Facility
Operating License NPF-72 and NPF-77
NRC Docket No. 50-456 and 50-457

Reference: TAC # 71465 and 71466

Dear Dr. Murley:

Pursuant to 10 CFR 50.90, Commonwealth Edison (CECo) proposes to amend Appendix A, Technical Specification of Facility Operating License NPF-72 and NPF-77. The proposed amendment changes the Braidwood Unit 2 heatup and cooldown curves, the power operated relief valve Low-Temperature Overpressure Protection (LTOP) setpoints, and their bases. The proposed amendment also shortens the applicability of the Unit 1 heatup and cooldown curves. These changes resulted from the NRC Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations.

A detailed description of the proposed change is presented in Attachment A. The revised Technical Specification pages are contained in Attachment B.

The proposed change has been reviewed and approved by both on-site and off-site review in accordance with CECo procedures. CECo has reviewed this proposed amendment in accordance with 10 CFR 50.92(c) and has determined that no significant hazards consideration exists. This evaluation is documented in Attachment C. An Environmental Assessment has been completed and is contained in Attachment D.

The Braidwood Unit 2 heatup and cooldown curves in the Technical Specification are valid until 2.2 Effective Full Power Years is reached. The earliest the Unit can reach this has been calculated to be June 27, 1991. Therefore, CECo requests that this proposed amendment be approved by June 1, 1991 to allow for the implementation of the LTOP setpoints and for the controller setpoint change for the PORVs.

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During the NRC review of a similar submittal for Byron Station several additional questions were asked of CECo. The answers to those questions, as they pertain to Braidwood, are contained in Attachment E.

It is permissible to distribute copies of the Letter Reports, MT/Smart 216 (88) and 078 (89), prepared by Westinghouse Electric for CECo.

CECo is notifying the State of Illinois of our application for this amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Please direct any questions regarding this matter to this office.

Very truly yours,

Allen R Checca

A. R. Checca
Nuclear Licensing Administrator

Enclosures: 1) MT-SMART-216(88)
2) MT-SMART-078(89)
3) September 12, 1990 letter from G.P. Toth to C.A. Moerke

cc: Braidwood Resident Inspector
R. Puisifer - Project Manager, NRR
Region III Office
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT A

TECHNICAL SPECIFICATION CHANGE REQUEST

Proposed Changes

The following is a brief description of the changes proposed to the Technical specifications, and the bases for these changes:

Specification 4.4.9.1.2 is being revised to address both heatup, cooldown, and cold overpressure protection setpoint figures as they exist in the Technical Specifications. These changes are administrative in nature and have no impact on the manner in which either unit is, or will be operated.

Figure 3.4-2a and 3.4-3a, Unit 1 heatup and cooldown limitations, are being revised to address shortened applicability. This change is based on an evaluation that was performed by Westinghouse to determine the applicability d:tes of the current Technical Specification curves. The results of this evaluation have shown that the heatup curve applicability must be shortened to 4.5 EFPY, while the cooldown curve is applicable for 12.0 EFPY.

Figure 3.4-2b and 3.4-3b, Unit 2 heatup and cooldown limitations, are being revised in their entirety. New curves have been generated based on the guidance given in Reg. Guide 1.99 Rev. 2. These new curves for Unit 2 will be applicable for 16 EFPY. The cold overpressure protection setpoint curve has also been revised appropriately.

Specification 3.4.9.3 has been revised to address unit specific cold overpressure protection curves. Figure 3.4-4a is the curve that has been in effect for both Unit 1 and 2 so far to date. This curve is applicable to Unit 1, and will be revised based on the evaluation to be done on Unit 1 prior to 4.5 EFPY. Figure 3.4-4b is applicable to Unit 2, and is required as a result of the reanalysis performed on Unit 2 in accordance with Reg. Guide 1.99 Rev. 2.

Specification 4.4.9.3.1.b has been revised to delete a note that has expired. This change is administrative in nature.

The bases section for specification 3/4.4.9 has been revised to address unit specific heatup and cooldown curves, and reference to requirements resulting from the reanalysis performed on Unit 2 in accordance with Reg. Guide 1.99. Revisions have been made to Reactor vessel material toughness tables B 3/4.4-1a and B 3/4.4-1b to address the nickel content for three (3) vessel weld locations found tc be limiting on both units. Table 3/4.4-16 is also being revised to correct the heat numbers. A transcription error occurred in the original submittal of the table.

Discussion

In Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, the NRC called attention to Revision 2 to Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," a document which became effective May 1988. A technical analysis was performed to determine heatup and cooldown limit curves and the reference temperature for pressurized thermal shock (PTS) based on RG 1.99 Rev. 2 methodology. Revised heatup and cooldown curves for Braidwood Unit 2 are proposed as a result of evaluating the analyses. Additionally, licensees were to evaluate the possibility for the need to revise the Low-Temperature Overpressurization (LTOP) setpoints and enable temperatures, as they are determined from the P-T limits which are revised as a result of RG 1.99 Rev. 2. The GL 88-11 stated that the Standard Review Plan (SRP) 5.2.2 "Overpressurization Protection" and the associated Branch Position RSB 5-2, in particular, paragraph II.B, was changed and might prevent overly restrictive operation based on LTOP setpoints. The revision date of the SRP was November, 1988.

Appendices G and H of 10 CFR Part 50 describes specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting pressure/temperature limits. Appendix G also requires that reactor vessel beltline materials in the surveillance capsules be tested in accordance with Appendix H. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11, requested that licensees use the methods in Regulatory Guide 1.99 Revision 2 to predict the effects of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of the unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

As a result of the revision made to Reg. Guide 1.99, revised heatup, cooldown, and cold overpressure protection curves for Braidwood Unit 2 are proposed, and the applicability date for the heatup and cooldown curves for Unit 1 have been reduced to 4.5 and 12.0 Effective Full Power Years (EFPY) respectively. The Unit 1 curves are conservative for operation until their applicability dates are reached. A Technical Specification change for the Unit 1 heatup, cooldown, and cold overpressure protection curves will be submitted prior to the Unit reaching the more restrictive of the two applicability dates. Based on current data, Unit 1 will reach 4.5 EFPY by approximately December of 1993.

Bases of the proposed Change

The changes made to the Unit 2 heatup, cooldown, and cold overpressure protection system curves are a result of two changes incorporated into Reg. Guide 1.99 Rev. 2. The analytical methods used to develop these changes are identical to those described in Byron Unit 1 & 2, Amendment 37 submitted on November 17, 1989 and approved on February 8, 1990 (TAC Nos. 71422 and 71473). The first of these two changes was a change in the methods used to determine the adjusted reference temperature (ART), and the second was a change made to the chemistry factor (CF) due to a reduced importance placed on the phosphorus term in this factor.

Unit 1

CECo evaluated the effect of neutron irradiation embrittlement on each beltline material in the Braidwood 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. As shown in enclosure 1, Table 1, page 8, CECo has determined that the material with the highest ART at 4.5 EFPY is the circumferential weld WF562 with 0.04% copper (Cu), 0.67% nickel (Ni), and an initial RT_{ndt} of 40°F.

CECo has removed one surveillance capsule from Braidwood 1. The data was published in WCAP 12685 which was submitted October 22, 1990. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld WF562, the calculated ART is 109°F at 1/4T (T = reactor vessel beltline thickness) and 84°F for 3/4T at 4.5 EFPY. Fluence values are presented in Table 2 on page 11 of enclosure 1.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. The flange reference temperature of-10°F is found in Table B 3/4.4-1a, page B3/4 4-11 in the Braidwood Technical Specification.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The unirradiated Charpy USE is 80 ft-lb for the upper to lower shell girth weld metal. This is also found in Table B3/4.4-1a of the Technical Specification.

Unit 2

CECo evaluated the effect of neutron irradiation embrittlement on each beltline material in the Braidwood 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. As shown in enclosure 2, Table 4, page 12, CECo has determined that the material with the highest ART at 16.0 EFPY is the circumferential weld (WF562) between the upper and lower shells with 0.04% copper (Cu), 0.67% nickel (Ni), and an initial RT_{ndt} of 40°F.

The licensee has removed one surveillance capsule from Braidwood 2. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal. The results of this capsule are not yet available.

For the limiting beltline material, (WF562), the calculated the ART is 146.5°F at 1/4T (T = reactor vessel beltline thickness) and 122.8°F at 3/4T at 16.0 EFPY. CECo used a neutron fluence of 9.5E18 n/cm^2 at 1/4T and 3.4E18 n/cm^2 at 3/4T. The ART was determined using Section 1 of RG 1.99, Revision 2. See enclosure 2, Table 4, page 12.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. The flange reference temperature of 20°F is found on Table 2, page 10, of enclosure 2.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The unirradiated Charpy USE is 80 ft-lb for the upper to lower shell weld metal is found on Table 2, page 10 of enclosure 2.