Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 16, 1994

Docket No. STN 50-457

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89346)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 51 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit No. 2. The amendment is in response to your application dated April 21, 1994.

The amendment effects a one-time only change to Technical Specification (TS) Surveillance Requirement 4.7.1.1 by adding a note which relieves Braidwood, Unit 2, from compliance with the provisions of TS 4.0.4 until initial entry into Mode 2. This will permit Braidwood, Unit 2, to enter Mode 3 to reset Main Steam Safety Valves (MSSVs) and proceed with a startup. This amendment is applicable only until entry into Mode 2 following forced outage A2F27.

This amendment is being issued as an exigent amendment. The same TS change was applicable to Braidwood, Unit 1 on April 18, 1994 (Amendment No. 49). It did not pertain to Unit 2 at the time because the unit was operating in Mode 1. However, Unit 2 tripped before the MSSVs could be reset. We have concluded that the need for this amendment could not have been foreseen.

Please note that this amendment does not remove the note previously added in a related amendment issued April 18, 1994, which permitted operation of Braidwood, Units 1 and 2, with an MSSV setpoint tolerance of ±3% until May 9, 1994. Any change to this note for Unit 1 or Unit 2 should be jointly requested in a separate application for a TS amendment.

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original Signed By:

Ramin R. Assa, Acting Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 51 to NPF-77
2. Safety Evaluation

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COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

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Ramin R. Assa, Acting Project Manager
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Division of Reactor Projects - III/IV
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cc w/enclosures:
See next page

Mr. D. L. Farrar
Commonwealth Edison Company

Braidwood Station
Units 1 and 2

cc:

Mr. William P. Poirier
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355, Bay 236 West
Pittsburgh, Pennsylvannia 15230

Joseph Gallo, Esquire
Hopkins and Sutter
888 16th Street, N.W., Suite 700
Washington, D.C. 20006

Regional Administrator
U. S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

Mr. Edward R. Crass
Nuclear Safeguards and Licensing
Division
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Rural Route #1, Box 79
Braceville, Illinois 60407

Mr. Ron Stephens
Illinois Emergency Services
and Disaster Agency
110 East Adams Street
Springfield, Illinois 62706

Howard A. Learner
Environmental Law and Policy
Center of the Midwest
203 North LaSalle Street
Suite 1390
Chicago, Illinois 60601

Chairman
Will County Board of Supervisors
Will County Board Courthouse
Joliet, Illinois 60434

Ms. Lorraine Creek
Rt. 1, Box 182
Manteno, Illinois 60950

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

Michael Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

George L. Edgar
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, D.C. 20036

Illinois Dept. of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Commonwealth Edison Company
Braidwood Station Manager
Rt. 1, Box 84
Braceville, Illinois 60407

EIS Review Coordinator
U.S. Environmental Protection Agency
77 W. Jackson Blvd.
Chicago, Illinois 60604-3590

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. NPF-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated April 21, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 51 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 16, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 51

FACILITY OPERATING LICENSE NO. NPF-77

DOCKET NOS. STN 50-457

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by amendment number and contains a vertical line indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

Remove Page	Insert Page
3/4 7-1	3/4 7-1

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for Braidwood, Unit 1, Cycle 5, until the initial entry into MODE 2. The provisions of Specification 4.0.4 are not applicable for Braidwood, Unit 2, until the initial entry into Mode 2 following forced outage A2F27.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BRAIDWOOD STATION, UNIT NO. 2

DOCKET NO. STN 50-457

1.0 INTRODUCTION

By letter dated April 21, 1994, Commonwealth Edison Company (CECo, the licensee) submitted an application for a License Amendment for Braidwood, Unit 2. The proposed amendment would revise Braidwood, Unit 2, Technical Specification (TS) 4.7.1.1 by relieving Unit 2 of compliance with the provisions of TS 4.0.4 until initial entry into Mode 2 following a forced outage. This one-time only change would allow Unit 2 to reach Operational Mode 3 in order to reset the lift setpoints of 17 main steam safety valves (MSSVs) which are known to have setpoint tolerances greater than the ±1% TS limit. The proposed change follows an amendment dated April 18, 1994, issued to Braidwood, Units 1 and 2, which granted Unit 2 approval to operate with a ±3% tolerance until May 9, 1994, at which time the valves were to be reset. Unit 1, in a refueling outage at the time, was granted relief from the provisions of TS 4.0.4 so that it could reach Mode 3 to reset the valves and restart. The April 18, 1994, amendment was originally requested after the licensee discovered that the as-left setpoints on certain MSSVs on both units were greater than the TS limit because the testing contractor, Furmanite, incorrectly calculated the valve mean-seat area used in the Trevitesting procedure.

The amendment issued on April 18, 1994, permitted Braidwood, Unit 2, to operate until May 9, 1994, with out-of-tolerance MSSVs. However, on April 5, 1994, Unit 2 experienced a reactor trip with complications due to a failed main power transformer, resulting in a forced outage. The plant must now restart from a cold shutdown condition. Normally, TS 4.0.4 does not allow entry into a higher operational mode if the Surveillance Requirements associated with a limiting condition for operation (LCO) have not been performed. However, since the MSSVs must be tested and set at the ambient conditions corresponding to nominal plant pressure and temperature, temporary relief from TS 4.0.4 is necessary to permit resetting the valves and to allow for the startup of
Unit 2.

It should be noted that the amendment under consideration was submitted for, and therefore applies only to, Braidwood, Unit 2. Granting of this amendment will not remove, for Unit 1 or 2, the note added to TS Table 3.7-2 in the April 18, 1994 amendment, which granted approval of a ±3% tolerance until

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April 21, 1994, which would have removed the note in Table 3.7-2, is not presently being considered. If a change to this note is desired, a separate amendment request should be submitted for the applicable dockets at a later date.

2.0 EVALUATION

The MSSVs at Braidwood were designed and manufactured as Class II components in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, 1971 edition. Testing of the valves is performed in accordance with Section XI of the ASME Code. Operability of the MSSVs ensures that secondary system pressure is limited to 110% of its design pressure of (1200 psia for Braidwood) during a turbine trip from 102% rated thermal power with no available path to the condenser (no steam dump capability). This represents the most severe anticipated operational transient. An increase on the positive side of the setpoint tolerance would potentially result in the MSSV lifting at a higher pressure, increasing the maximum pressure in the secondary system.

In its submittal of April 21, 1994, CECo assessed the safety impact of plant operation with the higher setpoint tolerance. The accident analyses considered in this application are the same as those considered in the submittal of March 21, 1994, as supplemented by a submittal of March 24, 1994, for the similar amendment issued April 18, 1994. Specifically, the licensee examined the effect of the increased MSSV setpoint tolerance on the existing licensing basis events analyses as presented in the Updated Final Safety Analysis Report (UFSAR), and concluded that the analyses remain valid with the exception of the loss-of-external load/turbine trip event. The licensee re-analyzed this event assuming the relaxed tolerance, and determined that all applicable acceptance criteria would continue to be met and that the UFSAR conclusions would remain valid. CECo concluded that the increased as-found setpoint tolerance has no significant impact on any system, operating mode, or accident analysis.

The licensee's findings are consistent with those of other similarly designed pressurized water reactor plants which have been granted relaxed setpoint tolerance for their MSSVs. These include the Seabrook, V.C. Summer, and Fort Calhoun stations, as well as the previously issued amendment for Braidwood. Additionally, Section XI of the 1989 edition of the ASME Code requires that MSSVs be tested in accordance with ASME/ANSI OM-1987, Part 1, which permits the tested setpoint pressure to exceed the nominal value by up to 3% before a test failure is declared. A higher tolerance is, therefore, consistent with recent editions of the ASME Code.

On the basis that the setpoints are within ±3%, which has been granted to other plants, including the previously issued amendment for Braidwood, Unit 1, and the relatively short duration of the proposed change (until the valves are reset in Mode 3), the staff is satisfied that the MSSVs will continue to accomplish their function with a ±3% tolerance, and that entering into Mode 3 with out-of-tolerance MSSVs involves minimal safety significance. Therefore, the staff finds the proposed temporary revision to the TS to be acceptable. It should be noted that any analyses used in support of future amendment requests for a permanent change of the setpoint tolerance are subject to further staff review.

3.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case where the staff and licensee need to act promptly, but failure to act promptly does not involve a plant shutdown, derating, or delay in startup. The exigency case usually represents an amendment involving a safety enhancement to the plant.

On April 5, 1994, a fault occurred in the main power transformer of Braidwood, Unit 2, resulting in a reactor trip. Because a control rod stuck out of the core and the transformer was severely damaged, the plant was forced into an outage.

A previously issued amendment dated April 18, 1994, permitted Unit 2 to operate until May 9, 1994, with MSSV tolerances of ±3%. However, the April 5, 1994, reactor trip prevented the licensee from resetting the valves by May 9, 1994. The valves must be set at the ambient conditions of the valve corresponding to nominal plant operating pressure and temperature. Since TS 4.0.4 prevents the plant from changing operational mode with the valves out of tolerance, the provisions of TS 4.0.4 must be temporarily waived to allow Unit 2 to reach Mode 3 to reset the valves and allow the plant to restart.

CECo currently has two large units in forced outages, six other units in outages for equipment repairs, and has outages planned for additional units prior to June 1, 1994. Delayed issuance of this amendment would prevent the startup of Braidwood, Unit 2, and, in view of the outages at other CECo facilities, could result in an inadequate supply of available power upon entry into the peak power usage months. The circumstances leading to this request for a TS amendment could not have been avoided since the licensee could not have anticipated the trip which occurred at Braidwood, Unit 2, nor was the situation created by failure of the licensee to submit a timely application for a TS amendment.

Because of the aforementioned circumstances, this amendment is being treated as an exigency, in accordance with 10 CFR 50.91. The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on May 12, 1994. The notice was published in the Joliet News Herald and the Morris Daily Herald on May 9, 1994. There were no public comments in response to the notices published in the local newspapers.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of that facility in accordance with the amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, the licensee provided its analysis of the issue of no significant hazards consideration which states that:

A. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   In the analysis performed for a ± 3% as-found MSSV setpoint, all of the applicable loss of coolant accident (LOCA) and non-LOCA design basis acceptance criteria remain valid both for the transients evaluated and the single event analyzed, Loss of External Load/Turbine Trip.

   The MSSVs are actuated after accident initiation to protect the secondary systems from overpressurization. Increasing the as-found setpoint tolerance will not result in any hardware modification to the MSSVs. Therefore, there is not an increase in the likelihood of spurious opening of a MSSV. Sufficient margin exists between the normal steam system operating pressure and the valve setpoint with the increased tolerance to preclude an increase in the probability of actuating the valves.

   The peak primary and secondary pressures remain below 110% of design at all times. The departure from nucleate boiling ratio (DNBR) and peak clad temperature (PCT) values remain within the specified limits of the licensing basis. Although increasing the valve setpoint tolerance may increase the steam release from the ruptured steam generator above the UFSAR value by approximately 2%, the steam generator tube rupture (SGTR) analysis indicates that the calculated break flow is still less than the value reported in the UFSAR. Therefore, the radiological analysis indicates that the slight increase in the steam release is offset by the decrease in the break flow such that the offsite radiation doses are less than those reported in the UFSAR. The evaluation also concluded that the existing mass releases used in the offsite dose calculation for the existing mass releases used in the offsite dose calculation for the remaining transients (i.e., steamline break, rod ejection) are still applicable. Therefore, based on the above, there is no increase in the dose releases.

The effects of increased tolerances for MSSV setpoints on the LOCA safety analyses has been previously performed for VANTAGE 5 fuel. Calculations performed to determine the response to a hypothetical large break LOCA do not model the MSSVs, since a large break LOCA is characterized by a rapid depressurization of the reactor coolant system below the pressure of the steam generators. Thus, the calculated consequences of a large break LOCA are not dependent upon assumptions of MSSV performance. Therefore, the large break LOCA analysis results are not adversely affected by revising setpoint tolerances.

The small break LOCA analyses presented in Appendix C of the Byron/Braidwood Stations, Units 1 and 2, VANTAGE 5 Reload Transition Safety Report were performed using a 3% higher safety valve setpoint pressure. The standard 3% accumulation between valve actuation and full flow was also accounted for in the analyses. These analyses calculated peak cladding temperatures well below the allowed 2200°F limit as specified in 10 CFR 50.46 demonstrating that the change to the MSSV setpoint tolerance can be accommodated for small break LOCAs.

Neither the mass and energy release to the containment following a postulated LOCA, nor the containment response following the LOCA analysis, credit the MSSV in mitigating the consequences of an accident. Therefore, changing the MSSV lift setpoint tolerances would have no impact on the containment integrity analysis. In addition, based on the conclusion of the transient analysis, the change to the MSSV tolerance will not affect the calculated steamline break mass and energy releases inside containment.

The loss of load/turbine trip event was analyzed in order to quantify the impact of the setpoint tolerance relaxation. As was demonstrated in the evaluation, all applicable acceptance criteria for this event have been satisfied and the conclusions presented in the UFSAR remain valid. The conclusions presented in the Overpressure Protection Report remain valid. Therefore, the probability or consequences of an accident previously evaluated in the UFSAR would not be increased as a result of increasing the MSSV lift setpoint as found tolerance to 3% above or below the current Technical Specification lift setpoint value.

The probability of an accident occurring will not be affected by granting this amendment request.

B. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

No new system configurations are introduced, and no equipment is being operated in a new or different manner than has been previously analyzed. Accordingly, no new or different failure modes are being created. Increasing the as-left setpoint tolerance on the MSSV does not create the possibility of an accident which is different than any already evaluated in the UFSAR. Increasing the as-left lift setpoint tolerance on the MSSVs does not introduce a new accident initiator mechanism. No new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified. No accident will be created that will increase the challenge to the MSSVs and result in increased actuation of the valves. Therefore, the possibility of an accident different than any already evaluated is not created.

C. The proposed change does not involve a significant reduction in a margin of safety.

Although the proposed amendment is requested for equipment utilized to prevent overpressurization on the secondary side and to provide an additional heat removal path, increasing the as-left lift setpoint tolerance on the MSSVs will not adversely affect the operation of the reactor protection system, any of the protection setpoints or any other device required for accident mitigation.

The proposed increase in the as-left MSSV lift setpoint tolerance will not invalidate the LOCA and non-LOCA conclusions presented in the UFSAR accident analyses. The new loss of load/turbine trip analysis concluded that all applicable acceptance criteria are still satisfied. For all the UFSAR non-LOCA transients, the departure from nucleate boiling (DNB) design basis, primary and secondary pressure limits and dose release limits continue to be met. Peak cladding temperatures remain well below the limits specified in 10 CFR 50.46. Thus, there is no reduction in the margin of safety.

The staff has completed its review of the licensee's proposed no significant hazards consideration and concludes that the amendments meet the three standards of 10 CFR 50.92(c). Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and that there has been no public comment on such finding. The proposed finding was issued in the local media described in Section 3.0 of this Safety Evaluation. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that because the requested changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: H. Dawson

Date: May 16, 1994