February 12, 1992

Docket Nos. STN 50-455
and STN 50-457

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M80872 AND M80873)

The Commission has issued the enclosed Amendment No. 45 to Facility Operating License No. NPF-66 for the Byron Station, Unit No. 2, and Amendment No. 34 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit No. 2. The amendments are in response to your application dated June 28, 1991.

The amendments revise Technical Specification Tables 2.2-1 and 3.3-4 for Model D-5 steam generator (SG) low-low and high-high level instrumentation. The changes are in accordance with a plant modification to be performed during the next refueling outage for each unit. The modification relocates the lower SG level instrument tap from 438 inches to 333 inches as measured from the top of the SG tube sheet.

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:

Anthony H. Hsia, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 45 to NPF-66
2. Amendment No. 34 to NPF-77
3. Safety Evaluation

NRC FILE CENTER COPY

cc w/enclosures:
See next page

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DOCUMENT NAME: [AMENDMENT M80872/73]

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Mr. Thomas J. Kovach
Commonwealth Edison Company

Byron/Braidwood Power Stations

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. NPF-66

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated June 28, 1991 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 1;

   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-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 45 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is 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 of its issuance and is to be implemented by April 20, 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 12, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. NPF-66

DOCKET NO. STN 50-455

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
2-5	2-5
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop minimum measured flow*	≥89.2% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low					
a. Unit 1	27.1	18.28	1.5	≥40.8% of narrow range instrument span	≥39.1% of narrow range instrument span
b. Unit 2	N.A	N.A	N.A.	≥36.3% of narrow range instrument span	≥35.4% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	12.0	0.7	0	≥5268 volts - each bus	≥4728 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	13.3	0	≥57.0 Hz	≥56.5 Hz
16. Turbine Trip					
a. Emergency Trip Header Pressure	N.A.	N.A.	N.A.	≥540 psig	≥520 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.

*Minimum measured flow = 97,600 gpm

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-2	7.7	0.71	1.5	≤8.2 psig	≤9.2 psig
d. Steam Line Pressure-Low (Above P-11)	21.2	14.81	1.5	≥640 psig*	≥617 psig*
e. Steam Line Pressure Negative Rate-High (Below P-11)	8.0	0.5	0	≤100 psi**	≤111.5 psi**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)					
1) Unit 1	6.0	4.28	1.5	≤81.4% of narrow range instrument span	≤82.7% of narrow range instrument span
2) Unit 2	18.9	12.02	3.2	≤80.8% of narrow range instrument span	≤82.8% of narrow range instrument span

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (continued)					
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump					
1) Unit 1	27.1	18.28	1.5	≥40.8% of narrow range instrument span	≥39.1% of narrow range instrument span
2) Unit 2	N.A.	N.A.	N.A.	≥36.3% of narrow range instrument span	≥35.4% of narrow range instrument span
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump	N.A.	N.A.	N.A.	≥5268 volts	≥4728 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34
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 June 28, 1991, 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 1;

   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. 34 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 of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

R J Barrett

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 12, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 34

FACILITY OPERATING LICENSE NO. NPF-77

DOCKET NO. STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
2-5	2-5
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
-	3/4 3-26a

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	$\geq$90% of loop minimum measured flow*	$\geq$89.2% of loop minimum measured flow*
13. Steam Generator Water Level Low-Low					
a. Unit 1	27.1	18.28	1.5	$\geq$40.8% of narrow range instrument span	$\geq$39.1% of narrow range instrument span
b. Unit 2	17.0 (Cycle 3) N.A. (Cycle 4 and after)	14.78 (Cycle 3) N.A. (Cycle 4 and after)	1.5 (Cycle 3) N.A. (Cycle 4 and after)	$\geq$17% (Cycle 3); $\geq$36.3% (Cycle 4 and after) of narrow range instrument span	$\geq$15.3% (Cycle 3); $\geq$35.4% (Cycle 4 and after) of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	12.0	0.7	0	$\geq$5268 volts - each bus	$\geq$4728 volts - each bus
15. Underfrequency - Reactor Coolant Pumps	14.4	13.3	0	$\geq$57.0 Hz	$\geq$56.5 Hz
16. Turbine Trip					
a. Emergency Trip Header Pressure	N.A.	N.A.	N.A.	$\geq$540 psig	$\geq$520 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	$\geq$1% open	$\geq$1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	N.A.	N.A.

*Minimum measured flow = 97,600 gpm

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-2	7.7	0.71	1.5	≤8.2 psig	≤9.2 psig
d. Steam Line Pressure-Low (Above P-11)	21.2	14.81	1.5	≥640 psig*	≥617 psig*
e. Steam Line Pressure Negative Rate-High (Below P-11)	8.0	0.5	0	≤100 psi**	≤111.5 psi**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)					
1) Unit 1	6.0	4.28	1.5	≤81.4% of narrow range instrument span	≤82.7% of narrow range instrument span
2) Unit 2	5.0 (Cycle 3) 18.9 (Cycle 4 and after)	2.18 (Cycle 3) 12.02 (Cycle 4 and after)	1.5 (Cycle 3) 3.2 (Cycle 4 and after)	≤78.1% (Cycle 3); ≤80.8% (Cycle 4 and after) of narrow range instrument span	≤79.9% (Cycle 3); ≤82.8% (Cycle 4 and after) of narrow range instrument span

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (continued)					
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-Low-Low-Start Motor-Driven Pump and Diesel-Driven Pump					
1) Unit 1	27.1	18.28	1.5	$\geq$40.8% of narrow range instrument span	$\geq$39.1% of narrow range instrument span
2) Unit 2	17.0 (Cycle 3) N.A. (Cycle 4 and after)	14.78 (Cycle 3) N.A. (Cycle 4 and after)	1.5 (Cycle 3) N.A. (Cycle 4 and after)	$\geq$17% (Cycle 3); $\geq$36.3% (Cycle 4 and after) of narrow range instrument span	$\geq$15.3% (Cycle 3); $\geq$35.4% (Cycle 4 and after) of narrow range instrument span

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (SE)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (continued)					
d. Undervoltage-RCP Bus-Start Motor Driven Pump and Diesel-Driven Pump	N.A.	N.A.	N.A	$\geq$5268 volts	$\geq$4728 volts
e. Safety Injection-Start Motor-Driven Pump and Diesel-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-66

AND AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NO. 2

BRAIDWOOD STATION, UNIT NO. 2

DOCKET NOS. STN 50-455 AND STN 50-457

1.0 INTRODUCTION

In a submittal dated June 28, 1991, the Commonwealth Edison Company (CECo) described reactor protection system (RPS) and engineered safety features actuation system (ESFAS) trip setpoint changes resulting from lowering of the lower narrow range steam generator (SG) level instrument taps at Byron and Braidwood No. 2 Units from 438 inches above the top of the SG tubesheet to 333 inches above the tubesheet. The upper instrument tap for the Model D-5 SGs in the Byron and Braidwood No. 2 Units design remains unchanged at 566 inches above the tubesheet. With the changes, the narrow range SG level instrument taps for the No. 2 Units will be at the same levels as those in the No. 1 Units which have Model D-4 SGs.

The submittal also provided an assessment of the impact of the changes on FSAR Chapter 15 analyses, and proposed Technical Specification (TS) changes to reflect the modifications.

2.0 STAFF EVALUATION

2.1 Setpoint Changes

The Byron and Braidwood TSs express the SG water level low-low and high-high trips in terms of percent of narrow range SG water level instrument span (NRS). The increase in the narrow range instrument span alters the correlation of level expressed in inches versus level expressed in percent of span. Also included in the consideration of revised setpoints is velocity head. Velocity head effects result in indicated levels for any given power less than or equal to the actual level, with the amount of discrepancy varying directly but not proportionally with power.

The high-high and low-low SG level trip setpoints for the Byron and Braidwood No. 2 Units TSs account for the above considerations, and are based on consistency with safety analysis assumptions and with the setpoint methodology described in the Westinghouse Topical Reports WCAP-12583 and WCAP-12523.

This methodology, incorporating the above considerations, has been used in previous Byron and Braidwood applications. Since the basic methodology has not been changed for this use, we also find it applicable to Byron and Braidwood No. 2 Units for this determination of setpoints.

2.2 Chapter 15 Analyses

2.2.1 Non-LOCA Event Analyses

The submittal provided an assessment of the impact of the changes on Final Safety Analysis Report (FSAR) Chapter 15 analyses. For most Chapter 15 events, the licensee found that the calculated results for existing Byron and Braidwood Updated FSAR analyses, performed assuming Model D-4 SGs, either would be unaffected by the changes or would remain bounding versus analyses assuming the modified Model D-5 SGs and associated trip settings.

Three Chapter 15 events were reanalyzed because of the potential for adverse effect due to the changes. The events are:

a. Feedwater System Malfunction Causing an Increase in Feedwater Flow (UFSAR Section 15.1.2)

   This event was reanalyzed for both zero power and full power case conditions. The zero power case was found to be bounded by the Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Condition event addressed in UFSAR Section 15.4.1. For the full power case, the licensee reports that the calculated minimum departure from nucleate boiling ratio (MDNBR) remains above the safety analysis limit value throughout the transient. This assures that departure from nucleate boiling (DNB) would not be encountered and that calculated event consequences meet criteria of acceptance.

b. Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature (UFSAR Section 15.1.1)

   The licensee's submittal indicates that this event is bounded by the Increase in Feedwater Flow event discussed above, and the DNB basis is met.

c. Loss of Non-Emergency AC Power to the Plant Auxiliaries/Loss of Flow (UFSAR 15.2.6/15.2.7)

   The licensee's submittal indicates that these reanalyses calculated that pressurizer overfill would not occur for these events and verified the natural circulation capability of the plants to prevent fuel or cladding damage during reactor pump coastdown.

The analyses of the above events were performed using the same methods as those for the UFSAR Chapter 15 event analyses of record. We find that these methods continue to be applicable.

2.2.2 Steam Generator Tube Rupture (SGTR)

The licensee's submittal indicated that the SGTR consequences reported in the Byron and Braidwood UFSAR will not be increased by the D-5 SG modifications. The staff concludes that the finding of acceptability for the SGTR analysis of record continues to apply.

Submittal Section 4.3.2 discusses additional SGTR analyses performed by the licensee which were submitted to the NRC for review. Because this analysis is still under review, we have not considered that analysis in our findings.

2.2.3 LOCA Analyses

The licensee's submittal indicates that LOCA analyses are not adversely affected by the changes because analysis assumptions are not changed. We find this acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

The licensee's submittal proposed changes to three TS pages to be implemented in the operating cycle after SG modification for each unit (Byron Unit 2 and Braidwood Unit 2) to reflect the setpoint modifications discussed in Section 2.1 of this report. These are:

a. TS page 2-5, Table 2.2-1, Item 13.b, SG Water Level Low-Low Reactor Coolant System (RCS) trip - Values for Total Allowance (TA), parameters not measured on a periodic basis (Z), and Sensor Error (SE) are identified as not applicable (N.A.). The new Trip Setpoint is 36.3% of NRS and the new Allowable Value is 35.4% of NRS.

b. TS page 3/4 3-25, Table 3.3-4, Item 5.b.2, SG Water Level High-High Turbine Trip and Feedwater Isolation - Values for TA, Z, and SE are increased to 18.9, 12.02, and 3.2, respectively. The new Trip Setpoint is 80.8% of NRS and the new Allowable Value is 82.8% of NRS.

c. TS page 3/4 3-26, Table 3.3-4, Item 6.c.2, SG Water Level Low-Low Start Motor-Driven Pump and Diesel-Driven Pump - The new values are the same as in a. above.

d. Because Braidwood, Unit 2, fuel cycle 3 began in November 1991, the proposed TS change and the corresponding modifications to relocate the lower sensing tap of the Unit 2 steam generator will not be effective until the start of fuel cycle 4. Therefore, for Braidwood, Unit 2, the above proposed TS changes will only be effective for cycle 4 and after. The existing TS will remain through cycle 3.

The licensee's submittal based its justification of these modified setpoints on consistency with FSAR Chapter 15 analyses assumptions as discussed in Section 2.2 of this report.

We find the licensee's submittal, describing lowered lower SG level instrumentation taps, associated trip setpoint changes, and analytical justifications, acceptable based on use of a setpoint methodology which had been previously used in an approved application, and on justifications citing applicable UFSAR analyses and reanalyses using approved methodologies.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments 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. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 57692). 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 amendments.

6.0 CONCLUSION

The Commission 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, (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: F. Orr, SRXB

Date: February 12, 1992