May 11, 1994

Docket No. 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: PUBLIC NOTICE OF APPLICATION FOR AMENDMENT TO OPERATING LICENSE FOR BRAIDWOOD NUCLEAR PLANT, UNIT 2.

The enclosed announcement has been forwarded to the Joliet News Herald and the Morris Daily Herald for publication. This announcement relates to your request for Specification (TS) Amendment for Specification 4.7.1.1, "Turbine Cycle Safety Valves" dated April 21, 1994.

A separate notice will be published later in the Federal Register concerning the revision to the TS requirements.

Sincerely,

Original signed by:

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

Enclosure:
Announcement

cc w/enclosure:
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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

PUBLIC NOTICE

NRC STAFF CONSIDERING LICENSE AMENDMENT REQUESTS FOR BRAIDWOOD, UNIT 2

The U.S. Nuclear Regulatory Commission (NRC) has received an application dated April 21, 1994, from Commonwealth Edison Company (the licensee) for an exigent amendment to Facility Operating License No. NPF-77 for the Braidwood Nuclear Plant, Unit 2, located in Braidwood, Illinois.

If approved, the amendment would revise the Surveillance Requirement associated with Braidwood, Unit 2, Technical Specification (TS) 4.7.1.1, by granting a one-time change to allow Unit 2 to reach Mode 3 in order to reset the lift setpoints of the Main Steam Safety Valves (MSSVs). Some of the MSSVs have setpoint tolerances of ±3%. The duration of this change would be until the initial entry into Mode 2 following the current forced outage, A2F27.

Normally, the TS do not allow entry into a higher operational mode if the surveillance requirements associated with a Limiting Condition for Operation (LCO) are not met. However, testing and resetting of the MSSVs must be performed at the nominal temperature and pressure corresponding to Mode 3. Temporary relief from the TS is therefore necessary to permit resetting the valves and to allow for the startup of Braidwood, Unit 2.

The proposed change follows an amendment issued March 18, 1994, which was noticed in the Federal Register on March 29, 1994 (59 FR 14685). This amendment permitted Braidwood, Unit 2, to operate until May 9, 1994, with MSSV tolerances of ±3% after the licensee was made aware that the tolerances were incorrect due to an error on the part of the contractor responsible for setting the MSSVs.

The NRC has determined that the licensee used its best efforts to make a timely application for the proposed amendment and that exigent circumstances do exist and were not the result of any fault of the licensee. The exigent circumstances arise from a reactor trip which occurred on April 5, 1994, as a result of a fault which occurred on the main power transformer. When the trip occurred, control rod K-2 of control bank B (CBB) failed to fully insert into the core. Complete insertion of all control rods is expected on a reactor trip. Due to the stuck rod and severe damage to the transformer due to the fault, Braidwood, Unit 2, was forced into an unplanned outage. Because of the time required to resolve the problems associated with the stuck rod and the time required to replace the main power transformer, Unit 2 will not be able to reset the MSSVs by the originally planned date of May 9, 1994.

The licensee has evaluated the requested amendments against the standards in 10 CFR 50.92 and the NRC staff has made a proposed (preliminary) determination that the requested amendments involve no significant hazards consideration. Under NRC regulations, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (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. The licensee's analyses are summarized below:

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 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.

Therefore, the requested amendment does not significantly increase the probability or consequences of an accident previously evaluated.

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 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.

If the proposed determinations that the requested license amendment involves no significant hazards considerations become final, the NRC will issue the amendment without first offering opportunity for a public hearing. An opportunity for a hearing will be published in the Federal Register at a later date and any hearing request will not delay the effective date of the amendments.

If the NRC staff decides in its final determinations that the amendment does involve a significant hazards consideration, a notice of opportunity for a prior hearing will be published in the Federal Register and, if a hearing is granted, it will be held before the amendment is issued.

Comments on the proposed determinations of no significant hazards considerations may be telephoned to James E. Dyer, Director, Project Directorate III-2, by collect call to 1-(301)-504-1995, or submitted in writing to the Rules and Directives Review Branch, Division of Freedom of Information and Publication Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. All comments received by close of business on May 12, 1994 will be considered in reaching a final determination. Copies of the applications may be examined at the NRC's Local Public Document Room located at the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481, and at the Commission's Public Document Room, the Gelman building, 2120 L Street, NW, Washington, DC 20555.