Mr. D. L. Farrar July 27, 1995
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M89092, M89093, M89072 AND M89091)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 73 to Facility Operating License No. NPF-37 and Amendment No. 73 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 65 to Facility Operating License No. NPF-72 and Amendment No. 65 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to Commonwealth Edison Company's (ComEd) application dated March 23, 1994, as supplemented on July 26, 1994, and subsequently superseded by a submittal dated February 15, 1995. The February 15, 1995, request was supplemented on February 28, 1995.

The amendments revise the technical specifications (TSs) to include a maximum moderator temperature coefficient (MTC) of +7 pcm/°F and move the control of the cycle specific MTC from the TSs to the operating limits report. Our review also approves the methodology proposed by ComEd for ensuring that the plants continue to meet the anticipated transient without scram (ATWS) rule (10 CFR 50.62) during operation with cycle specific MTCs.

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

George F. Dick Jr., Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 73 to NPF-37
2. Amendment No. 73 to NPF-66
3. Amendment No. 65 to NPF-72
4. Amendment No. 65 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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

July 27, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
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SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M89092, M89093, M89072 AND M89091)

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Sincerely,

George F. Dick, Jr.

George F. Dick Jr., Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 73 to NPF-37
2. Amendment No. 73 to NPF-66
3. Amendment No. 65 to NPF-72
4. Amendment No. 65 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. NPF-37

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated February 15, 1995, as supplemented February 28, 1995, 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-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 73 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr.

George F. Dick Jr., Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1995

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.73
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 February 15, 1995, as supplemented on February 28, 1995, 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. 73 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 shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr.

George F. Dick Jr., Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 73 AND 73

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND 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. The page indicated by an asterisk is provided for convenience only.

Remove Pages	Insert Pages
IV	IV
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
--	3/4 1-5a
*3/4 1-6	*3/4 1-6
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
6-22	6-22
--	6-22a

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3/4.1.1 BORATION CONTROL	
Shutdown Margin - T_{avg} > 200°F	3/4 1-1
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FIGURE 3.1-0 MODERATOR TEMPERATURE COEFFICIENT VERSUS POWER LEVEL	3/4 1-5a
Minimum Temperature for Criticality	3/4 1-6
3/4.1.2 BORATION SYSTEMS	
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Position Indication Systems - Operating	3/4 1-17
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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Operating Limits Report (OLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

APPLICABILITY: Beginning of Life (BOL) limit - MODES 1 and 2* only#.
End of Life (EOL) limit - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the BOL limit specified in the OLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the OLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the MTC more negative than the EOL limit specified in the OLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.
#See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the predicted MTC to establish administrative rod withdrawal limits, as necessary, to assure that the BOL limit specified in the OLR, is met throughout core life, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading, and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the OLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the OLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the OLR, at least once per 14 EFPD during the remainder of the fuel cycle.

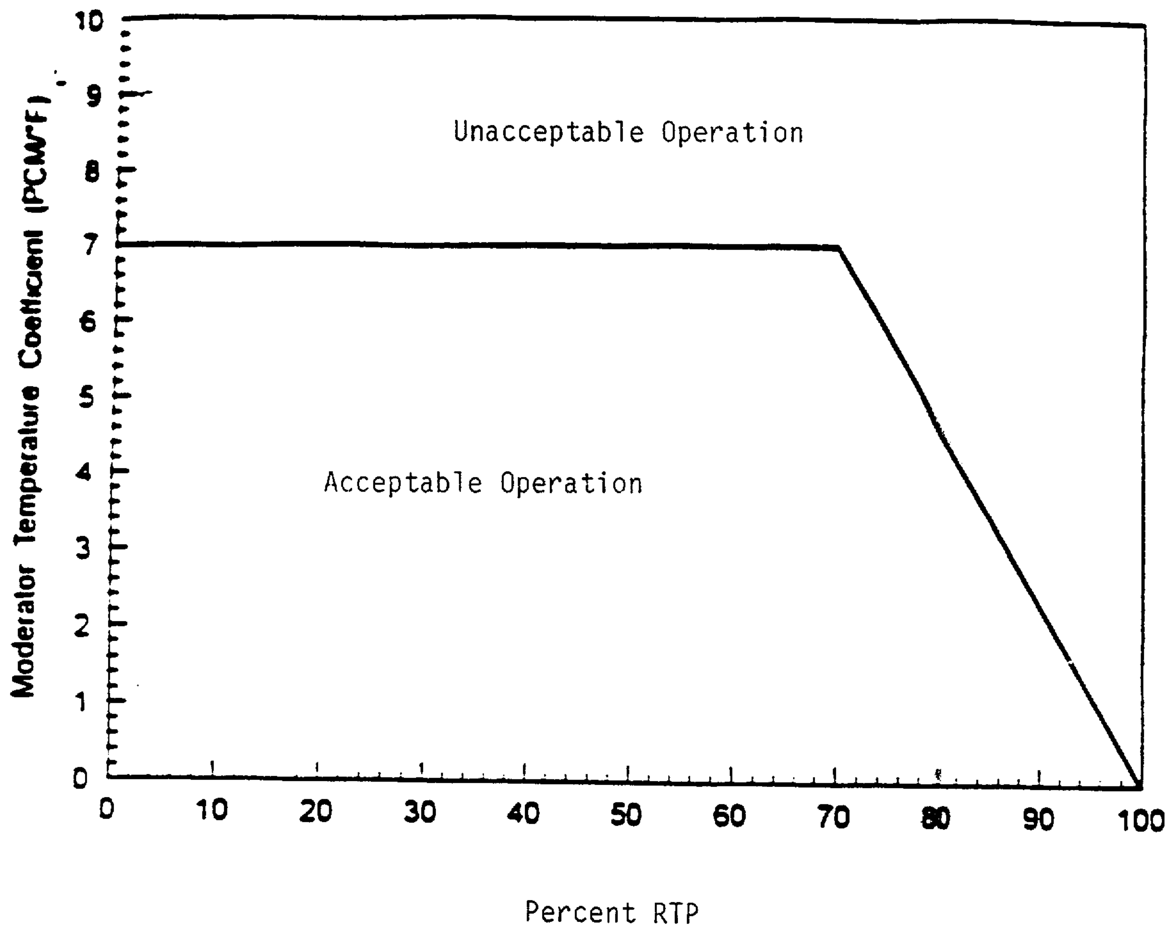

FIGURE 3.1-0

MODERATOR TEMPERTURE COEFFICIENT vs. POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 550°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 550°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 550°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 557°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.

*See Special Test Exceptions Specification 3.10.3.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection provided that boration dilution paths are isolated. A 1.3% $\Delta k/k$ SHUTDOWN MARGIN is required to ensure the OPERABILITY of the automatic Boron Dilution Protection System.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the UFSAR accident and transient analyses. The limitations on MTC also ensure that the Anticipated Transient Without Scram (ATWS) risk is acceptable. A cycle specific Unfavorable Exposure Time (UET) value will be calculated to ensure < 5% of the cycle operations occur when the reactivity feedback is not sufficient to prevent exceeding an ATWS overpressurization condition of ≥ 3200 psig in the RCS. This UET value will be updated for each core reload and appropriately considers the effects of changes in MTC, including any variations that are more adverse than those originally modeled in the analyses supporting the basis for the final ATWS rule.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the UFSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting EOL MTC value specified in the OLR. The 300 ppm surveillance limit represents a conservative value (with corrections for burnup and soluble boron) with an equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement, combined with the predicted MTC throughout core life, will be used to impose administrative limits on rod withdrawal, as required during core life to ensure that MTC will always remain within the limits specified in the OLR. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement is 13,487 (15,780*) gallons of 7000-ppm borated water from the boric acid storage tanks or 54,014 (70,450*) gallons of 2300-ppm (2000-ppm*) borated water from the refueling water storage tank. A Boric Acid Storage System level of 40% ensures that there is a volume of greater than or equal to 13,487 (15,780*) gallons available. A RWST level of 89% ensures that there is a volume of greater than or equal to 395,000 gallons available.

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*Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.

REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 A Radioactive Effluent Release Report covering the operation of the facility during the previous year shall be submitted prior to May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

OPERATING LIMITS REPORT

6.9.1.9 Operating limits shall be established and documented in the OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in Topical Reports:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluations Methodology" dated July 1985.
2. WCAP-8385, "Power Distribution Control and Load Following Procedures-Topical Report" dated September 1974.

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*A single submittal may be made for a multi-unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

OPERATING LIMITS REPORT (Continued)

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods" dated July 1983.
4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," dated July 1990.
5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Controls Systems."

The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
License No. NPF-72

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated February 15, 1995, as supplemented on February 28, 1995, 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-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 65 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1995

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. 65
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 February 15, 1995, as supplemented on February 28, 1995, 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. 65 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 65 AND 65

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND 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. The page indicated by an asterisk is provided for convenience only.

Remove Pages	Insert Pages
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3/4 1-5	3/4 1-5
--	3/4 1-5a
*3/4 1-6	*3/4 1-6
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
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--	6-22a

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Operating Limits Report (OLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

APPLICABILITY: Beginning of Life (BOL) limit - MODES 1 and 2* only#.
End of Life (EOL) limit - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the BOL limit specified in the OLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the OLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the MTC more negative than the EOL limit specified in the OLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.
#See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL predicted MTC to establish administrative rod withdrawal limits, as necessary, to assure that the BOL limit specified in the OLR, is met throughout core life, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading, and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the OLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the OLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the OLR, at least once per 14 EFPD during the remainder of the fuel cycle.

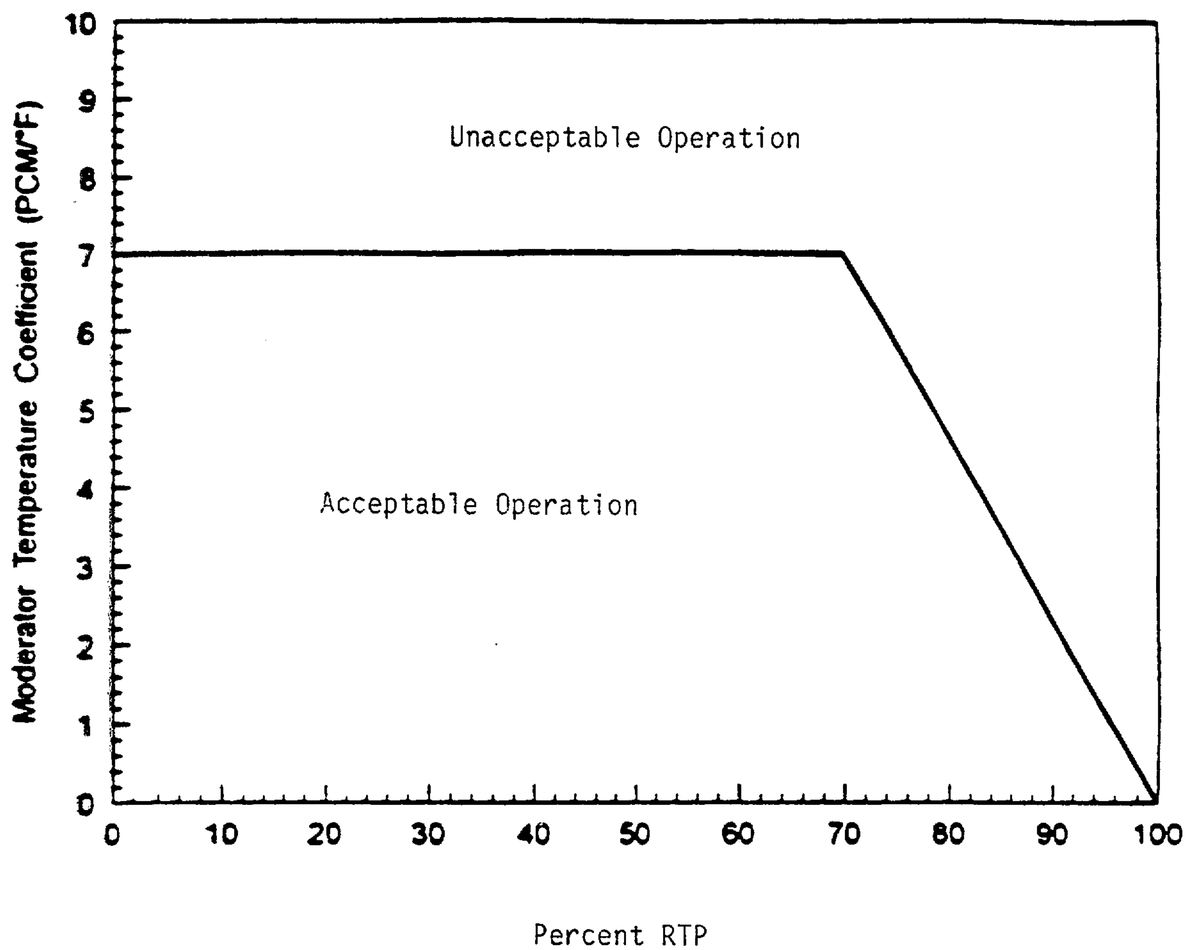

FIGURE 3.1-0

MODERATOR TEMPERTURE COEFFICIENT vs. POWER LEVEL

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 550°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 550°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 550°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 557°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.

*See Special Test Exceptions Specification 3.10.3.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection provided that boration dilution paths are isolated. A 1.3% $\Delta k/k$ SHUTDOWN MARGIN is required to ensure the OPERABILITY of the automatic Boron Dilution Protection System.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the UFSAR accident and transient analyses. The limitations on MTC also ensure that the Anticipated Transient Without Scram (ATWS) risk is acceptable. A cycle specific Unfavorable Exposure Time (UET) value will be calculated to ensure < 5% of the cycle operations occur when the reactivity feedback is not sufficient to prevent exceeding an ATWS overpressurization condition of ≥ 3200 psig in the RCS. This UET value will be updated for each core reload and appropriately considers the effects of changes in MTC, including any variations that are more adverse than those originally modeled in the analyses supporting the basis for the final ATWS rule.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the UFSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting EOL MTC value specified in the OLR. The 300 ppm surveillance limit represents a conservative value (with corrections for burnup and soluble boron) with an equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement combined with the predicted MTC with core burnup, can be used to impose administrative limits on rod withdrawal to ensure that MTC will always remain within the limits specified in the OLR. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement is 15,780 (13,487)* gallons of 7000-ppm borated water from the boric acid storage tanks or 70,450 (54,014)* gallons of 2000-ppm (2300-ppm)* borated water from the refueling water storage tank.

*Applicable to Unit 1 and Unit 2 starting with Cycle 6.

REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 A Radioactive Effluent Release Report covering the operation of the facility during the previous year shall be submitted prior to May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

OPERATING LIMITS REPORT

6.9.1.9 Operating limits shall be established and documented in the OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in Topical Reports:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluations Methodology" dated July 1985.

*A single submittal may be made for a multi-unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

OPERATING LIMITS REPORT (Continued)

2. WCAP-8385, "Power Distribution Control and Load Following Procedures-Topical Report" dated September 1974.
3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods" dated July 1983.
4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," dated July 1990.
5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Controls Systems."

The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

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

1.0 INTRODUCTION

By letter of March 23, 1994, Commonwealth Edison Company (ComEd, the licensee) proposed changes to the technical specifications (TSs) for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The submittal requested changes to the TSs to: (1) support plant operating cycles designed with a positive moderator temperature coefficient (PMTC); and (2) change the core thermal design flow requirements to support an increase in the steam generator tube plugging limit to 15 percent. The portion of the request related to the thermal design flow requirements was granted by the staff in Amendments 65 for Byron and Amendments 56 and 55 for Braidwood, Units 1 and 2, respectively, on October 21, 1994. This Safety Evaluation addresses the portion of the licensee's request related to the PMTC. The March 23, 1994, submittal was supplemented on July 26, 1994, and subsequently superseded by a submittal dated February 15, 1995. The February 15, 1995, request was supplemented on February 28, 1995.

The proposed moderator temperature coefficient (MTC) change would permit the licensee to maintain the MTC value within the limits specified in a cycle specific operating limits report with a maximum upper limit specified in the TSs of +7 pcm/°F for lower power operation. The current TSs require a zero or negative MTC at the beginning of cycle, hot, zero power and with all control rods withdrawn. The current requirement had been reviewed with regard to its effects on design basis accidents and is compatible with previous Westinghouse anticipated transients without scram (ATWS) analyses. A more positive MTC would not necessarily continue to fall within the bounds indicated in those analyses and, thus, required further licensee analysis and staff review.

The analysis of the effects of the PMTC on the relevant transients and accidents, as provided by the licensee in the March 23, 1994, submittal, covered the standard events normally considered in Chapter 15 of the Updated Final Safety Analysis Report and was found to be acceptable by the staff. However, the staff found that the analysis of effects of the proposed PMTC on ATWS and the relationship to the basis for the ATWS rule was not sufficiently addressed.

The staff concerns regarding the ATWS aspects of the PMTC were transmitted to the licensee on July 26, 1994. Subsequently there were several communications with ComEd including letters (August 16 and December 21, 1994) and a public meeting on September 15, 1994. A February 15, 1995, amendment request which superseded the previous request included revisions to the proposed TSs, the safety analysis for the changes, and a description of methodology to be used in the calculation of the MTC for each cycle to assure that the MTC to be specified will be compatible with the Westinghouse analyses performed during the development of the basis for the ATWS rule.

With regard to meeting the ATWS rule, ComEd first proposed to use the methodology and supporting bases described in Westinghouse Topical Reports WCAP-11992, "ATWS Rule Administration," and WCAP-11993, "Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs," both dated December 1988, to provide the technical justification for the higher MTC. The methodology presented in WCAP-11993 is a probability analysis of parameters important for ATWS, which had been developed during a previous review of PMTC interaction with the analysis included in the basis of the ATWS rule. While the report had been submitted previously, it was for information only, had not been previously reviewed and, therefore, could not provide a basis for justification of the proposed TS change without an extensive review. In order to expedite the review schedule, the licensee proposed a deterministic approach as described in WCAP-11992 to justify the specific MTC for each operating cycle.

2.0 EVALUATION

The processes described in WCAP-11992 are based on calculations submitted to the NRC in a Westinghouse letter dated December 30, 1979, "ATWS Submittal," which presented analyses of ATWS events and sensitivity studies for significant parameters and provided the Westinghouse contribution to the basis for the ATWS rule.

The approach used by ComEd focuses on two aspects of the WCAP-11992 methodology to provide a comparison to the Westinghouse ATWS analyses. These are the Unfavorable Exposure Time (UET) and critical trajectory methodologies. The critical trajectories are calculated loci of plant conditions (e.g., power vs. inlet temperature) which provide a peak pressure in the transient analysis of the limiting ATWS event which is then compared to the specified limit (3200 psig). The UET is the time during the cycle when reactivity feedback is insufficient to maintain pressure under 3200 psig for a given reactor state. Information for the trajectory and UET calculations was derived from the ATWS submittal and WCAP-11992. Again, in order to focus the review only on the immediate licensee request, the extent of staff review of WCAP-11992 required to evaluate this approach was restricted to relevant Sections 4.3.8, 4.6.8 and B.7.1.

In summary, in the application of the UET methodology, the ATWS transient point kinetics information is transferred into steady state conditions for comparison with cycle specific core condition evaluation calculations and the critical trajectories are determined. During peak ATWS pressure conditions heat-up is relatively slow so that steady state analysis is acceptable. The licensee's methodology uses the "base case" conditions from the "ATWS Submittal," with 100 percent power-operated relief valve (PORV) capacity, 100 percent auxiliary feed water and no control rod insertion. The cycle specific calculations are done with appropriate ATWS initial conditions of full power, rods out, equilibrium xenon and 3200 psig pressure. Criticality is determined as a function of inlet temperature. These calculations are compared to the critical trajectory from the transient analyses. This comparison provides cycle specific design conditions which would result in transient conditions exceeding 3200 psig. These calculations show any core design conditions which would result in exceeding the 3200 psig pressure limit. Calculations as a function of the time in cycle and, thus, as a function of MTC, show the time during the cycle that the core design critical trajectory is greater than the transient trajectory. From this, the UET is determined. The analysis must show that the UET, given the cycle design (including MTC), will be less than 5 percent, or equivalently, that ATWS pressure limit will be met for at least 95 percent of the cycle. If the limit is not met the core design would be changed until the 95 percent level is achieved.

This 95 percent probability level for the UET is equivalent to the probability level in the reference analyses for the ATWS rule basis. In those analyses, staff requirements were that all parameters should be best estimate values with the exception of the MTC initial condition. That was to be at a level not to be exceeded (i.e., not less negative) at full power conditions for at least 95 percent of the cycle. The ComEd approach provides a similar level of assurance for the effectiveness of the reactivity feedback.

This proposed analysis provides an acceptable justification for the requested change in the allowed MTC limit with respect to ATWS requirements. The initial review found the justification acceptable for all other analyses and, therefore, the changes requested by the licensee may be made to the MTC limit.

3.0 TECHNICAL SPECIFICATIONS

The following TS and Bases changes were proposed. The changes were justified or follow from methodology or analysis presented in the submittal, as discussed above and are all directly related to the specification of a PMTC.

The changes for both Byron and Braidwood are as follows:

(1) Table of contents; Figure 3.1-0 added.

(2) TS 3.1.1.3; Moderator Temperature Coefficient, was changed to show that the MTC limits are now to be specified in the Operating Limits Report (OLR), with a maximum upper limit given in Figure 3.1-0 of the TS. This provides for a +7 pcm/°F coefficient approved based on an analysis of transients during the initial portion of the staff review and now based on the licensee's methodology for meeting the ATWS rule. There are changes to the Action statements to indicate the limits are now in the OLR. Similar changes referring to the OLR are also in the surveillance section. The movement of MTC limits to the OLR is within staff approved practice.

(3) The corresponding Bases 3/4.1.1.3 is augmented to discuss the ATWS aspects of the MTC and the UET, as well as the use of the OLR.

(4) TS 6.9.1.9; Operating Limits Report, has a change in format and the addition of a reference to discuss the methodology.

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TSs to be included as part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

In evaluating previous proposals to move cycle-specific parameters from the TSs to a licensee controlled document, the staff described (Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits from Technical Specifications) an acceptable alternative to keeping the values of cycle specific parameters in the TSs which consisted of three separate actions to modify the plants TSs: (1) the addition, into a formal report, of the values of cycle-specific parameter limits that have been established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis; (2) an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information; and (3) the modification of individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the formal report. The proposed changes satisfy all three of the actions for an acceptable alternative.

In the evaluation of this alternative, the NRC staff concluded that it is essential to safety that the plant is operated within the bounds of cycle-specific parameter limits and that a requirement to maintain the plant within the appropriate bounds must be retained in the TSs. However, the specific values of these limits may be modified by licensees, without affecting nuclear safety, provided that these changes are determined using NRC-approved methodology (specified in the TS) and consistent with all applicable limits of the plant safety analysis that are addressed in the updated Final Safety Analysis Report. Additionally, it was concluded that a formal report should be submitted to the NRC with values of these limits. This will allow continued trending of this information, even though prior NRC approval of the changes to these limits would not be required.

The current method of controlling reactor physics parameters to assure conformance to 10 CFR 50.36 is to specify the specific value(s) determined to be within specified acceptance criteria (usually the limits of the safety analyses) using an approved calculation methodology. The alternative contained in the licensee's proposal controls the values of cycle-specific MTC and assures conformance to 10 CFR 50.36, by specifying the calculation methodology and acceptance criteria in the TSs. This permits operation at any specific value determined by the licensee, using the specified methodology, within the acceptance criteria. The Operating Limits Report will document the specific values of parameter limits resulting from licensee's calculations including any mid-cycle revisions to such parameter values.

The licensee's proposal provides a satisfactory means of controlling these parameters and is acceptable. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the licensee's OLR.

4.0 SUMMARY

The staff has reviewed the information submitted by ComEd for Byron and Braidwood to justify proposed TS changes which provide for changes to the requirements or limits for the moderator temperature coefficient and has concluded that appropriate information was submitted and that the proposed changes to the TS are acceptable.

5.0 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.

6.0 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 and change surveillance requirements. The amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. 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 (60 FR 18623). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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.

7.0 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: H. Richings

Date: July 27, 1995