September 27, 1988

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

DISTRIBUTION:

Docket Files	BGrimes
NRC & Local PDRs	TBarnhart (16)
PDIII-2 Rdg. File	WJones
DMuller	EButcher
GHolahan	NTrehan
LLuther	JStewart
LOlshan	ACRS (10)
SSands	GPA/PA
OGC-Rockville	ARM/LFMB
DHagan	EJordan
PDIII-2 Plant File	

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

The Commission has issued the enclosed Amendment No. 23 to Facility Operating License No. NPF-37 and Amendment No.23 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 12 to Facility Operating License No. NPF-72, and Amendment No.12 to Facility Operating License No. NPF-77 for Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated January 18, 1988.

These amendments approved changes that remove two tables from the Technical Specification which list reactor trip system instrumentation response times and engineered safety features response times. These two tables are being placed in the Byron/Braidwood Final Safety Analysis Report.

These amendments also changed the bases on page B 2-5 to explain that the Source Range, Neutron Flux trip provides primary protection for the core in Modes 3, 4 and 5.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/S/ /S/

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

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

Enclosures:
1. Amendment No. 23to NPF-37
2. Amendment No. 23to NPF-66
3. Amendment No.12 to NPF-72
4. Amendment No.12 to NPF-77
5. Safety Evaluation

cc: w/enclosures
See next page

PDIII-2:PM	PDIII-2:PM	PDIII-2:LA	OGC-Rockville	PDIII-2:PD
LOlshan:km	SSands	LLuther	RBachmann	DMuller P.S. FOR
9/7/88	9/8/88	9/19/88	9/20/88	9/19/88

8810130263 880927
PDR ADOCK 05000454
P PDC

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

September 27, 1988

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

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

The Commission has issued the enclosed Amendment No.23 to Facility Operating License No. NPF-37 and Amendment No.23 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 12 to Facility Operating License No. NPF-72, and Amendment No.12 to Facility Operating License No. NPF-77 for Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated January 5, 1988.

These amendments approved changes that remove two tables from the Technical Specification which list reactor trip system instrumentation response times and engineered safety features response times. These two tables are being placed in the Byron/Braidwood Final Safety Analysis Report.

These amendments also changed the bases on page B 2-5 to explain that the Source Range, Neutron Flux trip provides primary protection for the core in Modes 3, 4 and 5.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Stephen P. Sands

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

Leonard N. Olshan

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

Enclosures:
1. Amendment No.23 to NPF-37
2. Amendment No.23 to NPF-66
3. Amendment No.12 to NPF-72
4. Amendment No.12 to NPF-77
5. Safety Evaluation

cc: w/enclosures
See next page

Mr. Henry Bliss
Commonwealth Edison Company

Byron/Braidwood

cc:

Mr. William Kortier
Atomic Power Distribution
Westinghouse Electric Corporation
Post Office Box 355
Pittsburgh, Pennsylvania 15230

Joseph Gallo, Esq.
Hopkins and Sutter
1050 Connecticut Ave., N.W.
Suite 1250
Washington, D. C. 20036

C. Allen Bock, Esquire
Post Office Box 342
Urbana, Illinois 61801

Regional Administrator
U. S. NRC, Region III
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

Mr. Edward R. Crass
Nuclear Safeguards and Licensing
Division
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR#1, Box 79
Braceville, Illinois 60407

Dr. Bruce von Zellen
Department of Biological Sciences
Northern Illinois University
DeKalb, Illinois 61107

U. S. Nuclear Regulatory Commission
Byron/Resident Inspectors Office
4448 North German Church Road
Byron, Illinois 61010

Ms. Lorraine Creek
Rt. 1, Box 182
Manteno, Illinois 60950

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

Douglass Cassel, Esq.
109 N. Dearborn Street
Suite 1300
Chicago, Illinois 60602

Ms. Pat Morrison
5568 Thunderidge Drive
Rockford, Illinois 61107

David C. Thomas, Esq.
77 S. Wacker Drive
Chicago, Illinois 60601

Elena Z. Kezelis, Esq.
Isham, Lincoln & Beale
Three First National Plaza
Suite 5200
Chicago, Illinois 60602

cc:

Mr. Charles D. Jones, Director
Illinois Emergency Services
and Disaster Agency
110 East Adams Street
Springfield, Illinois 62706

Mr. Michael C. Parker, Chief
Division of Engineering
Illinois Department of
Nuclear Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Michael Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

George L. Edgar
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, D.C. 20036

Commonwealth Edison Company
Byron Station Manager
4450 North German Church Road
Byron, Illinois 61010

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
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 5, 1988, 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 Specification 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:

8810130264 880927
PDR ADOCK 05000454
P PDC

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 23 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

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 27, 1988

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
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 5, 1988, 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-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. 23 and revised by Attachment 2 to NPF-60, and the Environmental Protection Plan contained in Appendix B, both of which are 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

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 27, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS.23 AND 23

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

DOCKET NOS. STN-50-454 AND STN 50-455

Revise Appendix A as follows:

Remove Pages	Insert Pages
V	V
B 2-5	B 2-5
3/4 3-1	3/4 3-1
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-13	3/4 3-13
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
B 3/4 3-2	B 3/4 3-2

SECTION PAGE

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE..... 3/4 2-1
FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER..... 3/4 2-3
3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR..... 3/4 2-4
FIGURE 3.2-2 K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT... 3/4 2-5
3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR..... 3/4 2-8
3/4.2.4 QUADRANT POWER TILT RATIO..... 3/4 2-10
3/4.2.5 DNB PARAMETERS..... 3/4 2-13
TABLE 3.2-1 DNB PARAMETERS..... 3/4 2-14

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION..... 3/4 3-1
TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION..... 3/4 3-2
TABLE 3.3-2 (THIS TABLE IS NOT USED)..... 3/4 3-7
TABLE 4.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS..... 3/4 3-9
3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION..... 3/4 3-13
TABLE 3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION..... 3/4 3-15
TABLE 3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS..... 3/4 3-23
TABLE 3.3-5 (THIS TABLE IS NOT USED)..... 3/4 3-30
TABLE 4.3-2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS..... 3/4 3-34

Power Range, Neutron Flux, High Rates (Continued)

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the limit value.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. Both of these trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels in MODE 2 while the Source Range, Neutron Flux trip provides primary protection for the core in MODES 3, 4 and 5. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-2

(THIS TABLE IS NOT USED)

TABLE 3.3-2 (Continued)

(THIS TABLE IS NOT USED)

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $$Z + RE + SE \leq TA$$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   RE = The "as measured" value (in percent span) of rack error for the affected channel,

   SE = Either the "as measured" value (in percent span) of the sensor error, or the value for Column SE (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

TABLE 3.3-5

(THIS TABLE IS NOT USED)

TABLE 3.3-5 (Continued)

(THIS TABLE IS NOT USED)

TABLE 3.3-5 (Continued)

(THIS TABLE IS NOT USED)

TABLE 3.3-5 (Continued)

(THIS TABLE IS NOT USED)

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) essential service water pumps start and automatic valves position.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
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 5, 1988, 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 Specification 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. 12 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

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 27, 1988

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
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 5, 1988, 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 Specification 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. 12 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 27, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 12 AND 12

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

DOCKET NOS. STN-50-456 AND STN 50-457

Revise Appendix A as follows:

Remove Pages	Insert Pages
V	V
B 2-5	B 2-5
3/4 3-1	3/4 3-1
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-13	3/4 3-13
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
B 3/4 3-2	B 3/4 3-2

SECTION PAGE

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE..... 3/4 2-1
FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER..... 3/4 2-3
3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR..... 3/4 2-4
FIGURE 3.2-2 K(Z)-NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT... 3/4 2-5
3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR..... 3/4 2-8
3/4.2.4 QUADRANT POWER TILT RATIO..... 3/4 2-10
3/4.2.5 DNB PARAMETERS..... 3/4 2-13
TABLE 3.2-1 DNB PARAMETERS..... 3/4 2-14

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION..... 3/4 3-1
TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION..... 3/4 3-2
TABLE 3.3-2 (THIS TABLE IS NOT USED)..... 3/4 3-7
TABLE 4.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS..... 3/4 3-9
3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION..... 3/4 3-13
TABLE 3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION..... 3/4 3-15
TABLE 3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS..... 3/4 3-23
TABLE 3.3-5 (THIS TABLE IS NOT USED)..... 3/4 3-30
TABLE 4.3-2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS..... 3/4 3-34

Power Range, Neutron Flux, High Rates (Continued)

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the limit value.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. Both of these trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels in Mode 2 while the Source Range, Neutron Flux trip provides primary protection for the core in Modes 3, 4 and 5. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-2

(THIS TABLE IS NOT USED)

TABLE 3.3-2 (Continued)

(THIS TABLE IS NOT USED)

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2* The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $$Z + RE + SE \leq TA$$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   RE = The "as measured" value (in percent span) of rack error for the affected channel,

   SE = Either the "as measured" value (in percent span) of the sensor error, or the value for Column SE (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

---

*Control Room isolation not required prior to initial criticality on Cycle 1. Auxiliary Building Ventilation actuation not required prior to initial operation at ≥ 20% Rated Thermal Power (RTP) on Cycle 1.

TABLE 3.3-5

(THIS TABLE IS NOT USED)

TABLE 3.3-5 (Continued)

(THIS TABLE IS NOT USED)

TABLE 3.3-5 (Continued)

(THIS TABLE IS NOT USED)

TABLE 3.3-5 (Continued)

(THIS TABLE IS NOT USED)

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) essential service water pumps start and automatic valves position.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY OPERATING LICENSES

NO. NPF-37 AND NPF-66

BYRON STATION, UNITS 1 AND 2

DOCKET NOS. 50-454 AND 50-455

AND

SUPPORTING AMENDMENT NO. 12 TO FACILITY OPERATING LICENSES

NO. NPF-72 AND NPF-77

BRAIDWOOD STATION UNITS 1 AND 2

DOCKET NOS. 50-456 AND 50-457

1.0 INTRODUCTION

By letter dated January 5, 1988, Commonwealth Edison Company (CECo), the licensee, submitted a proposed amendment to Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2, and Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2. The proposed amendments remove two tables from the Technical Specifications which list reactor trip system instrumentation response times and engineered safety features response times.

2.0 DISCUSSION AND EVALUATION

The proposed amendments remove two tables from the Technical Specifications. The tables are being placed in a new section of the Byron/Braidwood FSAR, Chapter 16.3. The response times are not being changed by this amendment. The requirements to periodically measure the response times will remain in the Technical Specifications. The actions that are taken if a response time limit is not met are not being changed. Thus, the staff considers this change administrative in nature and concludes that it is acceptable.

The January 5, 1988 letter also proposed a change to the Bases on page B 2-5. The change explains that the Source Range, Neutron Flux trip provides primary protection for the core in Modes 3, 4 and 5. The trip is taken credit for in order to mitigate a rod withdrawal accident in Modes 3, 4 and 5 as described in the licensee's April 15, 1987 letter. The staff finds the change to the Bases acceptable.

8810130265 880927
PDR ADOCK 05000454
P PDC

3.0 ENVIRONMENTAL CONSIDERATION

This amendment relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(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 this amendment.

4.0 CONCLUSION

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

Principal Contributor: L. Olshan

Dated: September 27, 1988

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

September 27, 1988

MEMORANDUM FOR: Sholly Coordinator

FROM: Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NOS. 66994, 66995, 66996 AND 66997)

Commonwealth Edison Company Docket Nos. STN 50-454 and STN 50-455, Byron Station Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50-456 and STN 457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: January 5, 1988

Brief description of amendments: These amendments remove two tables from the Technical Specifications which list reactor trip system instrumentation response times and engineered safety features response times. These two tables are being placed in the Byron/Braidwood Final Safety Analysis Report.

Date of issuance: September 27, 1988

Effective Date: September 27, 1988

Amendment Nos.: 23 for Byron, 12 for Braidwood

Facility Operating License Nos. NPF-37, NPF-66, NPF-72, and NPF-77: Amendments revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: August 24, 1988 (53 FR 32290)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 27, 1988.

No significant hazards consideration comments received: No

8810130267 880927
PDR ADOCK 05000454
P PDC

Local Public Document Room location: For Bryon Station the Rockford Public Library, 215 N. Wyman Street, Rockford, Illinois 61101; for Braidwood Station the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

September 20, 1988

Local Public Document Room location: For Bryon Station the Rockford Public Library, 215 N. Wyman Street, Rockford, Illinois 61101; for Braidwood Station the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Leonard N. Olshan

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

DISTRIBUTION:
Docket file
LLuther
PDIII-2 r/f
LOlshan
SSands
OGC-Rockville

PDIII-2	PDIII-2	PDIII-2	OGC	PDIII-2
SSands:km	LOlshan	LLuther	RBachmann	DMuller FOR
9/8/88	9/7/88	9/19/88	9/20/88	9/19/88