Commonwealth Edison
Braidwood Nuclear Power Station
Route #1, Box 84
Braceville, Illinois 60407
Telephone 815/458-2801

March 11, 1994

Mr. John Zwolinski
Assistant Director for Projects
Office of Nuclear Reactor Regulation
U.S Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Byron Station Units 1 and 2
Braidwood Station Unit 2
Request for NRR Notice of Enforcement
Discretion to Technical Specification 3.7.1.1
NRC Docket Numbers 50-454, 50-455, and 50-457

Dear Mr. Zwolinski,

The purpose of this letter is to document the results of a teleconference between Commonwealth Edison Company (CECo) and the Nuclear Regulatory Commission (NRC) staff on March 10, 1994, in which CECo requested issuance of a Notice of Enforcement Discretion (NOED) from Technical Specification 3.7.1.1 for Byron Station Units 1 and 2 and Braidwood Station Unit 2.

On March 10, 1994, at 1510, Byron Units 1 and 2 and Braidwood Unit 2 entered Technical Specification 3.7.1.1, Action Statement "a" due to inoperable Main Steam Safety Valves.

The basis of the request for Enforcement Discretion is provided in Attachment 1 and includes:

- The Technical Specification that will be violated;
- The circumstances surrounding the condition, including the need for prompt action;
- The safety basis for the request that enforcement discretion be exercised, including an evaluation of the safety significance and potential consequences of the proposed course of action;
- Any proposed compensatory measure(s);
- Justification for the duration of the request;
- The basis for the conclusion that the request will not have a potential adverse impact on the public health and safety and that a significant safety hazard is not involved, and
- The basis for the conclusion that the request will not involve adverse consequences to the environment.

A001 /1

CECo requested that Enforcement Discretion be in effect until Emergency Technical Specification Amendments are approved. Requests for Emergency Technical Specification Amendments will be submitted on March 21, 1994, for NRC review.

This request for Enforcement Discretion has been reviewed and approved by the Byron On-Site Review Committee and the Braidwood On-Site Review Committee, in accordance with station procedures.

CECo sincerely appreciates the NRC staff's effort and participation in the review of this request. Please direct any questions or comments to Denise Saccomando at (708)663-6484.

Sincerely,

Denise Saccomando

Denise Saccomando
Nuclear Licensing Administrator

Attachment

cc: R. Assa, Braidwood Project Manager-NRR
G. Dick, Byron Project Manager-NRR
S. DuPont, Senior Resident Inspector-Braidwood
H. Peterson, Senior Resident Inspector-Byron
B. Clayton, Branch Chief-Region III
NRC Document Control

ATTACHMENT 1
REQUEST FOR ENFORCEMENT DISCRETION
BRAIDWOOD AND BYRON STATION
March 10, 1994

Commonwealth Edison Company is requesting enforcement discretion for Braidwood Unit 2 and Byron Units 1 and 2 from Technical Specification 3.7.1.1.

1. TECHNICAL SPECIFICATION OR LICENSING CONDITION THAT WILL BE VIOLATED

At 1510 hours on March 10, 1994, Braidwood Unit 2 and Byron Units 1 and 2 entered Technical Specification 3.7.1.1 Action Statement "a" due to inoperable Main Steam Safety Valves (MSSVs).

Technical Specification 3.7.1.1 states that with one or more MSSVs inoperable, operation in MODES 1, 2, and 3 may proceed provided the valve is restored to OPERABLE status or the Power Range High Flux trip setpoints are reduced within 4 hours. If these requirements are not met, the plant is required to be in HOT STANDBY in 6 hours and COLD SHUTDOWN in the next 30 hours.

2. CIRCUMSTANCES SURROUNDING THE SITUATION:

The current situation rendering the main steam code safety valves inoperable was initially brought to the attention of Commonwealth Edison in a March 9, 1994 phone call from Furmanite Testing to Braidwood Station. In this call, Furmanite indicated that an improper value for mean seat area was used in the Trevitest calculation for main steam safety valve setpoints. Additional phone calls on March 10, 1994 and a letter from Furmanite revealed that the scope of the concern also included Byron Station. Calculations to determine the as-left condition of the MSSVs for each unit based on the revised mean seat area were completed at approximately 1500 hours on March 10, 1994. Results indicate that 16 valves for Byron Unit 1, 19 valves for Byron Unit 2, and 17 valves for Braidwood Unit 2 fall outside of the Technical Specification requirement of ±1%. All valves for the three units fall within ±3% of the nominal setpoint for the individual valve.

The situation requires enforcement discretion in order to prevent forced shutdown of Byron Units 1 and 2 and Braidwood Unit 2 per TS 3.7.1.1. The situation could not be avoided in that Commonwealth Edison Company was only recently informed of the situation by Furmanite based on their experience with Palo Verde.

3. EVALUATION OF SAFETY SIGNIFICANCE AND CONSEQUENCES

The requested enforcement discretion from Specification 3/4.7.1.1 due to the as-left setpoints of the MSSVs being greater than the allowed maximum of ±1% does not impact the safety margin. The MSSVs are analyzed for as-found setpoints of up ±3%, which Commonwealth Edison Company will apply for in an upcoming amendment request.

The effects of increasing the as-found lift setpoint tolerance on the MSSV have been examined, and it has been determined that, with the exception of the Loss of Load/Turbine Trip, the current accident analyses as presented in the UFSAR remain valid. The loss of load/turbine trip event was analyzed in order to quantify the impact of the setpoint tolerance relaxation. All applicable acceptance criteria for this event remain satisfied and the conclusion presented in the UFSAR remains valid. For LOCAs, neither the mass and energy release to the containment following a postulated loss of coolant accident (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 conclusions presented in the Overpressure Protection Report remain valid.

No operating conditions or modes will be changed as a result of this evaluation. No new failure modes have been determined to exist as a result of this new analysis. The MSSVs will continue to relieve any unlikely system overpressure during all applicable operating modes. The increased as-found setpoint tolerance has no significant negative impact on any system, operating mode, or accident analysis.

4. COMPENSATORY ACTIONS

Braidwood and Byron will prepare and submit an amendment request for a one time exemption to Technical Specification 3.7.1.1 for Braidwood Unit 2 and Byron Units 1 and 2. The proposed emergency Technical Specification amendment will allow the current as-left setpoints to be acceptable until they can be reset in accordance with the schedule to be provided in the request. The schedule for resetting the MSSV to within ±1% of the required setpoint will be determined by Commonwealth Edison based on availability of testing equipment from Furmanite and unit operating schedules. The amendment request will be submitted to NRC no later than 2400 hours on March 21, 1994.

In addition, for the duration of this requested enforcement discretion and until resetting of the MSSVs to within ±1% of the required setting from Table 3.7-1 of Specification 3.7.1.1, Braidwood Unit 2 and Byron Units 1 and 2 will maximize availability of Steam Dumps and Steam Generator PORVs.

5. JUSTIFICATION FOR THE DURATION OF THE REQUEST:

This enforcement discretion is requested for Braidwood Unit 2 and Byron Units 1 and 2 until NRC approval of an amendment request for a one time exemption to Technical Specification 3.7.1.1 to allow as-left setpoint tolerance to exceed ±1%. The proposed amendment, which will apply to both Byron and Braidwood, will be submitted no later than 2400 hours on March 21, 1994, and will also address the restart of Braidwood Unit 1 from its current refueling outage.

The proposed emergency Technical Specification amendment will allow the current as-left setpoints to be acceptable until they can be reset in accordance with the schedule to be provided in the request. The schedule for resetting the MSSV to within ±1% of the required setpoint will be determined by Commonwealth Edison based on availability of testing equipment from Furmanite and Unit operating schedules.

6. EVALUATION OF SIGNIFICANT HAZARD CONSIDERATION:

Commonwealth Edison has evaluated the proposed enforcement discretion and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed 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.

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

In 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 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 the 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 primary below the pressure of the steam generator secondaries. 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 accomodated 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 enforcement discretion.

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

b. **The proposed enforcement discretion 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 enforcement discretion does not involve a significant reduction in a margin of safety.**

Although the proposed enforcement discretion 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 to safety.

Based on the review above, Braidwood and Byron conclude that this request for enforcement discretion does not involve a significant hazards consideration.

7. ENVIRONMENTAL ASSESSMENT:

Braidwood and Byron have evaluated the proposed enforcement discretion against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion as provided for under 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as enforcement discretion to a license issued pursuant to 10 CFR 50, and that the change requested involves extension of the Allowed Outage Time of a component located within the restricted area, and the change involves no significant hazards. There is no change in effluents that may be released offsite. There is no significant increase in individual or cumulative occupational radiation exposure.

8. APPROVAL BY ON SITE REVIEW:

This request has been reviewed and approved by the Braidwood On-site Review Committee and the Byron On-Site Review Committee, in accordance with the respective station procedures.