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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 22 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 22 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. 50-454, 50-455, 50-456, AND 50-457

1.0 INTRODUCTION

By letter dated December 4, 1987 (Ref. 1), the Commonwealth Edison Company (CECo), the licensee, requested a modification to Figure 3.2-2 in the Technical Specifications for the Byron/Braidwood power stations. Additional information was submitted on October 26, 1988 (Ref. 2). Figure 3.2-2 (known as the K(z) curve) represents the normalized heat flux hot channel factor as a function of core height. The modification was proposed after a reanalysis of the small break loss of coolant accident, which was performed as part of the licensee's T_{hot} reduction program.

The T_{hot} reduction program was being performed under the provisions of 10 CFR 50.59 and is a program to reduce the primary system vessel outlet temperature (T_{hot}) from 618.4°F to 600°F, a reduction of 18.4°F from the original design. The purpose of this reduction is to reduce the potential for initiation and propagation of primary water stress corrosion cracking in the steam generators for the Byron/Braidwood Stations, Units 1 and 2.

The K(z) curve has three distinct segments which define the limits for K(z) at various core axial positions. The third segment is that portion at the higher core locations defined by the points (10.8, 0.94) and (12.0, 0.65). This portion represents the limiting K(z) values assumed in the small break loss of coolant accident (SBLOCA).

During a postulated SBLOCA, only the higher elevations can potentially become uncovered. The appropriate value of K(z) is used to calculate local hot rod power at a particular core elevation. This in turn determines the location of peak cladding temperature (PCT) during the postulated SBLOCA event. CECo performed a new analysis at the limiting condition. This made it possible to eliminate the third line segment. Greater heat flux values could then be obtained at higher axial positions and the limits on the core design for hot channel factors could be relaxed.

The CECo amendment request proposed to eliminate the third line segment and, in place, extend the segment defined by the points (6.0, 1.0) and (10.8, 0.94) to the 12 foot core elevation, which defined a new point (12.0, 0.925) as shown on the attached Figure 3.2-2. CECo provided a description of the analyses supporting their amendment for change of Figure 3.2-2.

In Reference 1, Attachment B, revised marked-up pages of Chapter 15.6.5 of the Byron/Braidwood FSAR were provided as background information (proprietary report, WCAP-11386, Revision 2 (Ref. 3)). The staff reviewed brief section in WCAP-11386 pertaining to the change requested for Figure 3.2-2 in the Technical Specifications.

2.0 EVALUATION

In the large break loss of coolant accident (LBLOCA) and SBLOCA reanalysis the approved NOTRUMP (Ref. 4) and BASH (Ref. 5) computer codes were used respectively. The following parameters were revised as indicated below in the new analysis:

	Original	Revised
Total Peaking Factor (F_Q)	2.32	2.40
Enthalpy Rise Peaking Factor ($F_{\Delta H}$)	1.55	1.62
Steam Generator Tube Plugging (uniform)	0%	10%
Safety Injection Flow	-	5% reduction
K(z) curve	-	eliminate third line segment

The increase in F_Q and $F_{\Delta H}$ were only addressed in the LOCA analysis.

In response to a request for additional information relating to Figure 3.2-2, the K(z) curve, CECo submitted a letter dated October 26, 1988 (Ref. 2) describing the methodology of the hot rod power shape for the large and small break LOCA to satisfy the requirements of 10 CFR 50.46, Appendix A. The licensee stated that the representative small break shape in the NRC approved WCAP-9500-A was used as well as additional criteria regarding power distribution mechanisms which govern peak clad temperature for the small break. CECo also provided a revised figure to show the correct hot rod power shape for the SBLOCA. For the LBLOCA, the licensee stated that Westinghouse had demonstrated that the most limiting power shape was being used in the 1981 evaluation model using the BASH large break LOCA analyses. Studies, as presented in Revision 1 of Addendum 1 to WCAP-10266-P-A, Revision 2, for F_Q peaking factors equal to

2.32 and 2.50 showed the chopped cosine power shape to be the most limiting. CECo stated that the chopped cosine power shape was used in the analysis for Byron/Braidwood and that the peaking factor used was 2.40, which is bounded by the above studies reported in WCAP-10266-P-A. Since the peaking factor is bounded by the previous studies, it is acceptable. However, the Technical Specifications will retain the current overall LOCA peaking factor (F_Q) of 2.32 until all of the pertinent FSAR non-LOCA analyses have been repeated at the higher F_Q value of 2.40.

The staff had requested that the licensee provide a new figure similar to their revised Figure 3.2-2 with the results of the analysis imposed for the power shape. This was to include curves of linear heat generation rate (KW/ft) versus elevation (ft) including core average and hot rod values and the K(z) limit for the SBLOCA. CECo submitted such a figure (Ref. 3) which provided a means to observe if the submitted Figure 3.2-2 K(z) was supported by the results of the analysis. The staff found the modified Figure 3.2-2 K(z) curve to be in agreement with the results of the analysis.

The results of the large and small break LOCA as shown in Tables 15.6-3 and 15.6-4 Reference 2 indicated that the maximum peak fuel element clad temperatures were 1754°F and 1630°F, respectively. This is well within the 2200°F limit specified in 10 CFR 50.46. Also, the total Zr/H_2O reactor was less than 0.3 percent for both the large and small break LOCA which is less than the 1 percent limit specified in 10 CFR 50.46.

3.0 CONCLUSIONS FOR MODIFICATION OF FIGURE 3.2-2

The staff has found the modification made to the Figure 3.2-2 K(z) curve to be acceptable since the analyses on which it is based used acceptable codes, and the results of these analyses meet the criteria of 10 CFR 50.46:

(1) The calculated peak fuel element clad temperature is below 2200°F; (2) the amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of zircaloy in the reactor; (3) the total oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation; (4) the core remains amenable to cooling during and after the break; (5) the core temperature is reduced and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

4.0 TECHNICAL SPECIFICATIONS

The Technical Specifications were changed as follows:

Figure 3.2-2, K(z)-normalized $F_Q(Z)$ is a function of core height, was modified for both the Byron and Braidwood plants. The third line segment was eliminated. In place of the third line segment, the segment defined by the points (6.0, 1.0) and (10.8, 0.94) was extended to the 12 foot elevation, defined by the new point (12.0, 0.925) as shown on the attached figure. The modification was found to be acceptable as explained in Sections 2.0 and 3.0.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, an environmental assessment and finding of no significant impact have been prepared and published (54 FR 40547) in the Federal Register on October 2 , 1989. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from K. A. Ainger, Commonwealth Edison Company (CECo) to NRC, dated December 4, 1987.

2. Letter from R. A. Chrzanowski, CECo to T. E. Murley, NRC, dated October 26, 1988.

3. WCAP-11386, Revision 2, "Byron/Braidwood T_{hot} Reduction Final Licensing Report," dated November 1987.

4. Meyer, P. E., NOTRUMP, A Nodal Transient Small Break General Network Code, WCAP-10079-P-A, August 1985 (Proprietary).

5. Kabadi, J. N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266, Revision 2, with addenda (Proprietary), August 1986.

Principal Contributor: H. Balukjian

Dated: October 4, 1989

October 4, 1989

Docket Nos. 50-454, 50-455
and 50-456, 50-457

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Kovach:

The Commission has issued the enclosed Amendment No. 34 to Facility Operating License No. NPF-37 and Amendment No. 34 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively and Amendment No. 22 to Facility Operating License No. NPF-72, and Amendment No. 22 to Facility Operating License No. NPF-77 for Braidwood Station, Units Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 4, 1987.

These amendments approve changes to Technical Specification Figure 3.2-2 which depicts the normalized heat flux hot channel factor as a function of core height.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be filed with the Office of the Federal Register for publication.

Sincerely,

Stephen P. Sands, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V, and Special Projects
Office of Nuclear Reactor Regulation

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V, and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 34 to NPF-37
2. Amendment No. 34 to NPF-66
3. Amendment No. 22 to NPF-72
4. Amendment No. 22 to NPF-77
5. Safety Evaluation

cc w/enclosures:
See next page

[BYRON/BRAIDWOOD AMEND]

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