September 24, 1991

Docket Nos. STN 50-454, STN 50-455
and STN 50-456, 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. 79724, 79725, 79726 AND 79727)

The Commission has issued the enclosed Amendment No. 43 to Facility Operating License No. NPF-37 and Amendment No. 43 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 32 to Facility Operating License No. NPF-72 and Amendment No. 32 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated January 26, 1990, as supplemented April 8, 1991.

The amendments would revise the Action statement for Technical Specification 3.1.3.1, Moveable Control Assemblies, and the associated Bases section. The revision adds an Action statement to address the condition when more than one full-length control rod is inoperable but still capable of insertion into the core upon receipt of a reactor trip signal. The associated Bases section is, therefore, expanded to cover this new Action statement.

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. 43 to NPF-37
2. Amendment No. 43 to NPF-66
3. Amendment No. 32 to NPF-72
4. Amendment No. 32 to NPF-77
5. Safety Evaluation

cc w/enclosures:
See next page

OFFICIAL RECORD COPY
DOCUMENT NAME: [AMENDMENT 79724/5/6/7]

Office:	LA/PDIII-2	PD/PDIII-2	PM/PDIII-2	SRXB	PD/PDIII-2	OGC-WF1	OTSB
Surname:	CMoore	RPulsifer	AHsia:ta	RJones	RBarrett		CGRIMES
Date:	6/4/91	7/23/91	6/12/91	7/19/91	9/9/91	9/6/91	9/6/91

DFol
with comments
CP

Mr. Thomas J. Kovach
Commonwealth Edison Company

Byron/Braidwood Power Station
Unit Nos. 1 and 2

cc:
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Braidwood Station Manager
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
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 January 26, 1990, as supplemented April 8, 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 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. 43 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.

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: September 24, 1991

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. 43
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 January 26, 1990, as supplemented April 8, 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. 43 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.

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: September 24, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 43 AND 43

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. Overleaf pages identified by an asterisk are provided for convenience.

Remove Pages	Insert Pages
*3/4 1-13	*3/4 1-13
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
*3/4 1-16	*3/4 1-16
*B 3/4 1-3	*B 3/4 1-3
B 3/4 1-4	B 3/4 1-4

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water level of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F, and

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

    1. The rod is restored to OPERABLE status within the above alignment requirements, or

    2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

    3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

        a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

        b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

c. With more than one full-length rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the group(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods shall be restored to OPERABLE status within 72 hours.

Otherwise, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics.

Rod Cluster Control Assembly Misalignment.

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System.

Single Rod Cluster Control Assembly Withdrawal at Full Power.

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident).

Major Secondary Coolant System Pipe Rupture.

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection).

BORATION SYSTEMS (Continued)

A Boric Acid Storage System level of 40% ensures that there is a volume of greater than or equal to 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.

With the RCS temperature below 350°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR Suction valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,652 gallons of 7000-ppm borated water from the boric acid storage tanks or 11,840 gallons of 2000-ppm borated water from the refueling water storage tank (RWST). A Boric Acid Storage System level of 7% ensures there is a volume of greater than or equal to 2652 gallons available. An RWST level of 9% ensures there is a volume of greater than or equal to 38,740 gallons available.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and

MOVABLE CONTROL ASSEMBLIES (Continued)

18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Inoperability or misalignment of a single, trippable rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by an inoperable or misaligned rod are reevaluated to confirm that the results remain valid during future operation. With multiple inoperable or misaligned, but trippable, rods; alignment of the remaining rods in the bank(s) to within $\pm$ 12 steps of the inoperable rods, and restriction in THERMAL POWER assures fuel rod integrity during continued operation.

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent on the plant to confirm trippability of the inoperable rod(s). This confirmation may be, for example, by verification of a control system failure, usually electrical in nature (such as an Urgent Failure Alarm), or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 550°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if a rod position deviation monitor is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
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 January 26, 1990, as supplemented by April 8, 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 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. 32 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.

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: September 24, 1991

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. 32
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 January 26, 1990, as supplemented April 8, 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. 32 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.

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: September 24, 1991

ATTACHMENT TO LICENSE AMENDMENT NOS. 32 AND 32

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. Overleaf pages identified by an asterisk are provided for convenience.

Remove Pages	Insert Pages
*3/4 1-13	*3/4 1-13
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
*3/4 1-16	*3/4 1-16
*B 3/4 1-3	*B 3/4 1-3
B 3/4 1-4	B 3/4 1-4

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water level of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F, and

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

c. With more than one full-length rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group step counter demand height by more than $\pm$ 12 steps (indicated position), POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the group(s) with the inoperable rods are aligned to within $\pm$ 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods shall be restored to OPERABLE status within 72 hours.

Otherwise, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics.

Rod Cluster Control Assembly Misalignment.

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System.

Single Rod Cluster Control Assembly Withdrawal at Full Power.

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident).

Major Secondary Coolant System Pipe Rupture.

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection).

BORATION SYSTEMS (Continued)

A Boric Acid Storage System level of 40% ensures that there is a volume of greater than or equal to 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.

With the RCS temperature below 350°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 330°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR Suction valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,652 gallons of 7000-ppm borated water from the boric acid storage tanks or 11,840 gallons of 2000-ppm borated water from the refueling water storage tank (RWST). A Boric Acid Storage System level of 7% ensures there is a volume of greater than or equal to 2652 gallons available. An RWST level of 9% ensures there is a volume of greater than or equal to 38,740 gallons available.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and

MOVABLE CONTROL ASSEMBLIES (Continued)

18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Inoperability or misalignment of a single, trippable rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by an inoperable or misaligned rod are reevaluated to confirm that the results remain valid during future operation. With multiple inoperable or misaligned, but trippable, rods; alignment of the remaining rods in the bank(s) to within ± 12 steps of the inoperable rods, and restriction in THERMAL POWER assures fuel rod integrity during continued operation.

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent on the plant to confirm trippability of the inoperable rod(s). This confirmation may be, for example, by verification of a control system failure, usually electrical in nature (such as an Urgent Failure Alarm), or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 550°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if a rod position deviation monitor is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 32 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 dated January 26, 1990, as supplemented by letter dated April 8, 1991, Commonwealth Edison Company (CECo) proposed an amendment to the Technical Specifications (TSs) for both Byron Station, Unit Nos. 1 and 2 and Braidwood Station, Unit Nos. 1 and 2. The proposed amendment would revise the Action statement for TS 3.1.3.1, Moveable Control Assemblies, and the associated Bases section. The revision adds an Action statement to address the condition when more than one full-length control rod is inoperable but still capable of insertion into the core upon receipt of a reactor trip signal. The associated Bases section is, therefore, expanded to cover this new Action statement. The April 8, 1991, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The proposed changes revise the Action statement for TS 3.1.3.1, Moveable Control Assemblies. The revision deletes the original Action statement (b) and adds an Action statement (c) to address the condition when more than one full-length control rod is trippable, but inoperable or misaligned from its group step counter demand height by more than ±12 steps. The revised Action statement permits continued power operation only if the remainder of the rods in the group(s) with the inoperable rods are aligned to within ±12 steps in 1 hour and the inoperable rods are restored to operable status within 72 hours. Otherwise, be in hot standby within 6 hours. The 72 hours provide the flexibility for diagnosis and repair of the inoperable but trippable rods, thereby, possibly eliminating an unnecessary reactor shutdown.

In most cases when more than one full-length control rod is trippable, aligned, but inoperable, the malfunction can be attributed to the rod control system since the design of the Control Rod Drive System (CRDS) assures that the essential elements of the CRDS required for reactor trip are isolated from the rod control system. Therefore, with multiple rods inoperable or misaligned, but trippable, the CRDS can still perform their intended safety function to trip the reactor.

The maximum control rod misalignment directly affects core power distribution and assumption of available shutdown margin. The revised TS requires that for continued operation, the remainder of the control rods be realigned with the inoperable rods to within ±12 steps in 1 hour. Thus, the local xenon redistribution during this short interval will not be significant. The proposed amendment does not alter the allowed maximum rod misalignment of ±12 steps and, therefore, there is no impact on any accident analysis assumptions.

The associated Bases section changes provide clarifications to the reasons for the revised Technical Specification and differentiate between the single and multiple inoperable or misaligned, but trippable control rods. The revision also places the responsibility on the plant to confirm the trippability of the inoperable rods or take action according to Action statement (a) of TS 3.1.3.1.

In summary, the proposed revisions for the TS and the associated Bases section will result in greater flexibility to the operation of the plant while maintaining safety of the unit.

3.0 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.

4.0 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 13660). 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.

5.0 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: A.H. Hsia

Date: September 24, 1991