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

February 13, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO TECHNICAL SPECIFICATIONS FOR IMPLEMENTATION OF THE BEST ESTIMATE ANALYZER FOR CORE OPERATIONS NUCLEAR POWER DISTRIBUTION MONITORING SYSTEM (TAC NOS. MA8254, MA8255, MA8252, AND MA8253)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 116 to Facility Operating License No. NPF-37 and Amendment No. 116 to Facility Operating License No. NPF-66 for the Byron Station, Units 1 and 2, respectively, and Amendment No. 110 to Facility Operating License No. NPF-72 and Amendment No. 110 to Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively. The amendments are in response to Commonwealth Edison Company's (ComEd's) application dated February 15, 2000, and supplemented by letter dated July 26, 2000.

The proposed changes will allow the use of the Westinghouse core monitoring and support system known as Best Estimate Analyzer for Core Operations Nuclear (BEACON). The BEACON power distribution monitoring system (PDMS) has been developed to improve the operational support for pressurized water reactors (PWRs). BEACON is an advanced core monitoring and support system which uses current instrumentation in conjunction with a fully analytical methodology for on-line generation of three-dimensional (3-D) core power distributions. The system provides core monitoring, core measurements reduction, core analysis, and core predictions.

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,

/RA/

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 116 to NPF-37
2. Amendment No. 116 to NPF-66
3. Amendment No. 110 to NPF-72
4. Amendment No. 110 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

Distribution

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NAME	MCHAWLA*	GDICK	THARRIS	WBECKNER**	RWEISMAN	AMENDIOLA
DATE	12/13/00	01/30/01	01/30/01	09/14/00	01/17/01	~~01/ /01~~ 2/13/01

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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,

/RA/

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

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DATE	12/13/00	12/27/00	12/27/00	01/ /00	01/17/00	01/ /00

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Sincerely,

George F. Dick Jr.

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 116 to NPF-37
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5. Safety Evaluation

cc w/encls: See next page

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-37

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

   A. The application for amendment by the licensee dated February 15, 2000, and supplemented by letter dated July 26, 2000, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 116 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 13, 2001

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-66

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

   A. The application for amendment by the licensee dated February 15, 2000, and supplemented by letter dated July 26, 2000, 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. 116 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. 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 13, 2001

ATTACHMENT TO LICENSE AMENDMENT NOS. 116 AND 116

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.

Remove Pages	Insert Pages
3.1.4-2	3.1.4-2
3.1.4-3	3.1.4-3
3.1.4-4	3.1.4-4
3.1.7-1	3.1.7-1
3.2.1-1	3.2.1-1
3.2.1-2	3.2.1-2
3.2.1-3	3.2.1-3
3.2.1-4	3.2.1-4
3.2.1-5	3.2.1-5
---------	3.2.1-6
3.2.2-1	3.2.2-1
3.2.2-2	3.2.2-2
3.2.2-3	3.2.2-3
3.2.3-1	3.2.3-1
3.2.3-2	----------
3.2.3-3	----------
3.2.3-4	----------
3.2.4-1	3.2.4-1
3.2.4-3	3.2.4-3
3.2.4-4	3.2.4-4
---------	3.2.5-1
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-4	3.3.1-4
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.3.1-7	3.3.1-7
3.3.1-8	3.3.1-8
3.3.1-9	3.3.1-9
3.3.1-10	3.3.1-10
3.3.1-11	3.3.1-11
3.3.1-12	3.3.1-12
3.3.1-13	3.3.1-13
3.3.1-14	3.3.1-14
3.3.1-15	3.3.1-15

Remove Pages	Insert Pages
3.3.1-16	3.3.1-16
3.3.1-17	3.3.1-17
3.3.1-18	3.3.1-18
3.3.1-19	-----------
5.6-3	5.6-3
5.6-4	5.6-4
----------	5.6-5
----------	5.6-6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limit.	B.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours from discovery of Condition B concurrent with inoperability of Power Distribution Monitoring System (PDMS)
	<u>AND</u>	
	B.3 Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>	
	B.4 Determine Heat Flux Hot Channel Factor ($F_Q(Z)$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$).	72 hours
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.5	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. More than one rod not within alignment limit.	C.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	C.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>		
	C.2	Be in MODE 3.	6 hours from discovery of Condition C concurrent with inoperability of PDMS
	<u>AND</u>		
	C.3	---------NOTE--------- Only required to be performed when PDMS is OPERABLE. --------------------- Restore rod(s) to within alignment limit.	 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B or Required Action C.3 not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 550°F; and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each inoperable DRPI and each demand position indicator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable DRPIs.	Once per 8 hours
	<u>OR</u>		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Initiate action to verify the position of the rods with inoperable DRPIs.	Immediately
	<u>OR</u>		
	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_Q^C(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	<u>AND</u>	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
B. $F_Q^W(Z)$ not within limit.	B.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------NOTES------------------- 1. During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained. 2. Not required to be performed until 12 hours after declaring Power Distribution Monitoring System (PDMS) inoperable. Performance of SR 3.2.1.3 satisfies the initial performance of this SR after declaring PDMS inoperable. ---	
	Verify $F_Q^C(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-----NOTES----- 1. During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained. 2. If $F_Q^W(Z)$ measurements indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits specified in the COLR; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has not increased. ----------	(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued) -----NOTES----- 3. Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.4 satisfies the initial performance of this SR after declaring PDMS inoperable. ----- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F_Q^C(Z)$ is within limit specified in the COLR.	7 days
SR 3.2.1.4	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

| LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. -------------------- $F^N_{\Delta H}$ not within limit.	A.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.2 Determine $F^N_{\Delta H}$.	24 hours
	AND	
	A.3 Reduce Power Range Neutron Flux–High trip setpoints to ≤ 55% RTP.	72 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	--------NOTE-------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------------------	
		Determine $F^N_{\Delta H}$.	Prior to exceeding 50% RTP <u>AND</u> Prior to exceeding 75% RTP <u>AND</u> 24 hours after reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-----NOTE----- Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable. --------------- Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 31 Effective Full Power Days thereafter
SR 3.2.2.2	-----NOTE----- Only required to be performed when PDMS is OPERABLE. --------------- Verify $F^N_{\Delta H}$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD shall be maintained within the limits specified in the COLR.

-----------------------------NOTE-----------------------------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	-------------------NOTE------------------- Not required to be performed until 12 hours after declaring PDMS inoperable. -- Verify AFD is within limits for each OPERABLE excore channel.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP when Power Distribution Monitoring System (PDMS) is inoperable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	<u>AND</u>		
	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>		
	A.3	Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	--------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. ------------------------------- Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after exceeding the THERMAL POWER limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-------------------NOTES------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channel inputs can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. 3. Not required to be performed until 12 hours after declaring PDMS inoperable. --- Verify QPTR is ≤ 1.02 by calculation.	7 days
SR 3.2.4.2	-------------------NOTES------------------- 1. Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. 2. Not required to be performed until 12 hours after declaring PDMS inoperable. --- Verify QPTR is ≤ 1.02 using the movable incore detectors.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

LCO 3.2.5 DNBR shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limit.	A.1 Restore DNBR to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify DNBR is within limit specified in the COLR.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	------------NOTE------------ While this LCO is not met for Function 18, 19, or 20 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. ----------------------------	
	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2.1 Initiate action to fully insert all rods.	48 hours
	<u>AND</u>	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
D. One Power Range Neutron Flux-High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. ----------------------------	
	D.1 Place channel in trip.	6 hours
	<u>OR</u>	
	D.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	E.1 Place channel in trip.	6 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1 Suspend operations involving positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1 Open Reactor Trip Breakers (RTBs).	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	J.2.1 Initiate action to fully insert all rods.	48 hours
	<u>AND</u>	
	J.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
K. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	K.1 Place channel in trip.	6 hours
	<u>OR</u>	
	K.2 Reduce THERMAL POWER to < P-7.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One Turbine Trip channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---------------------------------	
	L.1 Place channel in trip.	6 hours
	OR	
	L.2 Reduce THERMAL POWER to < P-8.	12 hours
M. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------------------------------	
	M.1 Restore train to OPERABLE status.	6 hours
	OR	
	M.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One RTB train inoperable.	------------NOTES------------ 1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. 2. One RTB may be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. ------------------------------	
	N.1 Restore train to OPERABLE status.	1 hour
	<u>OR</u>	
	N.2 Be in MODE 3.	7 hours
O. One or more channels inoperable.	O.1 Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>	
	O.2 Be in MODE 3.	7 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
P. One or more channels inoperable.	P.1	Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>		
	P.2	Be in MODE 2.	7 hours
Q. One trip mechanism inoperable for one RTB.	Q.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	<u>OR</u>		
	Q.2	Be in MODE 3.	54 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE	FREQUENCY
SR 3.3.1.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------NOTES------------------ 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours
SR 3.3.1.3	-------------------NOTES------------------ 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Only required to be performed with THERMAL POWER > 15% RTP. --- Compare results of the incore measurements to NIS AFD.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	-------------------NOTE------------------- This Surveillance must be performed on the RTBB prior to placing the bypass breaker in service. -- Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.6	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 75% RTP. -- Calibrate excore channels to agree with incore measurements.	92 EFPD
SR 3.3.1.7	-------------------NOTE------------------- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. -- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-------------------NOTE------------------- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. --- Perform COT.	-----NOTE----- Only required when not performed within previous 92 days -------------- Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> Every 92 days thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	------------------NOTE------------------ Verification of setpoint is not required. --- Perform TADOT.	92 days
SR 3.3.1.10	------------------NOTE------------------ This Surveillance shall include verification that the time constants are adjusted to the prescribed values. --- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	------------------NOTE------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	------------------NOTE------------------ Verification of setpoint is not required. --- Perform TADOT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	-----NOTE------ Only required when not performed within previous 31 days -------------- Prior to reactor startup
SR 3.3.1.15	-------------------NOTE------------------- Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3[(a)], 4[(a)], 5[(a)]	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1[(b)],2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[(b)], 2[(c)]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2[(d)]	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3[(a)], 4[(a)], 5[(a)]	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1875 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2393 psig
9. Pressurizer Water Level - High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.5% of instrument span
10. Reactor Coolant Flow - Low (per loop)	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.3% of loop minimum measured flow
11. Reactor Coolant Pump (RCP) Breaker Position (per train)	1(e)	4	K	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	$1^{(e)}$	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4920 V
13. Underfrequency RCPs (per train)	$1^{(e)}$	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.08 Hz
14. Steam Generator (SG) Water Level – Low Low (per SG)					
a. Unit 1	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.1% of narrow range instrument span
b. Unit 2	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 34.8% of narrow range instrument span
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	$1^{(f)}$	3	L	SR 3.3.1.10 SR 3.3.1.14	≥ 910 psig
b. Turbine Throttle Valve Closure (per train)	$1^{(f)}$	4	L	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	M	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Source Range Block Permissive, P-6	$2^{(d)}$	2	O	SR 3.3.1.11 SR 3.3.1.12	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7					
(1) P-10 Input	1	3	P	SR 3.3.1.11 SR 3.3.1.12	NA
(2) P-13 Input	1	2	P	SR 3.3.1.10 SR 3.3.1.12	NA
c. Power Range Neutron Flux, P-8	1	3	P	SR 3.3.1.11 SR 3.3.1.12	≤ 32.1% RTP
d. Power Range Neutron Flux, P-10	1,2	3	O	SR 3.3.1.11 SR 3.3.1.12	≥ 7.9% RTP and ≤ 12.1% RTP
e. Turbine Impulse Pressure, P-13	1	2	P	SR 3.3.1.10 SR 3.3.1.12	≤ 12.1% turbine power
18. Reactor Trip Breakers (RTBs)$^{(g)}$	1,2	2 trains	N	SR 3.3.1.4	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	Q	SR 3.3.1.4	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	M	SR 3.3.1.5	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Source Range Block Permissive) interlock.

(g) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, ≤ *.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, = *.

K_1 = *	K_2 = *	K_3 = *
τ_1 = *	τ_2 = *	τ_3 ≤ *
τ_4 = *	τ_5 = *	τ_6 ≤ *

$f_1(\Delta I)$ =	*{* + (q_t − q_b)}	when $q_t - q_b$ < * RTP
	0% of RTP	when * RTP ≤ $q_t - q_b$ ≤ * RTP
	*{(q_t − q_b) − *}	when $q_t - q_b$ > * RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

* As specified in the COLR.

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T" is the nominal T_{avg} at RTP, ≤ *.

K_4 = *

K_5 = * for increasing T_{avg}
* for decreasing T_{avg}

K_6 = * when T > T"
* when T ≤ T"

τ_1 = *
τ_2 = *
τ_3 ≤ *
τ_6 ≤ *
τ_7 = *

$f_2(\Delta I)$ = *

* As specified in the COLR.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)";
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration";

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluations Methodology," July 1985.

2. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

11. WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

12. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met; and

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates, and Power Operated Relief Valve (PORV) lift settings shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System";

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report"; and

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Reports

a. Following each inservice inspection of SG tubes, the number of tubes plugged or repaired in each SG shall be reported to the NRC within 15 days.

b. The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

   1. Number and extent of tubes inspected,

   2. Location and percent of wall thickness penetration for each indication of an imperfection, and

   3. Identification of tubes plugged or repaired.

c. Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC within 30 days and prior to resumption of unit operation. The report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-72

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

   A. The application for amendment by the licensee dated February 15, 2000, and supplemented by letter dated July 26, 2000, 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-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 110 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 13, 2001

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-77

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

   A. The application for amendment by the licensee dated February 15, 2000, and supplemented by letter dated July 26, 2000, 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-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 110 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 13, 2001

ATTACHMENT TO LICENSE AMENDMENT NOS. 110 AND 110

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

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

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3.1.4-2	3.1.4-2
3.1.4-3	3.1.4-3
3.1.4-4	3.1.4-4
3.1.7-1	3.1.7-1
3.2.1-1	3.2.1-1
3.2.1-2	3.2.1-2
3.2.1-3	3.2.1-3
3.2.1-4	3.2.1-4
3.2.1-5	3.2.1-5
---------	3.2.1-6
3.2.2-1	3.2.2-1
3.2.2-2	3.2.2-2
3.2.2-3	3.2.2-3
3.2.3-1	3.2.3-1
3.2.3-2	---------
3.2.3-3	---------
3.2.3-4	---------
3.2.4-1	3.2.4-1
3.2.4-3	3.2.4-3
3.2.4-4	3.2.4-4
---------	3.2.5-1
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-4	3.3.1-4
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.3.1-7	3.3.1-7
3.3.1-8	3.3.1-8
3.3.1-9	3.3.1-9
3.3.1-10	3.3.1-10
3.3.1-11	3.3.1-11
3.3.1-12	3.3.1-12
3.3.1-13	3.3.1-13
3.3.1-14	3.3.1-14
3.3.1-15	3.3.1-15
3.3.1-16	3.3.1-16

Remove Pages	Insert Pages
3.3.1-17	3.3.1-17
3.3.1-18	3.3.1-18
3.3.1-19	-----------
5.6-3	5.6-3
5.6-4	5.6-4
-------	5.6-5
-------	5.6.6

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One rod not within alignment limit.	B.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours from discovery of Condition B concurrent with inoperability of Power Distribution Monitoring System (PDMS)
	<u>AND</u>		
	B.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>		
	B.4	Determine Heat Flux Hot Channel Factor ($F_Q(Z)$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$).	72 hours
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.5	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. More than one rod not within alignment limit.	C.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	C.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>		
	C.2	Be in MODE 3.	6 hours from discovery of Condition C concurrent with inoperability of PDMS
	<u>AND</u>		
	C.3	--------NOTE--------- Only required to be performed when PDMS is OPERABLE. --------------------- Restore rod(s) to within alignment limit.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B or Required Action C.3 not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 550°F; and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable DRPI and each demand position indicator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable DRPIs.	Once per 8 hours
	<u>OR</u>		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Initiate action to verify the position of the rods with inoperable DRPIs.	Immediately
	<u>OR</u>		
	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q^C(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes
	<u>AND</u>		
	A.2	Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	<u>AND</u>		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
B. $F_Q^W(Z)$ not within limit.	B.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	-------------------NOTES------------------- 1. During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained. 2. Not required to be performed until 12 hours after declaring Power Distribution Monitoring System (PDMS) inoperable. Performance of SR 3.2.1.3 satisfies the initial performance of this SR after declaring PDMS inoperable. --	
	Verify $F_Q^C(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified <u>AND</u> 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-----NOTES----- 1. During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained. 2. If $F_Q^W(Z)$ measurements indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits specified in the COLR; or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has not increased. -----	
		(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued) -----NOTES----- 3. Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.4 satisfies the initial performance of this SR after declaring PDMS inoperable. ---------- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	------NOTE------ Only required to be performed when PDMS is OPERABLE. ---------------- Verify $F_Q^C(Z)$ is within limit specified in the COLR.	7 days
SR 3.2.1.4	------NOTE------ Only required to be performed when PDMS is OPERABLE. ---------------- Verify $F_Q^W(Z)$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 and A.4 must be completed whenever Condition A is entered. --------------------- $F^N_{\Delta H}$ not within limit.	A.1 Reduce THERMAL POWER to < 50% RTP. AND A.2 Determine $F^N_{\Delta H}$. AND A.3 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP. AND	4 hours 24 hours 72 hours
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 ---------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. --------------------- Determine $F^N_{\Delta H}$.	Prior to exceeding 50% RTP AND Prior to exceeding 75% RTP AND 24 hours after reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-------------------NOTE------------------- Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable. -- Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Prior to exceeding 75% RTP after each refueling <u>AND</u> 31 Effective Full Power Days thereafter
SR 3.2.2.2	-------------------NOTE------------------- Only required to be performed when PDMS is OPERABLE. -- Verify $F^N_{\Delta H}$ is within limit specified in the COLR.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD shall be maintained within the limits specified in the COLR.

-----NOTE-----

The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is inoperable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	-----NOTE----- Not required to be performed until 12 hours after declaring PDMS inoperable. ---------- Verify AFD is within limits for each OPERABLE excore channel.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP when Power Distribution Monitoring System (PDMS) is inoperable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	<u>AND</u>		
	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>		
	A.3	Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	--------NOTE-------- Perform Required Action A.6 only after Required Action A.5 is completed. -------------------- Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after exceeding the THERMAL POWER limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------NOTES------ 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channel inputs can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. 3. Not required to be performed until 12 hours after declaring PDMS inoperable. ------------------ Verify QPTR is ≤ 1.02 by calculation.	7 days
SR 3.2.4.2	------NOTES------ 1. Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel is inoperable with THERMAL POWER > 75% RTP. 2. Not required to be performed until 12 hours after declaring PDMS inoperable. ------------------ Verify QPTR is ≤ 1.02 using the movable incore detectors.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Departure from Nucleate Boiling Ratio (DNBR)

LCO 3.2.5 DNBR shall be within the limit specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP when Power Distribution Monitoring System (PDMS) is OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DNBR not within limit.	A.1 Restore DNBR to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify DNBR is within limit specified in the COLR.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	------------NOTE------------ While this LCO is not met for Function 18, 19, or 20 in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. -----------------------------	
	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2.1 Initiate action to fully insert all rods.	48 hours
	<u>AND</u>	
	C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
D. One Power Range Neutron Flux-High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. -----------------------------	
	D.1 Place channel in trip.	6 hours
	<u>OR</u>	
	D.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ------------------------------	
	E.1 Place channel in trip.	6 hours
	OR	
	E.2 Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	OR	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1 Suspend operations involving positive reactivity additions.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1	Open Reactor Trip Breakers (RTBs).	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	J.2.1	Initiate action to fully insert all rods.	48 hours
	<u>AND</u>		
	J.2.2	Place the Rod Control System in a condition incapable of rod withdrawal.	49 hours
K. One channel inoperable.	-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.		
	K.1	Place channel in trip.	6 hours
	<u>OR</u>		
	K.2	Reduce THERMAL POWER to < P-7.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One Turbine Trip channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	L.1 Place channel in trip.	6 hours
	<u>OR</u>	
	L.2 Reduce THERMAL POWER to < P-8.	12 hours
M. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------	
	M.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	
	M.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One RTB train inoperable.	----------NOTES---------- 1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. 2. One RTB may be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. --------------------------------	
	N.1 Restore train to OPERABLE status.	1 hour
	OR	
	N.2 Be in MODE 3.	7 hours
O. One or more channels inoperable.	O.1 Verify interlock is in required state for existing unit conditions.	1 hour
	OR	
	O.2 Be in MODE 3.	7 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
P. One or more channels inoperable.	P.1	Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>		
	P.2	Be in MODE 2.	7 hours
Q. One trip mechanism inoperable for one RTB.	Q.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	<u>OR</u>		
	Q.2	Be in MODE 3.	54 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------NOTES------------------ 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours
SR 3.3.1.3	-------------------NOTES------------------ 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Only required to be performed with THERMAL POWER > 15% RTP. -- Compare results of the incore measurements to NIS AFD.	Prior to exceeding 75% RTP after each refueling AND 31 Effective Full Power Days (EFPD) thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	-------------------NOTE------------------- This Surveillance must be performed on the RTBB prior to placing the bypass breaker in service. -- Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.6	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 75% RTP. -- Calibrate excore channels to agree with incore measurements.	92 EFPD
SR 3.3.1.7	-------------------NOTE------------------- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. -- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-------------------NOTE------------------- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	-----NOTE----- Only required when not performed within previous 92 days -------------- Prior to reactor startup AND Four hours after reducing power below P-10 for power and intermediate instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 92 days thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.9	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	92 days
SR 3.3.1.10	-------------------NOTE------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	-----NOTE----- Verification of setpoint is not required. ---------- Perform TADOT.	-----NOTE----- Only required when not performed within previous 31 days ---------- Prior to reactor startup
SR 3.3.1.15	-----NOTE----- Neutron detectors are excluded from response time testing. ---------- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 1 (Page 3.3.1-18)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3.1-19)
8. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1875 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2393 psig
9. Pressurizer Water Level - High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.5% of instrument span
10. Reactor Coolant Flow - Low (per loop)	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.3% of loop minimum measured flow
11. Reactor Coolant Pump (RCP) Breaker Position (per train)	1(e)	4	K	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Undervoltage RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4920 V
13. Underfrequency RCPs (per train)	1(e)	4	K	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.08 Hz
14. Steam Generator (SG) Water Level - Low Low (per SG)					
a. Unit 1	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.1% of narrow range instrument span
b. Unit 2	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 34.8% of narrow range instrument span
15. Turbine Trip					
a. Emergency Trip Header Pressure (per train)	1(f)	3	L	SR 3.3.1.10 SR 3.3.1.14	≥ 910 psig
b. Turbine Throttle Valve Closure (per train)	1(f)	4	L	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	M	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Source Range Block Permissive, P-6	2[d]	2	O	SR 3.3.1.11 SR 3.3.1.12	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7					
(1) P-10 Input	1	3	P	SR 3.3.1.11 SR 3.3.1.12	NA
(2) P-13 Input	1	2	P	SR 3.3.1.10 SR 3.3.1.12	NA
c. Power Range Neutron Flux, P-8	1	3	P	SR 3.3.1.11 SR 3.3.1.12	≤ 32.1% RTP
d. Power Range Neutron Flux, P-10	1,2	3	O	SR 3.3.1.11 SR 3.3.1.12	≥ 7.9% RTP and ≤ 12.1% RTP
e. Turbine Impulse Pressure, P-13	1	2	P	SR 3.3.1.10 SR 3.3.1.12	≤ 12.1% turbine power
18. Reactor Trip Breakers (RTBs)[g]	1,2	2 trains	N	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	Q	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	1 each per RTB	C	SR 3.3.1.4	NA
20. Automatic Trip Logic	1,2	2 trains	M	SR 3.3.1.5	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(d) Below the P-6 (Source Range Block Permissive) interlock.

(g) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.04% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \left[T \frac{1}{(1+\tau_6 s)} - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, ≤ *.

P is the measured pressurizer pressure, psig.
P' is the nominal RCS operating pressure, = *.

K_1 = *	K_2 = *	K_3 = *
τ_1 = *	τ_2 = *	τ_3 ≤ *
τ_4 = *	τ_5 = *	τ_6 ≤ *

$f_1(\Delta I)$ =	*{* + (q_t - q_b)}	when $q_t - q_b$ < * RTP
	0% of RTP	when * RTP ≤ $q_t - q_b$ ≤ * RTP
	*{(q_t - q_b) - *}	when $q_t - q_b$ > * RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

* As specified in the COLR.

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.60% of ΔT span.

$$\Delta T \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[\frac{1}{1+\tau_3 s}\right] \leq \Delta T_0 \left\{K_4 - K_5 \frac{\tau_7 s}{1+\tau_7 s} \left[\frac{1}{1+\tau_6 s}\right] T - K_6 \left[T \frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T″ is the nominal T_{avg} at RTP, ≤ *.

K_4 = *

K_5 = * for increasing T_{avg}
* for decreasing T_{avg}

K_6 = * when T > T"
* when T ≤ T"

τ_1 = *
τ_2 = *
τ_3 ≤ *
τ_6 ≤ *
τ_7 = *

$f_2(\Delta I)$ = *

* As specified in the COLR.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.2.5, "Departure from Nucleate Boiling Ratio (DNBR)";
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration";

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluations Methodology," July 1985.

2. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

11. WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

12. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met; and

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates, and Power Operated Relief Valve (PORV) lift settings shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System";

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in NRC letter dated January 21, 1998, "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report"; and

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Post Accident Monitoring Report

When a report is required by Condition C or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Reports

a. Following each inservice inspection of SG tubes, the number of tubes plugged or repaired in each SG shall be reported to the NRC within 15 days.

b. The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

    1. Number and extent of tubes inspected,

    2. Location and percent of wall thickness penetration for each indication of an imperfection, and

    3. Identification of tubes plugged or repaired.

c. Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC within 30 days and prior to resumption of unit operation. The report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY

BYRON STATION, UNITS 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

In a letter of February 15, 2000, the licensee, Commonwealth Edison Company (ComEd) requested an amendment to the Byron and Braidwood Station operating licenses which would revise the applicable technical specification (TS) requirements to implement the Best Estimate Analyzer for Core Operations Nuclear (BEACON) power distribution monitoring system (PDMS), using the topical report WCAP-12472-P-A (Ref. 1) as the basis for this amendment. The Topical Report WCAP-12472-P-A, "BEACON: Core Monitoring and Operations Support System," was approved by the Nuclear Regulatory Commission (NRC) on February 16, 1994, (Ref. 2). This submittal addresses the implementation of BEACON system and the related technical specification changes at the Byron and Braidwood Stations.

The proposed change would allow the use of BEACON and PDMS, which have been developed by Westinghouse to improve the operational support for pressurized water reactors (PWRs). BEACON is an advanced core monitoring and support system that uses current instrumentation in conjunction with a fully analytical methodology for on-line generation of three-dimensional (3-D) core power distributions. The system provides core monitoring, core measurements reduction, core analysis, and core predictions.

As part of the implementation of BEACON, the licensee also intends to utilize the NRC approved Relaxed Axial Offset Control (RAOC) methodology (Ref. 4), for determining the axial flux difference (AFD). The RAOC methodology was implemented by Byron and Braidwood Nuclear Stations when they changed from a Constant Axial Offset Control (CAOC) to RAOC as one of the options provided in the approved BEACON Topical Report, WCAP-12472-P-A. On July 3, 2000, the staff issued a request for additional information. The licensee provided its response by letter dated July 26, 2000. The July 26, 2000, letter provided clarifying information that did not change the scope of the February 15, 2000, application or the proposed no significant hazards consideration determination.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to state the TSs to be included as part of the license. The Commission's regulatory requirements related to the content of the TSs are set forth in 10 CFR 50.36. That regulation requires the TSs to include items in five specific categories, including: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

Under 10 CFR 50.36(c)(2)(ii), a limiting condition for operation must be included in TSs for any item meeting one of the following four criteria:

1. installed instrumentation that is used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary;

2. a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

3. a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and

4. a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As a result, existing TS requirements that fall within or satisfy any of the criteria in 10 CFR 50.36 must be retained in the TSs, while those TS requirements that do not fall within or satisfy these criteria may be relocated to other licensee controlled documents. While the BEACON WCAP-12472-P-A proposed TS include the BEACON system per se, as implemented at Byron and Braidwood, the PDMS itself does not meet any of the 10 CFR 50.36(c)(2)(ii) selection criteria for inclusion into the Technical Specifications (TS), since key parameters are separately covered. Therefore, there is not a proposed "PDMS" TS.

3.0 EVALUATION

In the present PWR core monitoring methodology, there is no direct margin assessment on a continuous basis. Current fuel cycles contain design margins to assure safe core operation under steady state and transient conditions due to the operator's inability to continuously monitor the core power distribution. The proposed changes allow the licensee to use the Westinghouse core monitoring and support system known as BEACON PDMS to improve the operational support for PWRs. The PDMS maintains an on-line 3-D nodal model that is continuously updated to reflect the current plant operational conditions. The Core Exit Thermocouples (CETCs) and excore neutron flux detectors are used with the reference 3-D power distribution to determine the measured power distribution. By coupling the measured 3-D power distribution with an on-line evaluation, actual core margins are better understood. The BEACON methodology would allow for changes in the core design methods and provide for more optimized core loading patterns. The system provides core monitoring, core measurement reduction, core analysis, and core predictions. The BEACON methodology also improves the quality of the surveillance process since it uses a depleted model to match the actual operational profile.

The key aspects of the BEACON system are: (1) the methodology used to obtain the measured power distribution from the Westinghouse standard instrumentation system, that is, movable incore detectors, core exit thermocouples and excore detectors, and (2) the methodology for assessing uncertainties to be applied to the measured power distribution and TSs with BEACON as the source of the measured power distribution. BEACON utilizes current incore instrumentation to generate 3-D core power distributions in conjunction with the computer code SPNOVA. SPNOVA generates detailed power distribution information and comparisons to core limits on a continuous basis and the PDMS makes this information available to the operators. SPNOVA uses a two-group diffusion equation to provide data reduction of incore flux maps, core parameters, (such as fuel burnup, xenon build-up, etc.) analysis and follow and core predictions (e.g., critical conditions and startup).

BEACON provides a greatly improved continuous on-line power distribution measurement and display, limit surveillance, and operation prediction information system. No new instrumentation or calculation system other than the interface systems and integration analysis is introduced. The licensee reviewed the NRC-approved Topical Report WCAP-12472(P) to make sure that the assumptions and conditions (power peaking uncertainties and assumptions uncertainties made in the BEACON analysis), imposed as conditions for approval of WCAP-12472(P), remain valid for all the core monitoring and operations support functions that will be performed by the BEACON system for Byron and Braidwood Stations. Results of the analysis conducted by the licensee indicates that all the assumptions and conditions imposed on the approved Topical Report WCAP-12472(P) are still satisfied. The staff has reviewed the licensee's analysis, agrees with it, and finds it acceptable.

The licensee stated that implementation of the PDMS at the Byron and Braidwood Stations does not replace, eliminate, or modify existing plant instrumentation. The PDMS software runs on a workstation connected to the plant process computer. The PDMS combines input from currently installed plant instrumentation and design data generated at each fuel cycle. Together, this provides a means to monitor power distribution limits continuously and to alert the operator when limits are being approached. The staff finds that the proposed changes allow for the power distribution surveillances to be performed by PDMS rather than using the movable incore detector system and the PDMS does not provide any direct protection or control function. The implementation of the PDMS at the Byron and Braidwood Stations does not change the safety-related instrumentation configuration. It will be treated as a non-safety-related system. Any program running at the process computer has no direct input to the safety-related protection system. The PDMS provides information to alert the operator, and there is no automatic action involved. The staff finds that the implementation of BEACON PDMS does not affect the existing safety-related instrument, and therefore, is acceptable.

3.1 Transient Analysis

The transient behavior of the Byron and Braidwood Stations were reexamined with respect to changes in cycle-dependent parameters, in accordance with NRC-approved reload design methodology, and the requirements of 10 CFR 50.59, "Changes, tests and experiments."

As part of the implementation of the BEACON system, the licensee is utilizing the RAOC methodology for determining the axial flux difference. The methodology was described in topical report WCAP-10216-P-A, Revision 1 (Ref. 3), and approved by the NRC staff on November 26, 1993, (Ref. 4). In Amendments 106 and 98, dated December 22, 1998, NRC staff approved conversion to the Improved Standard Technical Specifications (ISTS), which incorporated the conversion from the total peaking factor ($F_{xy}(z)$) methodology to the heat flux hot channel factor times the transient function ($F_Q(Z)W(Z)$) methodology for Byron and Braidwood Stations.

As part of the implementation of the BEACON system, the Byron and Braidwood Stations are converting from the CAOC methodology to the RAOC methodology. The RAOC methodology was developed and NRC staff approved to accommodate a wider range of constraints on the axial power distribution controls. This methodology allows a wider axial flux difference (Δ I) versus power operating space relative to the CAOC operation, particularly at reduced power levels, while ensuring that reactor safety considerations are satisfied.

The effects of the wider (Δ I) band on the consequences of anticipated transients were discussed in the above stated Topical Report (Ref. 3). The results were examined for violations of peak power density and departure form nucleate boiling (DNB) limits.

The implementation of PDMS and RAOC will not result in a reduction in the safety margin. Core reloads will continue to be conducted in accordance with 10 CFR 50.59 and other established safety analysis acceptance limits and all the required safety margins will be maintained. Consequently, the Byron and Braidwood Station's Updated Final Safety Analysis Report (UFSAR) transients and subsequent reload specific analyses and evaluations will be performed in accordance with the NRC-approved methodologies. Thus, the margins of safety will be maintained as defined in the bases of the TS and the UFSAR for Byron and Braidwood Stations. The staff finds the results to be acceptable.

3.2 Technical Specification Changes

The Byron and Braidwood Stations converted to the "Improved Standard Technical Specifications (ISTS)," using NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 1, in February 1999, as its basis. Therefore, the TS markups contained in WCAP-12472-P-A, which are based on NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Draft Revision 5, are not directly applicable to the Byron and Braidwood Stations. Consequently, many changes were proposed by the licensee to the Byron/Braidwood TS to accommodate the ISTS.

The staff reviewed the contents of the affected TS contained in the submittal for Byron and Braidwood Stations. Where they differed from those specified in WCAP-12472-P-A, the licensee provided the technical basis to support the proposed change, including changes to the Core Operating Limits Report (COLR) . In addition, the licensee implemented BEACON in accordance with the conditions stated in WCAP-12472-P-A.

The licensee proposes changes to the following TSs for Byron and Braidwood Stations:

TS 3.1.4, Rod Group Alignment Limits;
TS 3.1.7, Rod Position Indication;
TS 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);
TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$);
TS 3.2.3, Axial Flux Difference (AFD);
TS 3.2.4, Quadrant Power Tilt Ratio (QPTR);
TS 3.3.1, Reactor Trip System (RTS) Instrumentation; and
TS 5.6.5, Core Operating limits Report (COLR).

The licensee proposed to add the following TS to the Byron and Braidwood Station TS:

TS 3.2.5, Departure From Nucleate Boiling Ratio (DNBR).

The licensee included associated TS Bases proposed changes for information for the following Byron and Braidwood Station TS Bases:

TS B 3.1.4, Rod Group Alignment Limits;
TS B 3.1.7, Rod Position Indication;
TS B 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$);
TS B 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$);
TS B 3.2.3, Axial Flux Difference (AFD);
TS B 3.2.4, Quadrant Power Tilt Ratio (QPTR);
TS B 3.2.5, Departure from Nucleate Boiling Ratio (DNBR); and
TS B 3.3.1, Reactor Trip System (RTS) Instrumentation.

The licensee also included associated Technical Requirements Manual (TRM) proposed changes for information for the following Byron and Braidwood Station TRM:

TRM 3.3.a, Movable Incore Detector; and
TRM 3.3.h, Power Distribution Monitoring System (PDMS) Instrumentation.

Finally, the licensee included proposed changes for the Core Operating Limit Report (COLR) for information.

3.3 Evaluation of Technical Specification Changes:

1) The PDMS does not need a TS requiring its operability because the PDMS itself does not meet any of the 10 CFR 50.36(c)(2)(ii) selection criteria for inclusion into the TS, since key parameters are covered separately. The PDMS is the Byron and Braidwood plant-specific nomenclature for the BEACON monitoring system presented in WCAP-12472. The BEACON parameter TS in WCAP-12472-P-A for peak linear heat rate (i.e., $F_Q(Z)$), $F^N_{\Delta H}$, and Departure from Nucleate Boiling Ratio (DNBR), are included in the proposed Byron and Braidwood TS.

These TSs have been modified to reflect the necessary changes to adopt the BEACON system and adequately address the parameters and requirements of WCAP-12472-P-A, and are acceptable.

2) Unlike the BEACON instrumentation functions, the proposed PDMS TS instrumentation functions do not include a pressurizer pressure requirement. PDMS uses the pressurizer pressure, among other inputs, to calculate DNBR, however, PDMS operability is not dependent upon the status of the pressurizer pressure input. During normal operation, the Revised Thermal Design Procedure (RTDP) analysis provides for any pressurizer pressure uncertainties and variations from the normal operating band. Accordingly, the staff finds the proposed TS acceptable.

3) The licensee proposed a change to TS 3.3.1, Reactor Trip System (RTS) Instrumentation. The proposed change will delete Actions D.1.2, D.2.1, and D.2.2, under a condition that one Power Range Neutron Flux-High channel becomes inoperable.

The RTS function is to maintain the Safety Limits during all anticipated operational occurrences (AOO) and mitigate the consequences of design-basis accidents in all modes in which the control rods are not fully inserted. The TS 3.3.1 LCO requires all instrumentation performing a RTS function to be operable when the unit status is within the TS applicability. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected functions.

The current TS LCO 3.3.1 Condition D, which involves One Power Range Neutron Flux-High channel inoperable, has REQUIRED ACTION as follows:

	D.1.1 Place channel in trip	Within 6 hours	AND
	D.1.2 Reduce THERMAL POWER to $\leq$75% RTP	within 12 hours	
OR	D.2.1 Place channel in trip	Within 6 hours	AND
	D.2.2 Perform SR 3.2.4.2	Once per 12 hours	

(Note: Only required to be performed when the power range neutron flux input to quadrant power tilt ratio (QPTR) is not operable)

SR 3.2.4.2 is the surveillance requirement to verify that QPTR is $\leq$1.02 using the movable incore detectors.

The licensee proposed to delete the required actions D.1.2, D.2.1, and D.2.2, and re-number D.1.1 to D.1, and D.3 to D.2. The justification for these changes is that TS LCO 3.2.4, "The QPTR shall be $\leq$ 1.02", Surveillance Requirement (SR) 3.2.4.1 and SR 3.2.4.2 have adequately addressed the similar requirements specified in LCO 3.3.1 Condition D. With the input to QPTR from one Power Range Neutron Flux Channel inoperable with Thermal Power > 75 percent rated thermal power (RTP), SR 3.2.4.2 verifies QPTR $\leq$ 1.02 using the movable incore detectors, thereby compensating for the potential lost monitoring capability due to the inoperable Power Range Neutron Flux Channel and allow continued operation at power level > 75 percent RTP.

The staff has verified that the surveillance requirements under LCO 3.2.4 have addressed the required actions D.1.2, D.2.1 and D.2.2 listed above. Deletion of required actions D.1.2, D.2.1, and D.2.2 does not affect the reactor trip system functional requirements. Therefore, the licensee's proposed changes related to LCO 3.3.1 are acceptable.

4) The adoption of the PDMS necessitates, in some specifications, two sets of surveillance requirements, one set for when PDMS is operable and another set for when PDMS is inoperable. TS 3.1.4 Required Action B.4 to perform SR 3.2.1.1, to determine the Heat Flux Hot Channel Factor, and TS 3.1.4 Required Action B.5 to perform SR 3.2.2.1, to determine the Nuclear Enthalpy Rise Hot Channel Factor, are combined into one action that states the hot channel factors are to be determined, rather than list by number the specific surveillance requirements that are to be performed. Similarly, TS 3.2.2 Required Actions A.2 and A.4 now state that the Nuclear Enthalpy Rise Hot Channel Factor is to be determined, rather than refer to a specific surveillance requirement number to be performed (i.e., "to perform SR 3.2.2.1"). When PDMS is operable, it will continuously fulfill surveillance requirements by performing one set of new SRs (SR 3.2.1.3 and SR 3.2.2.2). Another set of surveillance requirements will need to be performed only when PDMS is inoperable (SR 3.2.1.1 and SR 3.2.2.1). Conveying that selection logic in the TS Required Actions statements would be involved, and it is therefore preferable to utilize a brief functional description of the SR to be performed rather than specify the exact SR number to be performed. This is acceptable. It is recommended that the Bases be updated to explain the SR selection options explicitly, by SR number, to minimize the possibility of confusion over what is required.

5) Certain Required Actions (RAs) in the TSs which restore the Limiting Conditions for Operation are no longer necessary and have been deleted. In TS 3.2.1, RAs A.4 and B.4, which require the performance of surveillances prior to exceeding a power level, are deleted. The deleted RAs had the licensee perform surveillances to confirm that LCO limits are met prior to increasing power; and that the Condition statement no longer applies. It is not necessary to explicitly state that operability or conditions are restored; it is an implicit requirement. These RAs unnecessarily clutter the TS, and therefore deleting them is acceptable.

6) A note has been proposed to be added to SR 3.2.1.1 (and SR 3.2.1.2) stating that the SR is "Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.1.3 (SR 3.2.1.4) satisfies the initial performance of this SR after declaring PDMS inoperable." Similarly, a note has been proposed to be added to SR 3.2.2.1 stating that the SR is "Not required to be performed until 12 hours after declaring PDMS inoperable. Performance of SR 3.2.2.2 satisfies the initial performance of this SR after declaring PDMS inoperable." Without the addition of these notes it can be interpreted that SR 3.0.4 would be violated; that is, "Entry into a Mode specified in the Applicability shall not be made unless the TS Surveillances have been met within their specified frequency," would not be satisfied. It is logical to take credit for the PDMS performance of the surveillances prior to its inoperability, since the surveillances to be performed when the PDMS is inoperable verify the same conditions. These changes are consistent with the STS SR 3.0.4 requirements and are therefore acceptable.

4.0 SUMMARY

The staff has reviewed the submittal by ComEd (Ref. 1) for implementing the BEACON methodology and its impact on TS to Byron and Braidwood Stations, and finds it acceptable. The proposed Byron and Braidwood TS changes meet the technical requirements of the Topical Report WCAP-12472-P-A, "BEACON: Core Monitoring and Support System." Also the implementation of the BEACON monitoring system at the Byron and Braidwood Stations is in accordance with the conditions imposed in the BEACON Safety Evaluation Report. The staff has concluded that: (1) implementation of BEACON does not affect the existing safety-related instruments; (2) implementation of the PDMS and RAOC will not result in a reduction in the safety margin; and (3) the revised TSs adopting the BEACON PDMS system are acceptable. In addition, applicable UFSAR limits will be maintained and the appropriate TS will continue to require operations within the core operational limits calculated by NRC approved methodologies.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or 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 offsite, 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 (65 FR 17909). 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.

7.0 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.

8.0 REFERENCES

1) WCAP-12472-P-A, "BEACON - Core Monitoring and Operations Support System," August 1994.

2) Letter from A.C.Thadani, (NRC) to N.J. Liparulo (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-12472-P, "BEACON - Core Monitoring and Operations Support System," February 16, 1994.

3) WCAP-10216-P-A, Revision 1, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.

4) Letter from A.C.Thadani, (NRC) to N.J. Liparulo (Westinghouse), "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P, Rev. 1, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," November 26, 1993.

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