November 25, 1987

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

Mr. L. D. Butterfield, Jr.
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

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Dear Mr. Butterfield:

The Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. NPF-37, Amendment No. 12 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively and Amendment No. 2 to Facility Operating License No. NPF-72 for Braidwood Station, Unit No. 1. This amendment also applies to the Technical Specifications for Braidwood Station, Unit No. 2, although Unit 2 does not have an operating license. Braidwood Station, Units 1 and 2, have common Technical Specifications. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated September 3, 1987.

These amendments revise the Technical Specifications to allow a one-time extension to 32 months for the interval for performing certain 18-month instrument surveillance. The amendments apply only to the first cycle of operation, and are therefore moot for Byron Station, Unit 1, which is in its second cycle operation.

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

Sincerely,

Original Signed by/

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

Original Signed by/

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

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

cc: w/enclosures
See next page

*See Previous Concurrence

PDIII-2:LA	PDIII-2:PM	PDIII-2:PM	OGC-Bethesda	PDIII-2:PD
*LLuther	*SSands	*LOlshan:bj	*RBachmann	for DMuller
11/10/87	11/23/87	11/10/87	11/13/87	11/25/87

8712040071 871125
PDR ADOCK 05000454
P PDR

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

Mr. L.D. Butterfield, Jr.
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

DISTRIBUTION
Docket Files
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PD3-2 Rdg. File
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TBarnhart (16)
WJones
EButcher
JStewart
ACRS (10)
GPA/PA
PD3-2 Plant File
ARM/LFMB

Dear Mr. Butterfield:

The Commission has issued the enclosed Amendment No. ~~13~~ 12 to Facility Operating License No. NPF-37, Amendment No. ~~13~~ 12 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively and Amendment No. 2 to Facility Operating License No. NPF-72 for Braidwood Station, Unit No. 1. This amendment also applies to the Technical Specifications for Braidwood Station, Unit No. 2, although Unit 2 does not have an operating license. Braidwood Station, Units 1 and 2, have common Technical Specifications. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated September 3, 1987.

These amendments revise the Technical Specifications to allow a one-time extension to 32 months for the interval for performing certain 18-month instrument surveillance. The amendments apply only to the first cycle of operation, and are therefore moot for Byron Station, Unit 1, which is in its second cycle operation.

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

Sincerely,

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

Enclosures:
1. Amendment No. ~~13~~ 12 to NPF-37
2. Amendment No. ~~13~~ 12 to NPF-66
3. Amendment No. 2 to NPF-72
4. Safety Evaluation

cc: w/enclosures
See next page

PDIII-2:LA	PDIII-2:PM	PDIII-2:PM	OGC-Beth.	PDIII-2:PD
LLuther	SSands	LOlshan:bj	R Bachmann	DMuller
11/10/87	11/23/87	11/10/87	11/10/87	/ /87

J. Scinto concurred 11/10/87 (See FR notice)

Mr. L. D. Butterfield, Jr.
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.
Isham, Lincoln & Beale
1150 Connecticut Ave., N.W.
Suite 1100
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
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 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
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
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. 12
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 September 3, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

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P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 12 and the Environment 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

Lawrence E. Olshan
for

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: November 25, 1987

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. 12
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 September 3, 1987, 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 (NUREG-1113), as revised through Amendment No. 12 and revised by Attachment 2 to NPF-60, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Jean R. Olsha
for

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: November 25, 1987

ATTACHMENT TO LICENSE AMENDMENT NOS. 12 AND 12

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 2-9	3/4 2-9
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-42	3/4 3-42
3/4 3-52	3/4 3-52
3/4 3-55	3/4 3-55
3/4 3-60	3/4 3-60
3/4 3-61	3/4 3-61
3/4 3-67	3/4 3-67
3/4 3-68	3/4 3-68
3/4 3-69	3/4 3-69
3/4 3-70	3/4 3-70
3/4 4-12	3/4 4-12
3/4 4-20	3/4 4-20
	3/4 4-20 a
3/4 4-41	3/4 4-41
3/4 5-2	3/4 5-2

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation defined by Specification 3.2.3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the region of acceptable operation of Specification 3.2.3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Specification 3.2.3 at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.*

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. The measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. Prior to the precision heat balance measurement, at least two of the four feedwater flow meter venturis shall be visually inspected and, if fouling is found, all venturis shall be cleaned.

*The specified 18 month interval may be extended to 32 months for Cycle 1 only.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5a)#	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)#	Q	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)#	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)#	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5a)#	Q	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5b, 12)#	Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)#	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R#	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R#	Q**	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R#	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (Above P-7)	S	R#	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Reactor Coolant Flow-Low	S	R#	Q	N.A.	N.A.	1
13. Steam Generator Water Level-Low-Low	S	R#	Q**	N.A.	N.A.	1, 2
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q**	N.A.	1
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q	N.A.	1
16. Turbine Trip (Above P-7)						
a. Emergency Trip Header Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Throttle Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)#	Q	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)#	Q (8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)#	Q (8)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)#	Q (8)	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R#	Q (8)	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.

**These channels also provide inputs to ESFAS. The Operational Test Frequency for these channels in Table 4.3-2 is more conservative and, therefore, controlling.

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

##Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5a) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5b) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low (Above P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low (Above P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-3	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Vent Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3.c. Containment Vent Isolation (Continued)								
2) Manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Surveillance Requirements.							
3) Manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Surveillance Requirements.							
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure - Negative Rate - High (Below P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Turbine Trip and Feedwater (Continued)								
b. Steam Generator Water Level-High-High (P-14)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Undervoltage-RCP Bus	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Division 11 for Unit 1 (Division 21 for Unit 2) ESF Bus Undervoltage	N.A.	R	N.A.	M(2)	N.A.	N.A.	N.A	1, 2, 3, 4
g. Auxiliary Feedwater Pump Suction Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Opening of Containment Sump Suction Isolation Valves								

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Automatic Opening (Continued)								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With Safety Injection	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage	N.A.	R	N.A.	M(2)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Feature Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14 (High-High)	S	R#	M	N.A.	M(1)	M(1)	Q	1, 2

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Undervoltage relay operability is to be verified independently. An inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.
(3) Verify undervoltage relay operability and alarm annunciation only. Do not maintain an undervoltage condition beyond the setpoint for the time delay relay.
The specified 18 month interval may be extended to 32 months for Cycle 1 only.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Fuel Building Isolation-Radioactivity-High and Criticality (ORE-AR055/56)	S	R#	M	*
2. Containment Isolation-Containment Radioactivity-High				
a) Unit 1 (1RE-AR011/12)	S	R#	M	All
b) Unit 2 (2RE-AR011/12)	S	R#	M	All
3. Gaseous Radioactivity-RCS Leakage Detection				
a) Unit 1 (1RE-PR011B)	S	R#	M	1, 2, 3, 4
b) Unit 2 (2RE-PR011B)	S	R#	M	1, 2, 3, 4
4. Particulate Radioactivity-RCS Leakage Detection				
a) Unit 1 (1RE-PR011A)	S	R#	M	1, 2, 3, 4
b) Unit 2 (2RE-PR011A)	S	R#	M	1, 2, 3, 4
5. Main Control Room Isolation-Outside Air Intake-Gaseous Radioactivity-High (ORE-PR031B/32B and ORE-PR033B/34B)	S	R#	M	All

*With new fuel or irradiated fuel in the fuel storage areas or fuel building.
#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Intermediate Range Neutron Flux	M	R#
2. Source Range Neutron Flux	M*	R#
3. Reactor Coolant Temperature - Wide Range	M	R#
4. Pressurizer Pressure	M	R#
5. Pressurizer Level	M	R#
6. Steam Generator Pressure	M	R#
7. Steam Generator Level	M	R#
8. RHR Temperature	M	R
9. Auxiliary Feedwater Flow Rate	M	R

*When below P-6.
#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R#
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R#
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R#
4. Reactor Coolant Pressure - Wide Range	M	R#
5. Pressurizer Water Level	M	R#
6. Steam Line Pressure	M	R#
7. Steam Generator Water Level - Narrow Range	M	R#
8. Steam Generator Water Level - Wide Range	M	R#
9. Refueling Water Storage Tank Water Level	M	R#
10. Auxiliary Feedwater Flow Rate	M	R
11. PORV Position Indicator* (Open/Closed)	M	N.A.
12. PORV Block Valve Position Indicator** (Open/Closed)	M	N.A.
13. Safety Valve Position Indicator (Open/Closed)	M	N.A.
14. Containment Floor Drain Sump Water Level (Narrow Range)	M	R#
15. Containment Water Level (Wide Range)	M	R#
16. In Core Thermocouples	M	R#
17. Containment High Range Area Radiation	M	R***,#
18. Auxiliary Building Vent Stack - Wide Range Noble Gas	M	R#
19. Main Steam Line Radiation	M	R#
20. Reactor Vessel Water Level	M	R#

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

***CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release					
Liquid Radwaste Effluent Line (ORE-PR001)	D	P	R(3)#	Q(1)	N.A.
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release					
a. Essential Service Water					
1) Unit 1					
a) RCFC 1A and 1C Outlet (1RE-PR002)	D	M	R(3)#	Q(2)	N.A.
b) RCFC 1B and 1D Outlet (1RE-PR003)	D	M	R(3)#	Q(2)	N.A.
2) Unit 2					
a) RCFC 2A and 2C Outlet (2RE-PR002)	D	M	R(3)#	Q(2)	N.A.
b) RCFC 2B and 2D Outlet (2RE-PR003)	D	M	R(3)#	Q(2)	N.A.
b. Station Blowdown Line (ORE-PR010)	D	M	R(3)#	Q(2)	N.A.
3. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line (Loop-WX001)	D(4)	N.A.	R#	N.A.	Q
b. Station Blowdown Line (Loop-CW032)	D(4)	N.A.	R	N.A.	Q

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Plant Vent Monitoring System - Unit 1					
a. Noble Gas Activity Monitor - Providing Alarm					
1) High Range (1RE-PR028D)	D	M	R(3)##	Q(2)	*
2) Low Range (1RE-PR028B)	D	M	R(3)##	Q(2)	*
b. Iodine Sampler (1RE-PR028C)	D	M	R(3)##	Q(2)	*
c. Particulate Sampler (1RE-PR028A)	D	M	R(3)##	Q(2)	*
d. Effluent System Flow Rate Measuring Device (LOOP-VA019)	D	N.A.	R##	Q	*
e. Sampler Flow Rate Measuring Device (1FT-PR165)	D	N.A.	R##	Q	*
2. Plant Vent Monitoring System - Unit Two					
a. Noble Gas Activity Monitor- Providing Alarm					
1) High Range (2RE-PR028D)	D	M	R(3)##	Q(2)	*
2) Low Range (2RE-PR028B)	D	M	R(3)##	Q(2)	*
b. Iodine Sampler (2RE-PR028C)	D	M	R(3)##	Q(2)	*
c. Particulate Sampler (2RE-PR028A)	D	M	R(3)##	Q(2)	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Plant Vent Monitoring System - Unit Two (continued)					
d. Effluent System Flow Rate Measuring Device (LOOP-VA020)	D	N.A.	R##	Q	*
e. Sampler Flow Rate Measuring Device (2FT-PR165)	D	N.A.	R##	Q	*
3. Gaseous Waste Management System					
a. Hydrogen Analyzer (OAT-GW8000)	D	N.A.	Q(4)	M	**
b. Oxygen Analyzer (OAIT-GW004 and OAT-GW8003)	D	N.A.	Q(5)	M	**
4. Gas Decay Tank System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ORE-PR002A and 2B)	P	P	R(3)##	Q(1)	*
5. Containment Purge System					
a. Noble Gas Activity Monitor - Providing Alarm (RE-PR001B)	D	P	R(3)##	Q(2)	*
b. Iodine Sampler (RE-PR001C)	P	P	R(3)##	N.A.	*
c. Particulate Sampler (RE-PR001A)	P	P	R(3)##	N.A.	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Radioactivity Monitors Providing Alarm and Automatic Closure of Surge Tank Vent Component Cooling Water Line (ORE-PR009 and RE-PR009)	D	M	R(3)##	Q(1)	*

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

*At all times.
**During WASTE GAS HOLDUP SYSTEM operation.
#All instruments required for Unit 1 or Unit 2 operation.
##The specified 18 month interval may be extended to 32 months for Cycle 1 only.

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing hydrogen and nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing oxygen and nitrogen.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: 1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and 2) apply the ACTION of b. or c. above, as appropriate for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION#, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. of Specification 3.4.4.

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE
LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,
b. The Containment Floor Drain and Reactor Cavity Flow Monitoring System, and
c. The Containment Gaseous Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

   1) Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

   2) Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

   3) Perform a Reactor Coolant System water inventory balance at least once per 8 hours.

   Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE
LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. Containment Floor Drain and Reactor Cavity Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months,# and

c. Verify the oil separator portion of the containment floor drain collection sump has been filled to the level of the overflow to the containment floor drain unidentified leakage collection weir box once per 18 months, following refueling, and prior to initial startup.

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months#; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve RH8702A is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RH8702B is open.

b. For RHR suction relief valve 8708A:

1) By verifying at least once per 31 days that RH8701B is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RH8701A is open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution, and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the MCC compartment is open and tagged out of service.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months# by the performance of a CHANNEL CALIBRATION.

#The specified 18 month interval may be extended to 32 months for Cycle 1 only.

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. 2
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 September 3, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 2 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

Jean R. Dilhan
for

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: November 25, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 2

FACILITY OPERATING LICENSE NO NPF-72

DOCKET NO. STN-50-456

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 2-9	3/4 2-9
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-42	3/4 3-42
3/4 3-52	3/4 3-52
3/4 3-55	3/4 3-55
3/4 3-62	3/4 3-62
3/4 3-63	3/4 3-63
3/4 3-69	3/4 3-69
3/4 3-70	3/4 3-70
3/4 3-71	3/4 3-71
3/4 3-72	3/4 3-72
3/4 4-12	3/4 4-12
3/4 4-20	3/4 4-20
3/4 4-41	3/4 4-41
3/4 5-2	3/4 5-2

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation defined by Specification 3.2.3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the region of acceptable operation of Specification 3.2.3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Specification 3.2.3 at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.*

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. The measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. Prior to the precision heat balance measurement, at least two of the four feedwater flow meter venturis shall be visually inspected and, if fouling is found, all venturis shall be cleaned.

*The specified 18 month interval may be extended to 32 months for cycle 1 only.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5a)#	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)#	Q	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)#	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)#	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5a)#	Q	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5b, 12)#	Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)#	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R#	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low (Above P-7)	S	R#	Q**	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R#	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High (Above P-7)	S	R#	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Reactor Coolant Flow-Low	S	R#	Q	N.A.	N.A.	1
13. Steam Generator Water Level-Low-Low	S	R#	Q**	N.A.	N.A.	1, 2
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q**	N.A.	1
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q	N.A.	1
16. Turbine Trip (Above P-7)						
a. Emergency Trip Header Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Throttle Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)#	Q	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)#	Q (8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)#	Q (8)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)#	Q (8)	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R#	Q (8)	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.

**These channels also provide inputs to ESFAS. The Operational Test Frequency for these channels in Table 4.3-2 is more conservative and, therefore, controlling.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

##Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5a) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5b) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Cooling Fans, Control Room Isolation, Phase "A" Isolation, Turbine Trip, Auxiliary Feedwater, Containment Vent Isolation and Essential Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low (Above P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low (Above P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-3	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Vent Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3.c. Containment Vent Isolation (Continued)								
2) Manual Phase "A" Isolation	See Item 3.a.1 above for all manual Phase "A" Isolation Surveillance Requirements.							
3) Manual Phase "B" Isolation	See Item 3.b.1 above for all manual Phase "B" Isolation Surveillance Requirements.							
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure - Negative Rate - High (Below P-11)	S	R#	M	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Turbine Trip and Feedwater (Continued)								
b. Steam Generator Water Level-High-High (P-14)	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Undervoltage-RCP Bus	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Division 11 for Unit 1 (Division 21 for Unit 2) ESF Bus Undervoltage	N.A.	R	N.A.	M(2)	N.A.	N.A.	N.A	1, 2, 3, 4
g. Auxiliary Feedwater Pump Suction Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Opening of Containment Sump Suction Isolation Valves								

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Automatic Opening (Continued)								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With Safety Injection	S	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
8. Loss of Power								
a. ESF Bus Undervoltage	N.A.	R	N.A.	M(2)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Grid Degraded Voltage	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Feature Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R#	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14 (High-High)	S	R#	M	N.A.	M(1)	M(1)	Q	1, 2

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Undervoltage relay operability is to be verified independently. An inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.
(3) Verify undervoltage relay operability and alarm annunciation only. Do not maintain an undervoltage condition beyond the setpoint for the time delay relay.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Fuel Building Isolation-Radioactivity-High and Criticality (ORE-AR055/56)	S	R#	M	*
2. Containment Isolation-Containment Radioactivity-High				
a) Unit 1 (1RE-AR011/12)	S	R#	M	All
b) Unit 2 (2RE-AR011/12)	S	R#	M	All
3. Gaseous Radioactivity-RCS Leakage Detection				
a) Unit 1 (1RE-PR011B)	S	R#	M	1, 2, 3, 4
b) Unit 2 (2RE-PR011B)	S	R#	M	1, 2, 3, 4
4. Particulate Radioactivity-RCS Leakage Detection				
a) Unit 1 (1RE-PR011A)	S	R#	M	1, 2, 3, 4
b) Unit 2 (2RE-PR011A)	S	R#	M	1, 2, 3, 4
5. Main Control Room Isolation-Outside Air Intake-Gaseous Radioactivity-High (ORE-PR031B/32B and ORE-PR033B/34B)	S	R#	M	All

*With new fuel or irradiated fuel in the fuel storage areas or fuel building.
#The specified 18 month interval may be extended to 32 months for cycle 1 only.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Intermediate Range Neutron Flux	M	R#
2. Source Range Neutron Flux	M*	R#
3. Reactor Coolant Temperature - Wide Range	M	R#
4. Pressurizer Pressure	M	R#
5. Pressurizer Level	M	R#
6. Steam Generator Pressure	M	R#
7. Steam Generator Level	M	R#
8. RHR Temperature	M	R
9. Auxiliary Feedwater Flow Rate	M	R

*When below P-6.
#The specified 18 month interval may be extended to 32 months for cycle 1 only.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R#
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R#
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R#
4. Reactor Coolant Pressure - Wide Range	M	R#
5. Pressurizer Water Level	M	R#
6. Steam Line Pressure	M	R#
7. Steam Generator Water Level - Narrow Range	M	R#
8. Steam Generator Water Level - Wide Range	M	R#
9. Refueling Water Storage Tank Water Level	M	R#
10. Auxiliary Feedater Flow Rate	M	R
11. PORV Position Indicator* (Open/Closed)	M	N.A.
12. PORV Block Valve Position Indicator** (Open/Closed)	M	N.A.
13. Safety Valve Position Indicator (Open/Closed)	M	N.A.
14. Containment Floor Drain Sump Water Level (Narrow Range)	M	R#
15. Containment Water Level (Wide Range)	M	R#
16. In Core Thermocouples	M	R#
17. Containment High Range Area Radiation	M	R***,#
18. Auxiliary Building Vent Stack - Wide Range Noble Gas	M	R#
19. Main Steam Line Radiation	M	R#
20. Reactor Vessel Water Level	M	R#

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

***CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release					
Liquid Radwaste Effluent Line (ORE-PR001)	D	P	R(3)#	Q(1)	N.A.
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release					
a. Essential Service Water					
1) Unit 1					
a) RCFC 1A and 1C Outlet (1RE-PR002)	D	M	R(3)#	Q(2)	N.A.
b) RCFC 1B and 1D Outlet (1RE-PR003)	D	M	R(3)#	Q(2)	N.A.
2) Unit 2					
a) RCFC 2A and 2C Outlet (2RE-PR002)	D	M	R(3)#	Q(2)	N.A.
b) RCFC 2B and 2D Outlet (2RE-PR003)	D	M	R(3)#	Q(2)	N.A.
b. Station Blowdown Line (ORE-PR010)	D	M	R(3)#	Q(2)	N.A.
3. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line (Loop-WX001)	D(4)	N.A.	R#	N.A.	Q
b. Station Blowdown Line (Loop-CW032)	D(4)	N.A.	R	N.A.	Q

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Plant Vent Monitoring System - Unit 1					
a. Noble Gas Activity Monitor - Providing Alarm					
1) High Range (1RE-PR028D)	D	M	R(3)##	Q(2)	*
2) Low Range (1RE-PR028B)	D	M	R(3)##	Q(2)	*
b. Iodine Sampler (1RE-PR028C)	D	M	R(3)##	Q(2)	*
c. Particulate Sampler (1RE-PR028A)	D	M	R(3)##	Q(2)	*
d. Effluent System Flow Rate Measuring Device (LOOP-VA019)	D	N.A.	R##	Q	*
e. Sampler Flow Rate Measuring Device (1FT-PR165)	D	N.A.	R##	Q	*
2. Plant Vent Monitoring System - Unit Two					
a. Noble Gas Activity Monitor- Providing Alarm					
1) High Range (2RE-PR028D)	D	M	R(3)##	Q(2)	*
2) Low Range (2RE-PR028B)	D	M	R(3)##	Q(2)	*
b. Iodine Sampler (2RE-PR028C)	D	M	R(3)##	Q(2)	*
c. Particulate Sampler (2RE-PR028A)	D	M	R(3)##	Q(2)	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Plant Vent Monitoring System - Unit Two (continued)					
d. Effluent System Flow Rate Measuring Device (LOOP-VA020)	D	N.A.	R##	Q	*
e. Sampler Flow Rate Measuring Device (2FT-PR165)	D	N.A.	R##	Q	*
3. Gaseous Waste Management System					
a. Hydrogen Analyzer (OAT-GW8000)	D	N.A.	Q(4)	M	**
b. Oxygen Analyzer (OAIT-GW004 and OAT-GW8003)	D	N.A.	Q(5)	M	**
4. Gas Decay Tank System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ORE-PR002A and 2B)	P	P	R(3)##	Q(1)	*
5. Containment Purge System					
a. Noble Gas Activity Monitor - Providing Alarm (RE-PR001B)	D	P	R(3)##	Q(2)	*
b. Iodine Sampler (RE-PR001C)	P	P	R(3)##	N.A.	*
c. Particulate Sampler (RE-PR001A)	P	P	R(3)##	N.A.	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS#

FUNCTIONAL UNIT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Radioactivity Monitors Providing Alarm and Automatic Closure of Surge Tank Vent Component Cooling Water Line (ORE-PR009 and RE-PR009)	D	M	R(3)##	Q(1)	*

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

*At all times.
**During WASTE GAS HOLDUP SYSTEM operation.
#All instruments required for Unit 1 or Unit 2 operation.
##The specified 18 month interval may be extended to 32 months for cycle 1 only.

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (monitor loss of communications - alarm only, detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out-of-service, or

e. Monitor loss of sample flow.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing hydrogen and nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing oxygen and nitrogen.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: 1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and 2) apply the ACTION of b. or c. above, as appropriate for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION,# and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. of Specification 3.4.4.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE
LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Floor Drain and Reactor Cavity Flow Monitoring System, and

c. The Containment Gaseous Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

   1) Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

   2) Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

   3) Perform a Reactor Coolant System water inventory balance at least once per 8 hours.

   Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Floor Drain and Reactor Cavity Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months,# and

c. Verify the oil separator portion of the containment floor drain collection sump has been filled to the level of the overflow to the containment floor drain unidentified leakage collection weir box once per 18 months, following refueling, and prior to initial startup.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months;# and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve RH8702A is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RH8702B is open.

b. For RHR suction relief valve 8708A:

1) By verifying at least once per 31 days that RH8701B is open with power to the valve operator removed, and

2) By verifying at least once per 12 hours that RH8701A is open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution, and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the MCC compartment is open and tagged out of service.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months# by the performance of a CHANNEL CALIBRATION.

#The specified 18 month interval may be extended to 32 months for cycle 1 only.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. NPF-66

AND AMENDMENT NO. 2 TO FACILITY OPERATING LICENSE NO. NPF-72

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated September 3, 1987, Commonwealth Edison (the licensee) submitted a request for revision of the Technical Specification, for Byron Station, Unit 2, and Braidwood Station, Units 1 and 2. The proposed revision would allow for a one time extension of the interval for performing certain 18-month instrument surveillances until the first refueling outages for Byron 2 and Braidwood 1 and 2. Byron Station, Unit 1 is not affected by the requested changes, since it has completed its first refueling outage. For Byron 2, the time period from the first pre-operational surveillance to the end of the first refueling outage is anticipated to be 32 months. Thus, the licensee had to justify an extension to 32 months on certain instrument surveillances. For consistency in the Technical Specifications, 32 months was requested for all the instrument surveillances on all three units.

2.0 EVALUATION

The proposed Technical Specification changes include instrumentation from the Reactor Trip System, Engineered Safety Features Actuation System, Radiation Monitoring, Remote Shutdown Monitoring, Accident Monitoring, Radioactive Liquid Effluent Monitoring, Radioactive Gaseous

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Effluent Monitoring, Power Distribution Limits, Reactor Coolant System and the Emergency Core Cooling System. Not all of the instrumentation in these systems are included in the proposed change. The September 3, 1987 submittal denotes the specific instrumentation that is included.

The current Technical Specifications are consistent with the Standard Technical Specifications in that many surveillance tests are at 18-month intervals which is the maximum anticipated interval between refueling. Some variation is allowed by Specification 4.0.2 which permits a 25% extension up to a maximum of 3.25 x 18 months for three consecutive surveillances.

During normal plant startup testing and required low power operation, the fuel burnup rate is much lower than full power operation. The licensee noted that the first fuel cycle for Byron Unit 1 was 24 months. For Byron Unit 1, the licensee took a six week surveillance outage and met the Technical Specification 18-month surveillance requirements. The licensee is requesting this amendment to prevent extended surveillance outages for the other three units' first fuel cycle. The licensee has stated in their submittal that due to the limitations on the number of people and testing equipment that can physically be used to work on an instrument rack at one time and since some surveillance must be performed sequentially, up to a 21-day outage is required using three work shifts a day.

The licensee safety evaluation notes that although the 18-month channel calibration will be extended, the shiftly, monthly, and quarterly testing will continue. These functional tests will detect degradation or inoperability of the channels and include the analog/digital channel operational test, channel check, source and trip actuating device operational test. The results of these tests are compared with the operational allowables (which are more conservative than the Technical Specifications) and if the allowables are exceeded a work request is issued and the channel is brought back

to specifications. These checks will not detect drift of the sensor instrument. Any channels that are out of the Technical Specification allowables will put the plant into the appropriate action statement.

As part of requested surveillance changes at Byron Station, Unit 1, the licensee trended instrument drift for approximately two years. This information was compared to the rack drift allowances outlined in the Westinghouse Statistical Setpoint Study. The staff has previously reviewed and accepted the Westinghouse setpoint methodology. The Byron study demonstrated that over 99 percent of all data taken remained within the drift tolerances of the Westinghouse Statistical Setpoint Study. Since the normal quarterly surveillances do not include the process sensors, sensors for the steam generator pressure, narrow and wide range steam generator level, pressurizer level and pressure, auxiliary feedwater suction pressure, RWST level, containment pressure and reactor coolant system loop flow were included to get drift information on the sensors. Of the channels found outside the allowables during this two year surveillance, it was noted that all turbine trip emergency trip header pressure channels were outside the limits and therefore Byron 2 and the Braidwood Stations are not requesting surveillance extensions for those channels. The remaining sensors demonstrated minimal drift that the trending showed to be random in nature.

The licensee has demonstrated that, for the instrumentation channels requested, the magnitude of undetected instrument drift will be small and have no effect on the probability of previously evaluated accidents. The Byron 1 study is applicable to Byron 2 and Braidwood 1 and 2. Deferral of the 18-month channel calibration to 32 months may increase the potential for undetected instrument drift but the functional tests will continue to be performed and will detect instrument failures.

The instrumentation channels have been calibrated at different times during startup testing such that for Byron 2 over half of instruments normal 18-month surveillance due date is within six months of the scheduled refueling outage.

The staff has concluded that delaying these instrument surveillances will not provide significant reduction in the accuracy or availability of the instrument channels. The staff has concluded based on the considerations discussed above that: (1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulation. The staff finds the requested one-time increase from 18 months to 32 months in the cycle one refueling surveillance interval for the instrumentation listed in the submittal to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these amendments involve (i) no significant harards, (ii) no significant increase in the amounts, and (iii) 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. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

We have 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Stewart
L. Olshan

Dated: November 25, 1987