October 4, 1993

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

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

DISTRIBUTION:
Docket Files
PDIII-2 P/f
J. Hickman
C. Moore
D. Hagan
GHill(4)
C. Grimes
B. Clayton, RIII
OC/LFDCB
J. Roe
J. Zwolinski
J. Dyer
OGC
OPA
F. Paulitz
ACRS(10)
NRC & Local PDRs

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M84238 AND M84239)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 55 to Facility Operating License No. NPF-37 and Amendment No. 55 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated August 5, 1992.

The amendments revise the Byron Station, Units 1 and 2, Technical Specifications (TS) regarding the Engineered Safety Features Actuation System (ESFAS) instrumentation. The ESFAS, Functional Units, Analog Channel Operational Test interval is changed from monthly to quarterly. Eighteen changes to the Reactor Trip System (RTS) are also included in this TS change.

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

Sincerely,

Original Signed By:

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 55 to NPF-37
2. Amendment No. 55 to NPF-66
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC	
NAME	CMOORE	JHICKMAN	JDYER	RBachmann	
DATE	9/22/93	9/22/93	9/22/93	9/28/93	
COPY	YES/NO	YES/NO	YES/NO	YES/NO	

2 Copies

DF01

9310140280 931004
PDR ADOCK 05000454
P PDR

NRC FILE CENTER COPY

October 4, 1993

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

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

DISTRIBUTION:
Docket Files
PDIII-2 P/f
J. Hickman
C. Moore
D. Hagan
GHill(4)
C. Grimes
B. Clayton, RIII
OC/LFDCB
J. Roe
J. Zwolinski
J. Dyer
OGC
OPA
F. Paulitz
ACRS(10)
NRC & Local PDRs

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M84238 AND M84239)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 55 to Facility Operating License No. NPF-37 and Amendment No. 55 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated August 5, 1992.

The amendments revise the Byron Station, Units 1 and 2, Technical Specifications (TS) regarding the Engineered Safety Features Actuation System (ESFAS) instrumentation. The ESFAS, Functional Units, Analog Channel Operational Test interval is changed from monthly to quarterly. Eighteen changes to the Reactor Trip System (RTS) are also included in this TS change.

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

Sincerely,

Original Signed By:

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 55 to NPF-37
2. Amendment No. 55 to NPF-66
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PDIII-2	PM:PDIII-2	D:PDIII-2	OGC	
NAME	CMOORE	JHICKMAN	JDYER	RBachmann	
DATE	9/21/93	9/22/93	9/24/93	9/28/93	
COPY	YES/NO	YES/NO	YES/NO	YES/NO	

2 Copies

Mr. D. L. Farrar
Commonwealth Edison Company

Byron Station
Unit Nos. 1 and 2

cc:

Mr. William P. Poirier, Director
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355, Bay 236 West
Pittsburgh, Pennsylvania 15230

Joseph Gallo, Esquire
Hopkins and Sutter
888 16th Street, N.W.
Suite 700
Washington, D.C. 20006

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

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

U. S. Nuclear Regulatory Commission
Byron Resident Inspectors Office
4448 North German Church Road
Byron, Illinois 61010-9750

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

Chairman, Ogle County Board
Post Office Box 357
Oregon, Illinois 61061

Robert Neumann
Office of Public Counsel
State of Illinois Center
100 W. Randolph
Suite 11-300
Chicago, Illinois 60601

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

Douglass Cassel, Esquire
17 East Monroe Street, Suite 212
Chicago, Illinois 60603

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

Attorney General
500 South 2nd Street
Springfield, Illinois 62701

EIS Review Coordinator
EPA Region V
230 S. Dearborn Street
Chicago, Illinois 60604

Illinois Department of
Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
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 August 5, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

9310140281 931004
PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 55 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
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 August 5, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 55 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 55 AND 55

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

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

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Pages marked with an asterisk are provided for convenience.

Remove Pages	Insert Pages
*B 2-7	*B 2-7
B 2-8	B 2-8
B 2-9	B 2-9
3/4 3-2	3/4 3-2
3/4 3-3	3/4 3-3
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6
3/4 3-6a	3/4 3-6a
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 3-12a	3/4 3-12a
3/4 3-15	3/4 3-15
3/4 3-17	3/4 3-17
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
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
B 3/4 3-1	B 3/4 3-1

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 30% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. For undervoltage, the delay is set so that the time required for a signal to cause a reactor trip after the Undervoltage Trip Setpoint is reached shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to cause a reactor trip after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by P-8 (a power level of approximately 30% (P-8) of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 30% (P-8) of full power equivalent); and on increasing power, reinstated automatically by P-8.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position trips are anticipatory trips which provide core protection against DNB. The Open/Close Position trips assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Trip System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent) an automatic Reactor trip will occur if more than one reactor coolant pump breaker is opened. Below P-7 the trip function is automatically blocked.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range Reactor trip (i.e., prevents premature block of Source Range trip), provides an automatic backup block for Source Range Neutron Flux doubling, and the manual block that de-energizes the high voltage to the Source Range detectors. On decreasing power, Source Range Level trips and Neutron Flux doubling circuits are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops and Turbine trip. On decreasing power, the P-8 automatically blocks the single loop low flow trip and Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip; and automatically blocks the Source Range Reactor trip and provides an automatic backup function to de-energize the Source Range high voltage power. On decreasing power, the Intermediate Range Reactor trip and the Low Setpoint Power Range Reactor trip are automatically reactivated and Source Range high voltage to the detectors is restored if power decreases below the P-6 setpoint. Provides input to P-7.

P-13 Provides input to P-7.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature ΔT	4	2	3	1, 2	6
8. Overpower ΔT	4	2	3	1, 2	6
9. Pressurizer Pressure-Low (Above P-7)	4	2	3	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Pressurizer Pressure-High	4	2	3	1, 2	6
11. Pressurizer Water Level-High (Above P-7)	3	2	2	1	6
12. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	6
13. Steam Generator Water Level-Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operaing stm. gen.	1, 2	6
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
16. Turbine Trip (Above P-8)					
a. Emergency Trip Header Pressure	3/Train	2/Train	2/Train	1	6
b. Turbine Throttle Valve Closure	4	4	1	1	6
17. Safety Injection Input from ESF	2	1	2	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Coolant Pump Breaker Position Trip Above P-7	1/breaker	2	1/breaker per operating loop	1	11
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
20. Reactor Trip Breakers	2	1	2	1, 2	12
	2	1	2	3*, 4*, 5*	10
21. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
22. Reactor Trip Bypass Breakers	2	1	1	@,*	13

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the Reactor Trip System breakers in the closed position and the Control Rod Drive System capable of rod withdrawal.
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
@Whenever the Reactor Trip Bypass Breakers are racked in and closed for bypassing a Reactor Trip Breaker.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1; and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER with 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)
ACTIONS STATEMENTS (Continued)

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers, suspend all operations involving positive reactivity changes, and verify valves CV-111B, CV-8428, CV-8439, CV-8441, and CV-8435 are closed and secured in position. With no channels OPERABLE verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and take the actions stated above within 1 hour and verify compliance at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel amy be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - Deleted

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hour.

TABLE 3.3-1 (Continued)
ACTIONS STATEMENTS (Continued)

ACTION 12 - a. With one of the diverse trip features (Undervoltage or Shunt Trip attachment) inoperable, restore it to OPERABLE status with 48 hours or declare the breaker inoperable and apply the requirements of b below. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance and testing to restore the diverse trip feature to OPERABLE status.

b. With one of the Reactor Trip Breakers otherwise inoperable, be in at least HOT STANDBY within 6 hours; however, one Reactor Trip Breaker may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other Reactor Trip Breaker is OPERABLE.

ACTION 13 - With the Reactor Trip Bypass Breaker inoperable, restore the Reactor Trip Bypass Breaker to OPERABLE status prior to using the Reactor Trip Bypass Breaker to bypass a Reactor Trip Breaker. If the Reactor Trip Bypass Breaker is racked in and closed for bypassing a Reactor Trip Breaker and it becomes inoperable, be in at least HOT STANDBY within 6 hours. Restore the Bypass Breaker to OPERABLE status within the next 48 hours or open the Bypass Breaker within the following hour.

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(14)	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)	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(10)	N.A.	1
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	N.A.	Q(10)	N.A.	1
16. Turbine Trip (Above P-8)						
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 form 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)	R	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	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)	R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(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*
22. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	(15), R(16)	N.A.	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.

##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 31 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) The initial single point comparison of incore to excore AXIAL FLUX DIFFERENCE following a refueling outage shall be performed prior to exceeding 75% of RATED THERMAL POWER. Otherwise the single point comparison of incore to excore AXIAL FLUX DIFFERENCE shall be performed 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. For the purposes of this surveillance, monthly shall mean at least once per 31 EFPD. The 24 hour completion time provisions of Specification 4.0.3 are not applicable.

(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. For the purposes of this surveillance, quarterly shall mean at least once per 92 EFPD.

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

(8) Not used.

(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.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall be performed such that each train is tested at least every 62 days on a STAGGERED TEST BASIS and following maintenance or adjustment of the Reactor Trip Breakers and shall include independent verification of the OPERABILITY of the Undervoltage and Shunt Trip Attachments of the Reactor Trip Breakers.

(12) Not used.

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

(14) Verify that the appropriate signals reach the Undervoltage and Shunt Trip Relays, for both the Reactor Trip and Bypass Breakers from the Manual Trip Switches.

(15) Manual Shunt Trip prior to the Reactor Trip Bypass Breaker being racked in and closed for bypassing a Reactor Trip Breaker.

(16) Automatic Undervoltage trip.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
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	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-1	3	2	2	1, 2, 3	19
d. Pressurizer Pressure-Low (Above P-11)	4	2	3	1, 2, 3#	19
e. Steam Line Pressure-Low (Above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	19
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair	2 pair	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-3	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
2) System	2	1	2	1, 2, 3	22
b. Automation Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure-High-2	3	2	2	1, 2, 3	19
d. Steam Line Pressure-Low (above P-11)	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	19
e. Steam Line Pressure-Negative Rate-High (below P-11)	3/steam line	2/steam line any steam line	2/steam line	3##	19
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level-High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	19
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to Essential Service Water)	1/Train	1/Train	1/Train	1, 2, 3	15a
7. Automatic Opening of Containment Sump Suction Isolation Valves					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level - Low-Low Coincident With Safety Injection	4	2	3	1, 2, 3, 4	15
	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. ESF Bus Undervoltage	2/Bus	2/Bus	2/Bus	1, 2, 3, 4	25a
b. Grid Degraded Voltage	2/Bus	2/Bus	2/Bus	1, 2, 3, 4	25b

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	4-2/Train	2/Train	2/Train	1, 2, 3	22
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed or up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 15a - With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated pump INOPERABLE and take the ACTION required by Specification 3.7.1.2.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

ACTION STATEMENTS (Continued)

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - a. With the number of OPERABLE channels one less than the Minimum Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. The inoperable channel may be bypassed for up to 2 hours for surveillance testing of the OPERABLE channel per Specification 4.3.2.1.

b. With the number of OPERABLE channels one less than the Minimum Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

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	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low (Above P-11)	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low (Above P-11)	S	R	Q	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	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
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

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.a. Phase "A" Isolation (continued)								
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	Q	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
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	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low (Above P-11)	S	R	Q	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
4. Steam Line Isolation (continued)								
e. Steam Line Pressure - Negative Rate - High (Below P-11)	S	R	Q	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
b. Steam Generator Water Level-High-High (P-14)	S	R	Q	N.A.	M(1)	M(1)	Q	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	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Undervoltage-RCP Bus	N.A.	R	N.A.	Q(3)	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, 3)	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

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 of Containment Sump Suction Isolation Valves								
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	Q	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, 3)	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	Q	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	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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) Setpoint verification is not applicable.

INTENTIONALLY LEFT BLANK

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. NPF-37

AND AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. NPF-66

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

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

1.0 INTRODUCTION

By letter dated August 5, 1992, Commonwealth Edison Company (CECo) proposed changes to the Byron Station, Units 1 and 2, Technical Specifications (TS) for the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) instrumentation (Reference 1). The proposed changes for the RTS and ESFAS provide relief by reducing the frequency of surveillance testing, increasing the allowable time to place an inoperable channel on trip and/or bypassed condition, and increase the allowable time for surveillance and maintenance. This change would reduce inadvertent reactor trips, which cause unnecessary plant transients and challenges to the protection system. Increasing the time to place an inoperable channel in test and/or bypass and increasing the outage time for maintenance should reduce human errors associated with short time restraints.

2.0 BACKGROUND

The Westinghouse Owners' Group (WOG) was requested to evaluate their TS in accordance with the recommendations in NUREG-1024, "Technical Specification-Enhancing the Safety Impact" Task Group on Technical Specifications dated November 1983. By letter dated February 3, 1983, the WOG submitted Topical Report WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" (Reference 2). On October 4, 1983, WOG submitted Supplement 1 to WCAP-10271 to the NRC for review and included responses to staff questions submitted (Reference 3).

The staff completed the review of Topical Report WCAP-10271 and issued a Safety Evaluation (SE) on February 21, 1985 (Reference 4). The staff concluded in the SE that the Topical Report was acceptable regarding surveillance intervals for RTS analog channel operational tests, time which an inoperable RTS analog channel may be maintained in a non-tripped condition, and time the inoperable RTS channel may be bypassed.

By letters dated March 20, 1986, and May 12, 1987, WOG submitted Topical Reports WCAP-10271 Supplement 2 and WCAP-10271 Supplement 2, Revision 1,

9310140283 931004
PDR ADOCK 05000454
P PDR

proposing extensions of surveillance test intervals (STI) and test and maintenance allowed outage times (AOT) for the ESFAS (References 5 and 6).

On February 22, 1989, the staff completed review of the Topical Report and issued an SE. The staff concluded in the SE that the analysis presented in WCAP-10271 Supplement 2, and WCAP-10271 Supplement 2, Revision 1, augmented by a Brookhaven National Laboratory Technical Evaluation Report (TER) are acceptable for resolving the STI and AOT extension issues. However, the staff also required that the licensee confirm the applicability of the generic analysis to their facility and confirm that any increase in instrument drift due to the extended STI is accounted for in the Setpoint Calculation Methodology (Reference 7).

By letter dated April 30, 1990, the staff issued the supplemental SE of Topical Report WCAP-10271 Supplement 2, Revision 1. The staff concurred with the evaluation of the Topical Report by the Brookhaven National Laboratory; however, the staff also concluded that STI and AOT extensions proposed in the Topical Report for trip breakers are not acceptable (Reference 8).

3.0 EVALUATION

CECo proposed to make the following changes to the Byron, Units 1 and 2, TS. Changes are underlined, bold and in brackets [ ]. Clarification are in parentheses ( ):

Table 3.3-1 "REACTOR TRIP INSTRUMENTATION"

3.3-1(1) Note # **[The provisions of Specification 3.0.4 are not applicable.]** This notation is deleted from the following Functional Unit Action Statements:

2 Power Range, Neutron Flux
3 Power Range, Neutron Flux High Positive Rate
4 Power Range, Neutron Flux High Negative Rate
7 Over-Temperature Delta T
8 Overpower Delta T
9 Pressurizer Pressure-Low (Above P-7)
10 Pressurizer Pressure-High
11 Pressurizer Water Level-High (Above P-8)
12 Reactor Coolant Flow-Low
13 Steam Generator Water Level-Low-Low
14 Under-Voltage-Reactor Coolant Pumps (Above P-7)
15 Under-Frequency-Reactor Coolant Pumps (Above P-7)
16 Turbine Trip (Above P-7 or P-8)

This note was previously deleted as part of Amendment No. 49 to the Byron TS dated August 11, 1992.

3.3-1(2) Note ** [The boron dilution flux doubling signals may be blocked during reactor startup.] Notation is deleted from the following Functional Unit Applicable Modes:

6 Source Range, Neutron Flux a. Startup

This note was previously deleted as part of Amendment No. 51 to the Byron TS dated October 5, 1992.

3.3-1(3) Note *** [These channels also provide inputs to ESFAS. The Action Statement for the channels in Table 3.3-3 is more conservative and, therefore, controlling.] Notation is deleted from the following Functional Unit Action Statements:

9 Pressurizer Pressure-Low (Above P-7)
13 Steam Generator Water Level-Low-Low
14 Undervoltage-Reactor Coolant Pumps (Above-P-7)

This note is deleted because the Action Statement for these Functional Units in Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation" is changed to agree with those in Table 3.3-1; therefore, the note is no longer applicable.

3.3-1(4) Note **** [A reactor trip on turbine trip is enabled above P-7 (10%) until the modification is implemented which enables reactor trip on turbine trip above P-8 (30%).] Notation is deleted from the following Functional Unit and [Above P-7] is also deleted:

16 Turbine Trip

The Byron, Units 1 and 2, Final Safety Analysis Report (FSAR) Section 10.4.4, "Steam Dump (or ByPass) System" states: "a reactor power level step change of 10% can be handled by the reactor control system without causing a reactor trip and the condenser steam dump (bypass) can handle a 40% power reduction; therefore, the plant can handle a 50% turbine load rejection without a reactor trip."

This note is deleted because of a modification where the reactor trip on a turbine trip occurs only when power is above P-8 (30%). On September 30, 1987, the licensee requested an amendment to the Facility Operating License for the above modification (Reference 10). The NRC staff requested CECo to address a concern regarding the potential increase in probability of a stuck-open pressurizer power relief valve as a result of implementing the above modification. The licensee responded to the NRC concern by letter dated October 30, 1987 (Reference 11). The staff accepted the above modification in the Safety Evaluation for Amendment No. 13 issued December 8, 1987 (Reference 12). Since the modification has been completed Note **** is no longer needed and is deleted.

3.3-1(5) Action Statement 9 is changed and is applicable to the following Functional Units:

17 Safety Injection Input from ESF
21 Automatic Trip and Interlock Logic

Action Statement 9:

With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, [restore the inoperable channel to OPERABLE status within 6 hours, or] be in at least HOT STANDBY within [the next] 6 hours; however, one channel may be bypassed for up to [4] hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

This change agrees with the NRC's SE (Reference 4) conclusion.

3.3-1(6) The following Functional Unit Action Statement 11 is changed:

18 Reactor Coolant Pump Breaker Position Trip

Action Statement 11:

With the number of OPERABLE channels less than the Total No. of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within [6] hours.

This change agrees with the NRC's SE (Reference 4) conclusion.

Table 4.3-1 "REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS"

4.3-1(1) Note ** [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]. Notation is deleted from the Analog Channel Operational Test of the following Functional Units:

9 Pressurizer Pressure-Low (Above P-7)
13 Steam Generator Water Level-Low-Low

The Note ** is also removed from the Trip Actuating Device Operational Test for the following Functional Unit:

14 Undervoltage-Reactor Coolant Pumps (Above-P-7)

This note is deleted because the operational test frequency for these Functional Units is being changed in Table 4.3-2 from Monthly to Quarterly and now agree with Table 4.3-1; therefore, it is no longer applicable.

4.3-1(2) Note *** [A reactor trip on turbine trip is enabled above P-7 (10%) until the modification is implemented which enables reactor trip on turbine trip above P-8 (30%).] Notation is deleted from the following Functional Unit and [Above P-7] is also deleted:

16 Turbine Trip

This note is deleted because of a modification where the reactor trip on a turbine trip only occurs when above P-8 (30%) power.

See change 3.3-1(4).

4.3-1(3) Note # [The specified 18 month interval may be extended to 32 months for cycle 1.] Notation is deleted from Channel Calibration of the following Functional Units:

2 Power Range, Neutron Flux a. High Setpoint; b. Low Setpoint
3 Power Range, Neutron Flux High Positive Rate
4 Power Range, Neutron Flux High Negative Rate
5 Intermediate Range, Neutron Flux
6 Source Range, Neutron Flux
7 Overtemperature Delta T
8 Overpower Delta T
9 Pressurizer Pressure-Low (Above P-7)
10 Pressurizer Pressure-High
11 Pressurizer Water Level-High (Above P-7)
12 Reactor Coolant Flow-Low
13 Steam Generator Water Level-Low-Low
19 Reactor Trip System Interlocks
   a. Intermediate Range Neutron Flux, P-6
   b. Low Power Reactor Trip Block, P-7
   c. Power Range Neutron Flux, P-8
   d. Low Setpoint Power Range Neutron Flux, P-10
   e. Turbine Impulse Chamber Pressure, P-13

This note is deleted because it reflects a cycle-specific relief that is no longer applicable.

4.3-1(4) Note 1 is changed for Trip Actuating Device Operational Test for the following Functional Units:

16 Turbine Trip
   a. Emergency Trip Header Pressure
   b. Turbine Throttle Valve Closure

Note 1:

If not performed in previous [31] days.

This change agrees with the NRC's SE (Reference 4) conclusion.

4.3-1(5) Note 3 for Monthly Channel Calibration for the following Functional Unit:

2 Power Range, Neutron Flux "a. High Setpoint"

Note 3:

[The initial single point comparison of incore to excore AXIAL FLUX DIFFERENCE following a refueling outage shall be performed prior to exceeding 75% of RATED THERMAL POWER. Otherwise the] [s]ingle point comparison of incore to excore AXIAL FLUX DIFFERENCE [shall be performed] 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. [For the purpose of this surveillance, monthly shall mean at least once per 31 EFPD (effective full power day). The 24 hour completion time provisions of Specification 4.0.3 are not applicable.]

This note was previously changed as part of Amendment No. 49 to the Byron TS dated August 11, 1992.

4.3-1(6) Note 6 is changed for Quarterly Channel Calibration test for Functional Unit:

2 Power Range, Neutron Flux a. High Setpoint

Note 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. [For the purpose of this surveillance, quarterly shall mean to least once per 92 EFPD.]

This note was previously changed as part of Amendment No. 49 to the Byron TS dated August 11, 1992.

4.3-1(7) Note 8 has been deleted from the Analog Channel Operational Quarterly Test for the following Functional Unit:

19 Reactor Trip System Interlocks
- b. Low Power Reactor Trip Block, P-7
- c. Power Range Neutron Flux, P-8
- d. Low Setpoint Power Range Neutron Flux, P-10
- e. Turbine Impulse Chamber Pressure, P-13

Note 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.]

This note has been removed because it is no longer used.

4.3-1(8) Analog Channel Operational Test frequency is changed from Quarterly to [Refueling Outage] for the following Functional Units:

19 Reactor Trip System Interlocks
- a. Intermediate Range Neutron Flux, P-6
- b. Low Power Reactor Trip Block, P-7
- c. Power Range Neutron Flux, P-8
- d. Low Setpoint Power Range Neutron Flux, P-10
- e. Turbine Impulse Chamber Pressure, P-13

This Note is changed because it has been generically approved in NRC's SE (Reference 4).

4.3-1(9) Note 9 is changed for Analog Channel Operational Test, for Functional Unit:

6 Source Range, Neutron Flux

Note 9:

Surveillance in MODES 3*, 4*, and 5* shall also include verification that permissive P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive window. Delete the following: [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.]

This note was previously changed as part of Amendment No. 51 to the Byron TS dated October 5, 1992.

4.3-1(10) NOTATION Note 12 is deleted for Channel Calibration of Functional Unit:

6 Source Range, Neutron Flux

Note 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.]

This note was previously deleted as part of Amendment No. 51 to the Byron TS dated October 5, 1992.

4.3-1(11) Note 14 is changed for Trip Actuating Device Operational Test of Functional Unit:

1 Manual Reactor Trip

Note 14:

Verify that the appropriate signals reach the Undervoltage and Shunt Trip Relays, for both the Reactor Trip and Bypass Breakers from the manual Trip Switches. Delete the following: [Initial performance of this surveillance requirement for Reactor Trip Bypass Breakers is to be completed prior to the startup following the third refueling outage for Unit 1 and the second refueling outage for Unit 2.]

This note was previously changed as part of Amendment No. 49 to the Byron TS dated August 11, 1992.

4.3-1(12) Note 16 is changed for Trip Actuating Device Operational Test of Functional Unit:

22 Reactor Trip Bypass Breakers

Note 16:

Automatic Undervoltage Trip. Delete the following: [Initial performance of this surveillance requirement is to be completed prior to the startup following the third refueling outage for Unit 1 and the second refueling outage for Unit 2.]

This note was previously changed as part of Amendment No. 49 to the Byron TS dated August 11, 1992.

Table 3.3-3 "ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION"

3.3-3(1) Functional Unit 9d, Steam Generator Water Level-High-High, is deleted because it is a duplication of Functional Unit 5b.

3.3-3(2) Note * [The provisions of Specification 3.0.4 are not applicable.] Notation has been deleted from the Action Statement of the following Functional Units:

1c Containment Pressure-High-1
1d Pressurizer Pressure-Low (above P-11)
1e Steam Line Pressure-Low (above P-11)
4c Containment Pressure-High-2
4d Steam Line Pressure-Low (above P-11)
4e Steam Line Pressure-Negative Rate-High (below P-11)
5b Steam Generator Water Level-High-High (P-14)
6g Auxiliary Feedwater Pump Suction Pressure-Low
8a ESF Bus Undervoltage
8b Grid Degraded Voltage

This note was previously deleted as part of Amendment No. 49 to the Byron TS dated August 11, 1992.

3.3-3(3) The following Functional Units Action Statement is changed from 15 to 19:

1c Containment Pressure-High 1
1d Pressurizer Pressure-Low (above P-11)
1e Steam Line Pressure-Low (above P-11)
4c Containment Pressure-High 2
4d Steam Line Pressure-Low (above P-11)
4e Steam Line Pressure-Negative Rate-High (below P-11)

Action Statement 15:

With the number of OPERABLE channels one less than the Total No. of Channels, operation may precede until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the trip condition within 1 hour.

Action Statement 19:

With the number of OPERABLE channels one less than the Total No. of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are met:

a. The inoperable channel is placed in the tripped condition within [6] hour [s] and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to [4] hours for surveillance testing of other channels per Specification 4.3.2.1.

This change is made because Action Statement 19 is in agreement NRC's SE (Reference 8).

3.3-3(4) Functional Unit 6g, Auxiliary Feedwater Pump Suction Pressure-Low, (Transfer to Essential Service Water) Total No. of Channels, Channels to Trip, and Minimum Channels Operable is changed from 2 to [1/Train]. Also the Action Statement is changed from 15 to 15a.

Action Statement 15:

With the number of OPERABLE channels one less than the Total No. of Channels, operation may precede until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the trip condition within 1 hour.

Action Statement 15a:

[With the number of OPERABLE channels one less than the Total No. of Channels, declare the associated pump INOPERABLE and take the ACTION required by Specification 3.7.1.2.]

Functional Unit 6g number of channels has been changed to agree with the as-built plant configuration of one pressure transmitter installed at the AF pump suction.

Action Statement 15 requires the inoperable channel be placed in a trip condition until the next required ANALOG CHANNEL OPERATION TEST. Although, a ESFAS signal is also required to cause the AF pump suction to transfer to the Essential Service Water System (SX) this armed condition could be present for up to a month. Therefore, should the AF pump receive a ESFAS signal there would be an inadvertent injection of SX water into the steam generators (SGs). The untreated SX water injection into the SGs would have a potential for causing future tube leaks or rupture. Therefore, Action Statement 15a requires that pump be declared inoperable with a 72 hour AOT.

3.3-3(5) Action Statement for the following Functional Unit is changed from 16 to 15:

7b RWST Level-Low-Low Coincident with Safety Injection (Auto opening of Containment Sump Suction Isolation Valves)

Action Statement 16:

With the number of OPERABLE channels one less than the Total No. of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels

OPERABLE requirement is met. One additional channel may be bypassed for up to [4] hours for surveillance testing per Specification 4.3.2.1.

Action Statement 15:

With the number of OPERABLE channels one less than the Total No. of Channels, operation may precede until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the trip condition within [6] hour [s].

This change to Functional Unit 7b, RWST-Low-Low Coincident with Safety Injection, Action Statement from 16 to 15 is proposed because the current design does not permit bypass of an inoperable channel which is required by Action Statement 16. The current design would require the use of jumpers which is not endorsed in the NRC's SE (Reference 4) as stated: "Testing of RTS Analog Signals in the bypassed conditions by use of temporary jumpers or by lifting leads is not acceptable." Therefore, the licensee proposes to place the inoperable channel in the tripped condition within 6 hours as required by Action Statement 15. Further, the time to place the inoperable channel in the test condition is changed from 1 hour to 6 hours which is in accordance with the conclusions of the SE (Reference 8). Although the WOG STS states: place the inoperable channel in bypass within 6 hours, the Action Statement 15, above, is for the trip condition which is more restricted.

3.3-3(6) Action Statement 14 for the following Functional Units is changed:

1b Automatic Actuation Logic And Actuation Relays (Safety Injection)
2b Automatic Actuation Logic and Actuation Relays (Containment Spray)
7a Automatic Actuation Logic and Actuation Relays (Recirculation)

Action Statement 14:

With the number of OPERABLE channels one less than the minimum Channels OPERABLE requirements, [restore the inoperable channel to OPERABLE status within 6 hours, or] be in at least HOT STANDBY within [the next] 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to [4] hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

This change agrees with the NRC's SE (Reference 8) conclusion.

3.3-3(7) Action Statement 16 for the following Functional Unit is changed:

2c Containment Pressure-High-3

Action Statement 16:

With the number of OPERABLE channels one less than the Total No. of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to [4] hours for surveillance testing per Specification 4.3.2.1.

This change agrees with the NRC's SE (Reference 8) conclusion.

3.3-3(8) Action Statement 21 for the following Functional Unit is changed:

4b Automatic Action Logic and Actuation Relays (Steam Line Isolation)

Action Statement 21:

With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, [restore the inoperable channel to OPERABLE status within 6 hours, or] be in at least HOT STANDBY within [the next] 6 hours and in at lest HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to [4] hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

This change agrees with the NRC's SE (Reference 8) conclusion.

3.3-3(9) Action Statement 24 for the following Functional Unit is changed:

5a Automatic Actuation Logic and Actuation Relays (Turbine Trip & Feedwater Isolation)

Action Statement 24:

With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirements, [restore the inoperable channel to OPERABLE status within 6 hours, or] be in at least HOT STANDBY within [the next] 6 hours; however, one channel may be bypassed for up to [4] hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

This change agrees with the NRC's SE (Reference 8) conclusion.

Table 4.3-2 "ESFAS INSTRUMENTATION SURVEILLANCE REQUIREMENTS"

4.3-2(1) Analog Channel Operational Test" (ACOT) interval is changed from Monthly to **[Quarterly]** for the following Functional Units:

1c Containment Pressure-High-1
1d Pressurizer Pressure-Low (above P-11)
1e Steam Line Pressure-Low (above P-11)
2c Containment Pressure-High-3
3b(3) Containment Pressure-High-3
4c Containment Pressure-High 2
4d Steam Line Pressure-Low (above P-11)
4e Steam Line Pressure-Negative Rate-High (below P-11)
5b Steam Generator Water Level-High-High (P-14)
6c Steam Generator Water Level-Low-Low
7b RWST Level-Low-Low Coincident with Safety Injection
9a Pressurizer Pressure, P-11
9c Low-Low T_{avg}, P-12

This change agrees with the NRC's SE (Reference 8).

4.3-2(2) Functional Unit 9d "Steam Generator Water Level-High-High" is removed from the table because it is a duplication with Functional Unit 5b.

4.3-2(3) Note # **[The specific 18 month interval may be extended to 32 months for Cycle 1 only.]** Notation is deleted from the following Channel Calibration Functional Units:

1c Containment Pressure-High-1
1d Pressurizer Pressure-Low (above P-11)
1e Steam Line Pressure-Low (above P-11)
2c Containment Pressure-High-3
3b(3) Containment Pressure-High-3
4c Containment Pressure-High 2
4d Steam Line Pressure-Low (above P-11)
4e Steam Line Pressure-Negative Rate-High (below P-11)
5b Steam Generator Water Level-High-High (P-14)
6c Steam Generator Water Level-Low-Low
7b RWST Level-Low-Low Coincident with Safety Injection
9a Pressurizer Pressure, P-11
9c Low-Low T_{avg}, P-12

The Note is deleted because the cycle-specific relief is no longer applicable.

4.3-2(4) Functional Unit 5b, Steam Generator Water Level-High-High (P-14), Actuation Logic Test and Master Relay Test is changed from Not Applicable to **[Monthly Note 1]** and the Slave Relay Test is changed from Not Applicable to **[Quarterly]**.

Note 1: "Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS."

This change agrees with the NRC's SE (Reference 8).

4.3-2(5) Functional Unit 6d, Undervoltage-RCP Bus, Trip Actuating Device Operational Test, is changed from Monthly to [Quarterly].

This change agrees with the NRC's SE (Reference 8).

The staff's SE (Reference 4) also required that the licensee confirm the applicability of the generic analysis to their facility and confirm that any increase in instrument drift due to the extended STI is accounted for in the Setpoint Calculation Methodology.

CECo has stated in their August 5, 1992, submittal that the generic analyses used in WCAP-10271 and Supplements is applicable to the Byron Station. The Byron Station uses the Westinghouse 7300 Process Control system and the Westinghouse Solid State Protection System (SSPS) for both the ESFAS and RTS. Both of these systems were specifically modelled in the generic analyses.

The ESFAS Functional Units implemented are all addressed by the generic analysis except, Functional Unit 7.b., Automatic Opening of Containment Sump Suction Valves - RWST Level-Low-Low Coincident with Safety Injection. This Functional Unit is addressed on a plant specific basis in the "Technical Specification Optimization Program, RWST Switchover Justification for Byron Nuclear Station" and it has been determined that the Functional Unit has a decrease in availability of less than 12%. Functional Units in the generic program and the evaluation determined that an increase in unavailability of less than 12% is acceptable.

CECo has implemented programs at the Byron Station to evaluate setpoint drift in accordance with WOG position given in the WOG Guidelines for Preparing Submittal Requesting Revisions of RPS TSs. The licensee has determined that the values used in the Byron Station setpoint methodology properly account for drift due to extended STIs.

The staff finds the proposed changes to Byron, Units 1 and 2, TS 3/4.3.1 and 3/4.3.2 acceptable because they are either deletions due to plant modifications, reflect a cycle-specific relief, are no longer applicable, or are in conformance with the WOG Topical Reports approved by the NRC (References 4 and 8) and reflected in the Westinghouse Standard Technical Specifications.

The generic analysis of WOG Topical Reports which formed the basis of increase of STI or AOT is applicable to the Byron Station and the licensee has adequately addressed any increase in instrument drift due to the extended STI.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off-site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 48816). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Paulitz

Date: October 4, 1993

7.0 REFERENCES

1. Letter from T.W. Simpkin (CECo) to T.E. Murley (NRC), "Byron Station Units 1 and 2 - Braidwood Station Units 1 and 2 Application for Amendment to Facility Operating License," August 5, 1992.

2. Westinghouse Letter CAW-83-7, submitting Topical Report WACP-10271, "Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," February 3, 1983.

3. Letter from J.J. Sheppard (WOG) to C. Thomas (NRC), "Supplement 1 to WCAP-10271," April 4, 1983.

4. Letter from Cecil O. Thomas (NRC) to J.J. Sheppard (WOG), Subject: "Acceptance for Referencing of Licensing Topical Report WCAP-10271, 'Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,'" February 21, 1985.

5. Westinghouse March 20, 1986, letter submitting G.R. Andre, R.C. Howard, R.L. Jansen, and K. Leonelli, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," WCAP-10271, Supplement 2, February 1986.

6. Westinghouse May 12, 1987, letter submitting G.R. Andre, R.C. Howard, R.L. Jansen, and K. Leonelli, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," WCAP-10271, Supplement 2, Revision 1, March 1987.

7. Letter from Charles E. Rossi (NRC) to Roger A. Newton (WOG), Subject: "Westinghouse Topical Reports WCAP-10271, Supplement 2 and WCAP-10271, Supplement 2, Revision 1, 'Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System,'" February 22, 1989.

8. Letter from Charles E. Rossi (NRC) to Gerard T. Goering (WOG), Subject: "Westinghouse Topical Reports WCAP-10271, Supplement 2, Revision 1, 'Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety features Actuation System,'" April 30, 1990.

9. Letter from J.B. Hickman (NRC) to T.J. Kovach (CECo), "Amendment No. 51 to Facility Operating Licenses No. NPF-37 and NPF-66 for Byron Station, Units 1 and 2," October 5, 1991.

10. Letter from K.A. Ainger (CECo) to NRC "Application for Amendment to Byron Unit 1 and 2 Facility Operating License," September 30, 1987.

11. Letter from K.A. Ainger (CECo) to NRC "Anticipatory Reactor Trip Upon Turbine Trip," October 30, 1987.

12. Letter from L.N. Olshan (NRC) to L.D. Butterfield, Jr. (CECo), "Amendment No. 13 to Allow Deletion of Reactor Trip on Turbine Trip Below 30% Power," December 8, 1987.