July 27, 1988

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

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

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. 21 to Facility Operating License No. NPF-37 and Amendment No.21 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 10 to Facility Operating License No. NPF-72, and Amendment No.10 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 approve changes to the Technical Specifications that (1) change the shutdown and reporting requirements resulting from high specific activity of the reactor coolant; (2) change the curve on page 3/4 4-40 to more conservative Cold Overpressure Protection System setpoints; (3) allow other indications, in addition to the absence of alarms, to be used to verify accumulator borated water level and nitrogen cover pressure; (4) correct typographical errors on Byron pages 3/4 6-23 and 3/4 7-14; (5) make Table 5.7-1 consistent with the FSAR and Section XI of the ASME code; and (6) correct and update the titles of various management personnel.

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, 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.21 to NPF-37
2. Amendment No.21 to NPF-66
3. Amendment No.10 to NPF-72
4. Amendment No.10 to NPF-77
5. Safety Evaluation

cc: w/enclosures
See next page

DFol
1/1

* SEE PREVIOUS CONCURRENCE

PDIII-2:PM	PDIII-2:PM	PDIII-2:LA	OGC-Rockville	PDIII-2:PD
LOlshan*:bj	SSands	LLuther		DMuller
06/30/88	7/14/88	7/14/88	7/21/88	7/19/88

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

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

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

These amendments approve changes to the Technical Specifications that (1) change the shutdown and reporting requirements resulting from high specific activity of the reactor coolant; (2) change the curve on page 3/4 4-40 to more conservative Cold Overpressure Protection System setpoints; (3) allow other indications, in addition to the absence of alarms, to be used to verify accumulator borated water level and nitrogen cover pressure; (4) correct typographical errors on Byron pages 3/4 6-23 and 3/4 7-14; (5) make Table 5.7-1 consistent with the FSAR and Section XI of the ASME code; and (6) correct and update the titles of various management personnel.

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, 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. to NPF-37
2. Amendment No. to NPF-66
3. Amendment No. to NPF-72
4. Amendment No. 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:bj	SSands	LLuther		DMuller
6/30/88	/ /88	/ /88	/ /88	/ /88

July 27, 1988

Docket Nos. STN 50-454, STN 50-455
and STN 50-456, STN 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	~~BGrimes~~

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. 21 to Facility Operating License No. NPF-37 and Amendment No.21 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 10 to Facility Operating License No. NPF-72, and Amendment No.10 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 approve changes to the Technical Specifications that (1) change the shutdown and reporting requirements resulting from high specific activity of the reactor coolant; (2) change the curve on page 3/4 4-40 to more conservative Cold Overpressure Protection System setpoints; (3) allow other indications, in addition to the absence of alarms, to be used to verify accumulator borated water level and nitrogen cover pressure; (4) correct typographical errors on Byron pages 3/4 6-23 and 3/4 7-14; (5) make Table 5.7-1 consistent with the FSAR and Section XI of the ASME code; and (6) correct and update the titles of various management personnel.

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, 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. 21 to NPF-37
2. Amendment No. 21 to NPF-66
3. Amendment No. 10 to NPF-72
4. Amendment No. 10 to NPF-77
5. Safety Evaluation

cc: w/enclosures
See next page

* SEE PREVIOUS CONCURRENCE

PDIII-2:PM	PDIII-2:PM	PDIII-2:LA	OGC-Rockville	PDIII-2:PD
LOlshan*:bj	SSands	LLuther		DMuller
06/30/88	7/ /88	7/ /88	7/ /88	7/ /88

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

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

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

These amendments approve changes to the Technical Specifications that (1) change the shutdown and reporting requirements resulting from high specific activity of the reactor coolant; (2) change the curve on page 3/4 4-40 to more conservative Cold Overpressure Protection System setpoints; (3) allow other indications, in addition to the absence of alarms, to be used to verify accumulator borated water level and nitrogen cover pressure; (4) correct typographical errors on Byron pages 3/4 6-23 and 3/4 7-14; (5) make Table 5.7-1 consistent with the FSAR and Section XI of the ASME code; and (6) correct and update the titles of various management personnel.

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, 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. to NPF-37
2. Amendment No. to NPF-66
3. Amendment No. to NPF-72
4. Amendment No. 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:bj	SSands	LLuther		DMuller
6/30/88	/ /88	/ /88	/ /88	/ /88

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

July 27, 1988

Docket Nos. STN 50-454, STN 50-455
and STN 50-456, STN 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. 21 to Facility Operating License No. NPF-37 and Amendment No.21 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 10 to Facility Operating License No. NPF-72, and Amendment No.10 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 approve changes to the Technical Specifications that (1) change the shutdown and reporting requirements resulting from high specific activity of the reactor coolant; (2) change the curve on page 3/4 4-40 to more conservative Cold Overpressure Protection System setpoints; (3) allow other indications, in addition to the absence of alarms, to be used to verify accumulator borated water level and nitrogen cover pressure; (4) correct typographical errors on Byron pages 3/4 6-23 and 3/4 7-14; (5) make Table 5.7-1 consistent with the FSAR and Section XI of the ASME code; and (6) correct and update the titles of various management personnel.

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.21 to NPF-37
2. Amendment No.21 to NPF-66
3. Amendment No.10 to NPF-72
4. Amendment No.10 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

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. 21
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 18, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 21 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: July 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. 21
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 18, 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. 21 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

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: July 27, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 21 AND 21

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
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-40	3/4 4-40
3/4 5-1	3/4 5-1
3/4 6-23	3/4 6-23
3/4 7-14	3/4 7-14
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6
5-6	5-6
6-7	6-7
6-8	6-8
6-13	6-13
6-18	6-18
	6-18a

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
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 18, 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. 10 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

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: July 27, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 10 AND 10

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
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-40	3/4 4-40
3/4 5-1	3/4 5-1
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6
5-6	5-6
6-8	6-8
6-13	6-13
6-18	6-18
	6-18a

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

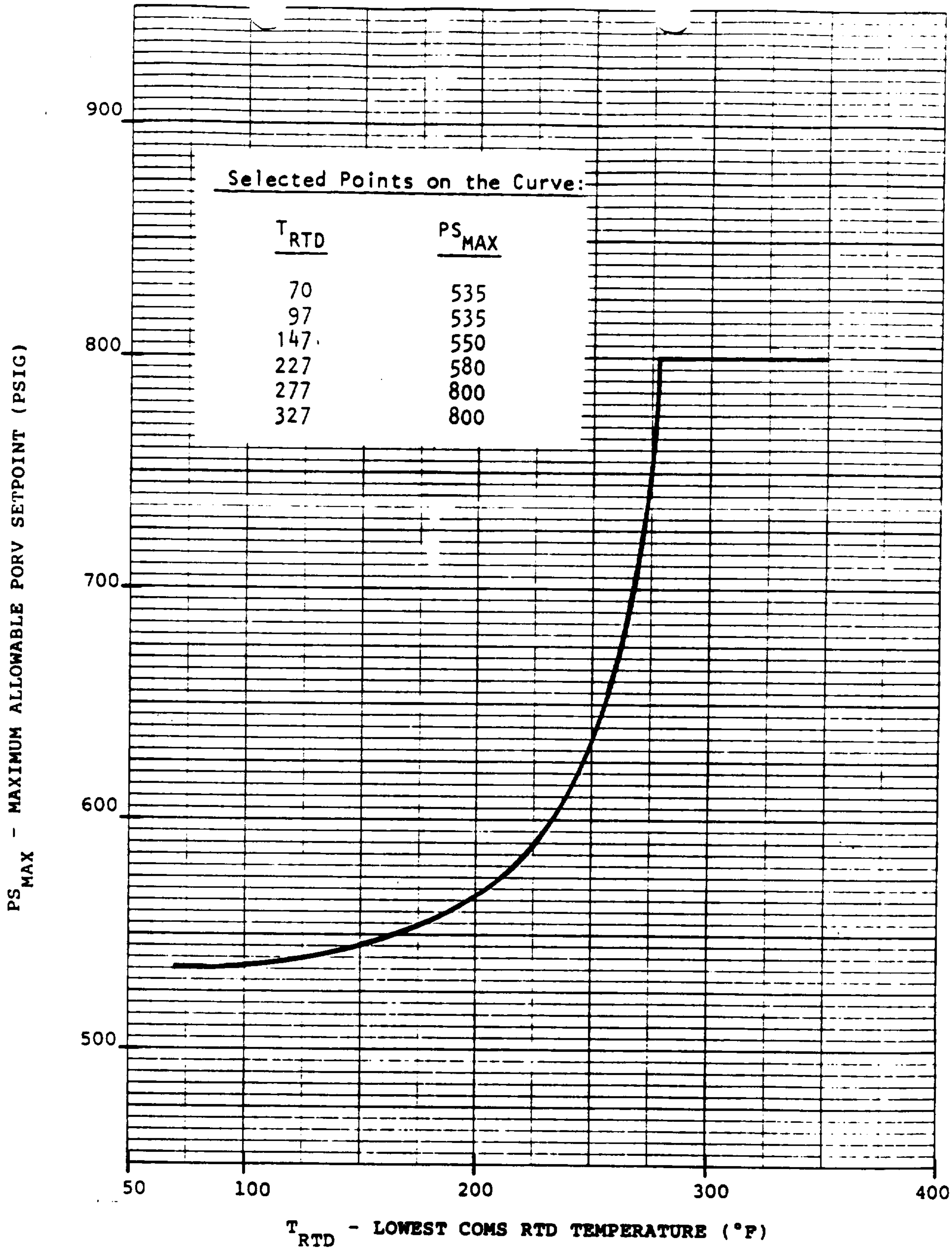

FIGURE 3.4-4

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM APPLICABLE UP TO 10 EFPY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water level of between 31% and 63%,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Braidwood Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 100°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 leak tests.	Pressurized to $\geq$ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 3107 psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 1481 psig.

OFFSITE (Continued)

5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;

6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;

7) All REPORTABLE EVENTS;

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Assistant Vice President and General Manager of Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The audit function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance (Operations) and the Director of Quality Assurance (Maintenance).

Either of the above, or designated Corporate Staff or Supervision approved by the Manager of Quality Assurance shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;

3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;

4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

ONSITE (Continued)

3) Review of all proposed changes to the Technical Specifications;

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Assistant Vice President and General Manager of Nuclear Stations and to the Superintendent of the Offsite Review and Investigative Function;

6) Review of all REPORTABLE EVENTS;

7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;

8) Review of the Station Security Plan and implementing procedures and submittal of recommended Security Plan changes to the Director of Corporate Security;

9) Review of the Emergency Plan and station implementing procedures and submittal of recommended changes to the Station Security Plan to the Director of Corporate Security;

10) Review of Unit operations to detect potential hazards to nuclear safety;

11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Assistant Vice President and General Manager of Nuclear Stations and the Superintendent of the Offsite Review and Investigative Function; and

12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

13) Review of the Fire Protection Program and implementing instructions and submittal of recommended changes to the Offsite Review and Investigative Function.

c. Authority

The Technical Staff Supervisor is responsible to the Station Manager and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager shall follow such recommendations or select a course of action that

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the Final Safety Analysis Report FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. Tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

REPORTING REQUIREMENTS (Continued)

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 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. 10 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 18, 1988, Commonwealth Edison (CECo), the licensee, submitted a proposed amendment to Facility Operating Licenses No. NPF-37, NPF-66, NPF-72, and NPF-77, for the Byron and Braidwood Stations, Units 1 and 2. The proposed amendment requests seven miscellaneous Technical Specification (TS) changes which are further discussed in Section 2.0.

A Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on April 6, 1988 (53 FR 11367). No requests for hearing and no public comments were received.

2.0 DISCUSSION AND EVALUATION

The following are descriptions and evaluations of each of the seven TS changes for Byron and Braidwood Units 1 and 2. None of the changes involve physical modifications to the facilities. It should be noted that some of the changes are specific to Byron Station.

Description of Change; Technical Specification Pages 3/4 4-27, 3/4 4-28, B3/4 4-5, B3/4 4-6, and 6-18

The proposed change deletes the requirement for a Special Report to the Commission if reactor coolant specific activity exceeds 1 microCurie per gram dose equivalent I-131 for greater than 500 hours in any consecutive six month period. It also deletes the requirement to shutdown a plant if reactor coolant iodine activity limits are exceeded for 800 hours in a 12-month period.

Evaluation

This change is based on Generic Letter 85-19, dated September 27, 1985. The letter stated that reporting requirements for iodine spiking can be reduced from a short term report to an item included in the Annual Report. The letter further states that the requirements to shut down a plant after 800 hours with iodine activity above the limit can be eliminated due to the fact that nuclear fuel has been greatly improved in recent years, with the result that normal coolant iodine is well below the limit. Appropriate actions would be initiated long before accumulating 800 hours above the iodine activity limit. The proposed TS changes are consistent with the model TS recommended in the Generic Letter, and are therefore considered acceptable.

Description of Change; Technical Specification Page 3/4 4-40

The proposed change revises Figure 3.4-4, "Nominal PORV Pressure Relief Setpoint Versus RCS Temperature for the Cold Overpressure Protection System Applicable up to 10 EFPY." The changes reflect a larger uncertainty in the wide range temperature instrumentation and prevent the need for additional stress analyses following an overpressure event.

Evaluation

The requested change to Cold Overpressure Mitigation System (COMS) setpoints is based on a letter from Westinghouse dated November 18, 1985 for two reasons: (1) A larger uncertainty in the wide range temperature instrumentation is assumed; and (2) The updated COMS setpoints eliminate the need for a detailed stress evaluation of the PORV inlet and discharge piping and steam generator tube sheet following a single overpressure event. The revised setpoints meet 10 CFR Part 50, Appendix G criteria and are consistent with the Byron/Braidwood FSAR. The margin of safety has not been reduced because the change is in the conservative direction and is therefore bounded by previous analyses. The proposed change is considered acceptable.

Description of Change; Technical Specification Page 3/4 5-1

The proposed change revises TS Surveillance 4.5.1.1, which deals with accumulator operability. The proposed change deletes the words "by the absence of alarms" from the phrase: "Verifying, by the absence of alarms, the contained borated water level and nitrogen cover-pressure in the tanks."

Evaluation

The licensee requested the change because the current wording could be interpreted that the unit must be shut down if an annunciator failed. Deleting the words "by the absence of alarms" permits the operators to verify the required accumulator borated water level and nitrogen cover pressure by using other instruments. Duplicate level channels and pressure channels provide signals to two sets of safety-related instruments in the control room which can be used to read accumulator water level and nitrogen cover pressure. The proposed TS change still requires the verification of accumulator parameters, but will allow the operators flexibility in how the parameters are verified. The proposed change is consistent with Section 6.3 of the FSAR which describes the accumulators and the associated instrumentation. The proposed TS change has no effect on safety and is considered acceptable.

Description of Change; Technical Specification Page 3/4 6-23

The proposed change corrects a typographical error for one Safety Injection Valve number on Table 3.6-1, from "SI 8805D" to SI 8905D," for Byron Station TS only.

Evaluation

The corrected valve number, "SI 8905D," is consistent with those listed in Braidwood TS Table 3.6-1 and Byron/Braidwood FSAR Table 6.2-58. The change is administrative in nature, has no effect on safety, and is considered acceptable.

Description of Change; Technical Specification Page 3/4 7-14

The proposed change corrects a typographical error in the value of the ultimate heat sink (UHS) cooling tower basin water level from 873.5 feet to 873.75 feet, for Byron Station TS only.

Evaluation

The corrected water level of 873.75 feet above mean sea level is consistent with other portions of Byron TS 3/4 7.5 which reference a minimum UHS cooling tower basin water level. The change is administrative in nature, has no effect on safety, and is considered acceptable.

Description of Change; Technical Specification Page 5-6

The proposed change revises TS Table 5.7-1, "Component Cyclic or Transient Limits," so that it is consistent with the design limits contained in Section 3.9 of Byron/Braidwood FSAR.

Evaluation

The proposed change raises the limit for reactor coolant system (RCS) leak tests from 50 to 200, the limit for RCS hydrostatic pressure tests from 5 to 10, and the limit for secondary coolant system hydrostatic tests from 5 to 10. It also raises the limits for primary and secondary pressures during hydrostatic testing to 1.25 times the design pressures as required by the ASME Code. The changes are consistent with Section 3.9 of the FSAR and Section XI of the ASME Code. Although these changes raise the number of transients the plants are permitted to withstand, the changes are consistent with the FSAR. Therefore, the proposed changes are acceptable.

Description of Change; Technical Specification Pages 6-7, 6-8, and 6-13

The proposed changes are being made to update some Commonwealth Edison management titles and clarify the functional authority of Quality Assurance personnel. The change requested for Page 6-7 has previously been corrected in the Braidwood TS.

The proposed changes are administrative in nature. Because there are no significant changes in duties, the changes have no adverse effect on safety, and are considered acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves primarily changes in the installation or use of facility components located within the restricted area defined in 10 CFR Part 20, and changes in reporting and surveillance requirements. The changes are primarily administrative in nature and do not involve any physical modifications to the facility. The amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets 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 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.

5.0 ACKNOWLEDGEMENT

This evaluation was prepared by B. A. Azab.

Dated: July 27, 1988

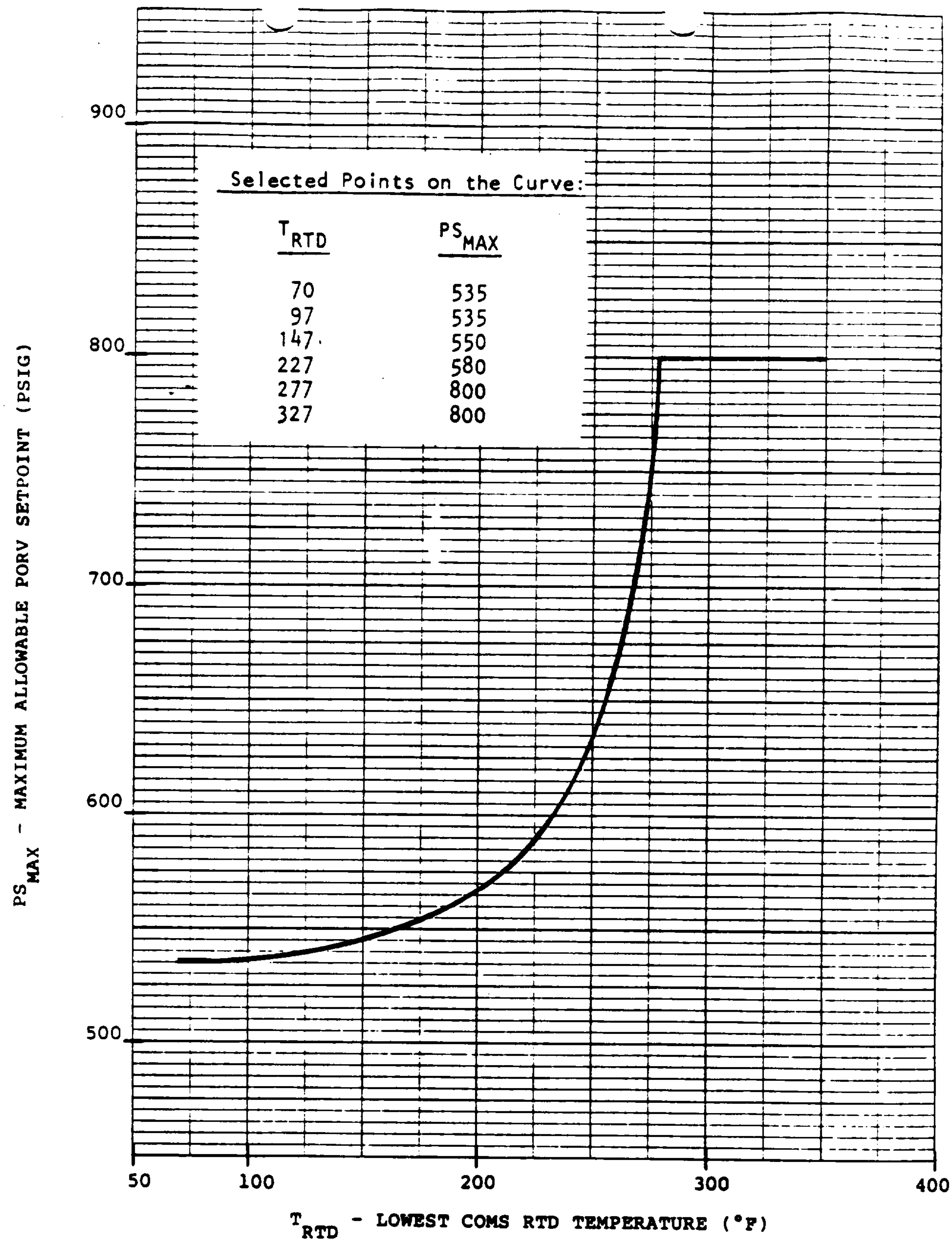

FIGURE 3.4-4

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 10 EFPY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water level of between 31% and 63%,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION	VALVE NO.	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
9. Manual (Continued)			
99	FW015D*,#	Feedwater	N.A.
100	FW015A*,#	Feedwater	N.A.
101	FW015B*,#	Feedwater	N.A.
102	FW015C*,#	Feedwater	N.A.
10. Check			
28	CV8113	RCP Seal Water Return	N.A.
37	CV8348*	RCS Loop Fill	N.A.
6	WØ007A	Chilled Water	N.A.
10	WØ007B	Chilled Water	N.A.
21	CC9534	RCP Mtr Brng Return	N.A.
24	CC9518	RCP Thermal Barrier Return	N.A.
25	CC9486	RCP Cooling Wtr Supply	N.A.
1	CS008A	Containment Spray	N.A.
16	CS008B	Containment Spray	N.A.
39	IA091	Instrument Air	N.A.
30	WM191	Make-Up Demin	N.A.
52	PR032	Process Radiation	N.A.
AL	PR002G	Process Radiation	N.A.
AL	PR002H	Process Radiation	N.A.
12	PS231A	Hydrogen Monitor	N.A.
31	PS231B	Hydrogen Monitor	N.A.
27	RY8047	PRT Nitrogen	N.A.
44	RY8046	PRT Make-Up	N.A.
26	SI8815*	Safety Injection	N.A.
50	SI8818A*	Safety Injection	N.A.
50	SI8818D*	Safety Injection	N.A.
51	SI8818B*	Safety Injection	N.A.
51	SI8818C*	Safety Injection	N.A.
59	SI8905A*	Safety Injection	N.A.
59	SI8905D*	Safety Injection	N.A.
60	SI8819A*	Safety Injection	N.A.
60	SI8819B*	Safety Injection	N.A.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With one essential service water makeup pump inoperable, restore the essential service water makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With the essential service water pump discharge water temperature not meeting the above requirement, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the minimum Rock River water level not meeting the above requirement, notify the NRC within 1 hour in accordance with the procedure of 10 CFR 50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days, verify the Rock River flow within 1 hour, and:

   (1) If Rock River flow is less than 700 cubic feet per second (cfs) be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours, or

   (2) If Rock River flow is equal to or greater than 700 cfs continue verification procedure every 12 hours or until Rock River water level exceeds 670.6 feet MSL or

   (3) If Rock River level is equal to or less than 664.7 feet MSL be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours

f. With one deep well inoperable and:

   (1) The Rock River water level predicted, through National Weather Service flood forecasts, to exceed 702 feet MSL, or

   (2) The Rock River water level at or below 670.6 feet MSL, or

   (3) A tornado watch issued by the NWS that includes the area for the Byron Station.

   Notify the NRC within 1 hour in accordance with the procedure of 10 CFR 50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days and restore both wells to OPERABLE status before the Rock River water level exceeds 702 feet MSL or the minimum Rock River level or flow falls below 664.7 feet MSL or 700 cfs, respectively, or within 72 hours, whichever occurs first, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE at least once per:

a. 24 hours by verifying the water level in each UHS cooling tower basin to be greater than or equal to 873.75 feet MSL. (50%), OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Byron Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $<$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 100°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential $>$ 320°F.
	200 leak tests.	Pressurized to $\geq$ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 3107 psig.
Secondary Coolant System	1 large steam line break.	Break in a $>$ 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 1481 psig.

6.5 REVIEW INVESTIGATION AND AUDIT (Continued)

OFFSITE

6.5.1 The Superintendent of the Offsite Review and Investigative Function shall be appointed by the Manager of Nuclear Safety responsible for nuclear activities. The audit function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. Offsite Review and Investigative Function

The Superintendent of the Offsite Review and Investigative Function shall: (1) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (2) select each participant for this function, (3) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (4) independently review and approve the findings and recommendations developed by personnel performing the review and investigative function, (5) approve and report in a timely manner all findings of non-compliance with NRC requirements to the Station Manager, Assistant Vice President and General Manager - Nuclear Stations, Manager of Quality Assurance, and the Vice President - Nuclear Operations. During periods when the Superintendent of Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate, who satisfies the formal training and experience for the Superintendent of the Offsite Review and Investigative Function. The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review:

1) The safety evaluations for: (1) changes to procedures, equipment, or systems as described in the safety analysis report, and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance;

2) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

3) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

4) Proposed changes in Technical Specifications or this Operating License;

OFFSITE (Continued)

5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;

6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;

7) All REPORTABLE EVENTS;

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Assistant Vice President and General Manager - Nuclear Stations, and Manager of Quality Assurance.

b. Audit Function

The audit function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance (Operations) and the Director of Quality Assurance (Maintenance).

Either of the above, or designated corporate staff or supervision approved by the Manager of Quality Assurance shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;

3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;

4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

ONSITE (Continued)

3) Review of all proposed changes to the Technical Specifications;

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Assistant Vice President and General Manager - Nuclear Stations and to the Superintendent of the Offsite Review and Investigative Function;

6) Review of all REPORTABLE EVENTS;

7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;

8) Review of the Station Security Plan and implementing procedures and submittal of recommended changes to the Station Security Plan to the Director of Corporate Security;

9) Review of the Emergency Plan and station implementing procedures and submittal of recommended changes to the Assistant Vice President and General Manager - Nuclear Stations;

10) Review of Unit operations to detect potential hazards to nuclear safety;

11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Assistant Vice President and General Manager - Nuclear Stations and the Superintendent of the Offsite Review and Investigative Function; and

12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

13) Review of the Fire Protection Program and implementing instructions and submittal of recommended changes to the Offsite Review and Investigative Function.

c. Authority

The Technical Staff Supervisor is responsible to the Station Manager and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager shall follow such recommendations or select a course

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the Final Safety Analysis Report FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. Tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

REPORTING REQUIREMENTS (Continued)

accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

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. 10
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 18, 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.10 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: July 27, 1988