September 9, 1987

Docket Nos. STN 50-454
and STN 50-455

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

Dear Mr. Butterfield:

SUBJECT: AMENDMENT NO. 10 TO THE BYRON TECHNICAL SPECIFICATIONS (TAC NOS. 62730, 62865 AND 63269)

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. NPF-37 and Amendment No. 10 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letters dated August 29, 1986 and September 15, 1986, supplemented March 24, 1987.

These amendments revise condition 2.C.(6) of NPF-37 and condition 2.E of NPF-66 and remove the Fire Protection Technical Specifications from Appendix A of those licenses. The amendments also revise Technical Specification Table 4.3-1 to replace the requirement to measure the source range neutron flux instrumentation high voltage plateau curve with a requirement to measure source range neutron flux instrumentation discriminator bias curves.

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,

/S/

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

Enclosures:
1. Amendment No. 10 to NPF-37
2. Amendment No. 10 NPF-66
3. Safety Evaluation

cc: w/enclosures
See next page

(*See Previous Concurrence)

PD3-2:LA	PD3-2:PM	PD3-2:D	OGC
*LLuther/ww	*LOlshan	DMuller	*RFonner
9/2 /87	9/ /87	9/9 /87	8/25/87

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P PDR

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These amendments revise condition 2.C.(6) of NPF-37 and condition 2.E of NPF-66 and remove the Fire Protection Technical Specifications from Appendix A of those licenses. The amendments also revise Technical Specification Table 4.3-1 to replace the requirement to measure the source range neutron flux instrumentation high voltage plateau curve with a requirement to measure source range neutron flux instrumentation discriminators bias curves.

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.

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8/2/87	8/ /87	8/ /87	8/25/87

Mr. L. D. Butterfield, Jr.
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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. 10
License No. NPF-37

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated August 29, 1986 and September 15, 1986, supplemented March 24, 1987 comply 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 paragraphs 2.C(2) and 2.C(6) of Facility Operating License No. NPF-37 are 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. 10 and the Environment Protection Plant 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.

(6) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report through Amendment 8, and as approved in the SER dated February 1987 through Supplement No 8, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 9, 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. 10
License No. NFP-66

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated August 29, 1986 and September 15, 1986, supplemented March 24, 1987 comply 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 paragraphs 2.C(2) and 2.E of Facility Operating License No. NPF-66 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix (NUREG-1113), as revised through Amendment No. 10 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 Technical Specifications and the Environment Protection Plan.

E. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report through Amendment 8, and as approved in the SER dated February 1987 through Supplement No. 8, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 9, 1987

ATTACHMENT TO LICENSE AMENDMENT NOS. 10 AND 10

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

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

Revise Appendix A as follows:

Remove Pages	Insert Pages
V*	V*
VI	VI
VII	VII
VIII*	VIII*
XI	XI
XII*	XII*
XVII	XVII
XVIII*	XVIII*
3/4 3-9	3/4 3-9
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6-15*	6-15*
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* Overleaf pages added for convenience

** Pages 3/4 3-56 through 3/4 3-63 were removed. Pages 3/4 3-64 through 3/4 3-81 have been renumbered to replace missing pages

*** Pages 3/4 7-30 through 3/4 7-47 were removed. Pages 3/4 7-48 and 3/4 7-49 have been renumbered to replace missing pages.

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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.

##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 4.3-1 (Continued)

TABLE NOTATIONS

(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.

(12) At least once per 18 months during shutdown verify that on a simulated Boron Dilution Doubling test signal CVCS valves 112D and E open and 112B and C close within 30 seconds.

(13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

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.

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and DIGITAL and ANALOG CHANNEL OPERATIONAL TEST at the fre-quencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
Liquid Radwaste Effluent Line (ORE-PR001)	1	31
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)	1	32
b) RCFC 1B and 1D Outlet (1RE-PR003)	1	32
2) Unit 2		
a) RCFC 2A and 2C Outlet (2RE-PR002)	1	32
b) RCFC 2B and 2D Outlet (2RE-PR003)	1	32
b. Station Blowdown Line (ORE-PR010)	1	32
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (Loop-WX001)	1	33
b. Station Blowdown Line (Loop-CW032)	1	33

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

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

(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.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3, and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION#

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Plant Vent Monitoring System - Unit 1			
a. Noble Gas Activity Monitor-Providing Alarm			
1) High Range (1RE-PR028D)	1	*	39
2) Low Range (1RE-PR028B)	1	*	39
b. Iodine Sampler (1RE-PR028C)	1	*	40
c. Particulate Sampler (1RE-PR028A)	1	*	40
d. Effluent System Flow Rate Measuring Device (LOOP-VA019)	1	*	36
e. Sampler Flow Rate Measuring Device (1FT-PR165)	1	*	36
2. Plant Vent Monitoring System - Unit 2			
a. Noble Gas Activity Monitor-Providing Alarm			
1) High Range (2RE-PR028D)	1	*	39
2) Low Range (2RE-PR028B)	1	*	39
b. Iodine Sampler (2RE-PR028C)	1	*	40
c. Particulate Sampler (2RE-PR028A)	1	*	40
d. Effluent System Flow Rate Measuring Device (LOOP-VA020)	1	*	36
e. Sampler Flow Rate Measuring Device (2FT-PR165)	1	*	36

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION#

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. Gaseous Waste Management System			
a. Hydrogen Analyzer (OAT-GW8000)	1	**	38
b. Oxygen Analyzer (OAIT-GW004 and OAT-GW8003)	2	**	38
4. Gas Decay Tank System			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ORE-PR002A and 2B)	2	*	35
5. Containment Purge System			
a. Noble Gas Activity Monitor - Providing Alarm (RE-PR001B)	1	*	37
b. Iodine Sampler (RE-PR001C)	1	*	40
c. Particulate Sampler (RE-PR001A)	1	*	40
6. Radioactivity Monitors Providing Alarm and Automatic Closure of Surge Tank Vent-Component Cooling Water Line (ORE-PR009 and RE-PR009)	2	*	41

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

*At all times.
**During WASTE GAS HOLDUP SYSTEM operation.
#All instruments required for Unit 1 or Unit 2 operation.

ACTION STATEMENTS

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 38 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

TABLE 3.3-13 (Continued)

ACTION STATEMENTS (Continued)

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml.

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.

(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.

INSTRUMENTATION

HIGH ENERGY LINE BREAK ISOLATION SENSORS

LIMITING CONDITION FOR OPERATION

3.3.3.11 The high energy line break instrumentation shown in Table 3.3-14 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-14.

ACTION:

a. With the number of OPERABLE auxiliary steam isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-14, restore the inoperable instrument(s) to OPERABLE status within 7 days, or suspend the supply of auxiliary steam to the Auxiliary Building, or establish a continuous watch in the affected area(s) until the inoperable sensors are restored to OPERABLE status.

b. With the number of OPERABLE steam generator blowdown line isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-14, restore the inoperable instrument(s) to OPERABLE status within 7 days, or limit the total steam generator blowdown flow rate to less than or equal to 60 gpm or establish a continuous watch in the affected area(s) until the inoperable sensors are restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.12 Each of the above high energy line break isolation instruments shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST AND CHANNEL CALIBRATION at least once per 18 months.

Table 3.3-14

HIGH ENERGY LINE BREAK INSTRUMENTATION

Isolation Function	Instrument Channel	Minimum Channels OPERABLE	Applicable MODES
1. Auxiliary Steam Isolation	OTS-AS031A OTS-AS032A	1	*
	OTS-AS031B OTS-AS032B	1	*
	OTS-AS031C OTS-AS032C	1	*
	OTS-AS031D OTS-AS032D	1	*
	OTS-AS031E OTS-AS032E	1	*
	OTS-AS031F OTS-AS032F	1	*
2. Steam Generator Blowdown Line Isolation	TS-SD045A TS-SD045B	1	1, 2, 3, 4
	TS-SD046A TS-SD046B	1	1, 2, 3, 4
	TS-SD045C TS-SD045D	1	1, 2, 3, 4
	TS-SD046C TS-SD046D	1	1, 2, 3, 4

*Required when auxiliary steam is being supplied, from any source, to the Auxiliary Building.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one throttle valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. During turbine operation at least once per 31 days by direct observation of the movement of the valves below through one complete cycle from the running position:

   1) Four high pressure turbine throttle valves,
   2) Four high pressure turbine governor valves,
   3) Six turbine reheat stop valves, and
   4) Six turbine reheat intercept valves.

b. Within 7 days prior to entering MODE 3 from MODE 4, by cycling each of the 12 extraction steam nonreturn check valves from the closed position,

c. During turbine operation at least once per 31 days by direct observation, of freedom of movement of each of the 12 extraction steam nonreturn check valve weight arms,

d. At least once per 18 months by performance of CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

e. At least once per 40 months by disassembling at least one of each of the valves given in Specifications 4.3.4.2a. and b. above, and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature limit of each area shown in Table 3.7-6 shall not be exceeded for more than 8 hours, or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above, and within 4 hours either return the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-6

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Misc. Electric Equipment and Battery Rooms	108
2.	ESF Switchgear Rooms	108
3.	Division 12 for Unit 1 (Division 22 for Unit 2) Cable Spreading Room	108
4.	Upper Cable Spreading Rooms	90
5.	Diesel-Generator Rooms	132
6.	Diesel Oil Storage Rooms	132
7.	Aux. Building Vent Exhaust Filter Cubicle	122
8.	Centrifugal Charging Pump Rooms	122
9.	Containment Spray Pump Rooms	130
10.	RHR Pump Rooms	130
11.	Safety Injection Pump Room	130
12.	Control Room	90
13.	Lower Cable Spreading Rooms	108

SEISMIC INSTRUMENTATION (Continued)

The response spectrum analyzer computes the response spectrum of the event for two sensor locations, compares it to the design response spectra of the plant, and indicates whether the event exceeded the operating basis earthquake criteria or the safe shutdown earthquake criteria.

This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquake," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG 0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

3/4.3.3.11 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The OPERABILITY of the high energy line break isolation sensors ensures that the capability is available to promptly detect and initiate protective action in the event of a line break. This capability is required to prevent the potential for damage to safety-related systems and structures in the auxiliary building.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY.

6.1 RESPONSIBILITY

6.1.1 The Station Manager, Byron Station, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Engineer (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Assistant Vice President and General Manager-Nuclear Stations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1; and

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. A Radiation Chemistry Technician,* qualified in radiation protection procedures, shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

*The Radiation Chemistry Technician may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, health physics personnel, equipment operators, and key maintenance personnel.

The amount of overtime worked by Unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

6.2.3 ONSITE NUCLEAR SAFETY GROUP (ONSG)

FUNCTION

6.2.3.1 The ONSG serves as an independent safety engineering group and shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The ONSG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, and the Station Manager, Byron Station.

COMPOSITION

6.2.3.2 The ONSG shall be composed of at least four, dedicated, full-time engineers located on site.

RESPONSIBILITIES

6.2.3.3 The ONSG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ONSG shall be prepared, maintained, and forwarded each calendar month to the Manager of Nuclear Safety, and the Station Manager, Byron Station.

6.2.4 SHIFT TECHNICAL ADVISOR

The Station Control Room Engineer (SCRE) may serve as the Shift Technical Advisor (STA) during abnormal operating or accident conditions. During these conditions the SCRE or other on duty STA shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

*Not responsible for sign-off function.

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 Supervisor 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 Supervisor of the Offsite Review and Investigative Function;

8) Review of the Station Security Plan and implementing procedures and submittal of recommended changes to the Assistant Vice President and General Manager - Nuclear Stations;

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 Supervisor 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

ONSITE (Continued)

of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Assistant Vice President and General Manager - Nuclear Stations and the Supervisor of the Offsite Review and Investigative Function.

d. Records

1) Reports, reviews, investigations, and recommendations shall be documented with copies to the Assistant Vice President and General Manager - Nuclear Stations, the Supervisor of the Offsite Review and Investigative Function, the Station Manager, and the Manager of Quality Assurance.

2) Copies of all records and documentation shall be kept on file at the station.

e. Procedures

Written administrative procedures shall be prepared and maintained for conduct of the Onsite Review and Investigative Function. These procedures shall include the following:

1) Content and method of submission and presentation to the Station Manager, Assistant Vice President and General Manager - Nuclear Stations, and the Supervisor of the Offsite Review and Investigative Function,

2) Use of committees,

3) Review and approval,

4) Detailed listing of items to be reviewed,

5) Procedures for administration of the quality control activities, and

6) Assignment of responsibilities.

f. Personnel

1) The personnel performing the Onsite Review and Investigative Function, in addition to the Station Manager, shall consist of persons having expertise in:

a) Nuclear power plant technology,
b) Reactor operations,
c) Reactor engineering,
d) Chemistry
e) Radiological controls,
f) Instrumentation and control, and
g) Mechanical and electrical systems.

ONSITE (Continued)

2) Personnel performing the Onsite Review and Investigative Function shall meet minimum acceptable levels as described in ANSI N18.1-1971, Sections 4.2 and 4.4.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Onsite Review and Investigative Function and the results of this review shall be submitted to the Offsite Review and Investigative Function and the Assistant Vice President and General Manager - Nuclear Stations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Assistant Vice President and General Manager - Nuclear Stations and the Offsite Review and Investigative Function shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Onsite Review and Investigative Function. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the Offsite Review and Investigative Function and the Assistant Vice President and General Manager - Nuclear Stations within 14 days of the violation; and

d. Critical operation of the Unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

PROCEDURES AND PROGRAMS (Continued)

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33;

c. Plant Security Plan implementation,

d. Radiological Emergency Response Plan implementation,

e. PROCESS CONTROL PROGRAM implementation,

f. OFFSITE DOSE CALCULATION MANUAL implementation,

g. Quality Assurance Program implementation for effluent and environmental monitoring, and

h. Fire Protection Program implementation.

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above, may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the Unit affected; and

c. The change is documented, reviewed by the Onsite Review and Investigative Function, and approved by the Station Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NO. NPF-37

AND AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NO. NFP-66

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated August 29, 1986, Commonwealth Edison (the licensee) requested a revision to the Fire Protection License Conditions, 2.C(6) in NPF-37 and 2.E in NPF-66, and requested removal of the Fire Protection Technical Specifications. By letter dated September 15, 1986, the licensee requested an amendment that replaces the requirement to measure the source range neutron flux instrumentation high voltage plateau curve with a requirement to measure source range neutron flux instrumentation discriminator bias curves. The March 24, 1987 letter was submitted in response to the NRC's February 5, 1987 request to clarify the original submittal and does not contain substantive changes.

2.0 EVALUATION

2.1 Fire Protection

Generic Letter 86-10, "Implementation of Fire Protection Requirements," dated April 24, 1986, provided guidance for a licensee to request an amendment to delete the Fire Protection Technical Specifications. The licensee, in its August 29, 1986, used that guidance and proposed a revision to the Fire Protection license condition and proposed an amendment that would remove the Fire Protection Technical Specifications.

The licensee stated that the fire protection program had been incorporated into the FSAR by reference, including limiting conditions for operation, action statements, and surveillance requirement for the fire protection program. The licensee also stated that its procedures provide a level of protection equivalent to that in the present Technical Specifications. Therefore, the staff concludes that the proposed revision to the license condition and the removal of the Fire Protection Technical Specifications are acceptable.

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2.2 Source Range Instrumentation

Source range neutron detectors monitor leakage flux at the lowest reactor shutdown levels and provide output to indicating alarms and reactor trips. Source range detectors used in Westinghouse reactors are of the boron triflouride (BF_3) type. These detectors emit a pulse rate output which is proportional to the amount of thermal neutron flux to which they are exposed. These pulses are conditioned by electronic circuitry which provides amplification and a means to discriminate between neutron flux and other types of radiation, primarily gamma. Gamma radiation stimulates the sensors to generate lower amplitude pulses. These low-level pulses are removed from the detector signal by use of a pulse amplifier-discriminator.

The current channel calibration requirements of Technical Specification(TS) Table 4.3-1 call for developing high voltage plateau curves for each BF_3 detector and comparing these curves with the manufacturer's recommendation. These curves are obtained by varying the supply voltage to the detectors and recording the corresponding change in voltage plateau (counting rate). Any appreciable deviation in the curve's profile from that of the manufacturer's recommendation indicates probable detector degradation.

Obtaining discriminator bias curves evolves a markedly different technique. This is done by varying the discriminator bias over a specified range of voltages and recording corresponding changes in the count rate. A graph of count rate versus discriminator bias is then made. The discriminator bias operating point is determined from this graph and set on the channel. The count rate is then observed to determine if the correct operating point has been obtained. If the correct point has not been determined, the procedure is repeated with increased voltage. Detector degradation is determined by a downward shift in the discriminator bias curve from that of previous curves. This downward shift results because pulses detected by failing sensors have lower amplitudes; thereby, reducing the actual number of pulses counted at various bias voltages.

The intent of TS Table 4.3-1 functional Unit 6 is to assure, as reasonably as possible, that means are available to periodically assess the condition of source range detectors. The TS refers to "voltage plateau curves" because these curves have traditionally been employed for the purpose. In the staff's view, there is no reason an alternate method cannot be used as long as the stated purpose is accomplished. The licensee has presented a convincing argument in support of the TS change. The staff finds acceptable the licensee's request to substitute "discriminator bias curves" for "voltage plateau curves" in TS Table 4.3-1, Functional Unit 6.

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 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. 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.

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:

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Dated: September 9, 1987