June 18, 1996

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Mr. Donald C. Shelton
Acting Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M95262)

Dear Mr. Shelton:

The Commission has issued the enclosed Amendment No. 85 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Improved Technical Specifications in response to your application dated April 26, 1996.

This amendment corrects minor technical and administrative errors in the Improved Technical Specifications.

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

Sincerely,
Original signed by:

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 85 to License No. NPF-58
2. Safety Evaluation

cc w/encls: See next page

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

June 18, 1996

Mr. Donald C. Shelton
Acting Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M95262)

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Sincerely,

Jon B. Hopkins

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 85 to License No. NPF-58
2. Safety Evaluation

cc w/encls: See next page

Mr. Donald C. Shelton
Centerior Service Company

Perry Nuclear Power Plant
Unit Nos. 1 and 2

cc:

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The Honorable Robert V. Orosz
Mayor, Village of North Perry
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. NPF-58

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

   A. The application for amendment by the Cleveland Electric Illuminating Company (CEICO), Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated April 26, 1996, 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-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 85 are hereby incorporated into this license. The Cleveland Electric Illuminating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

Jon B. Hopkins, Senior Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: June 18, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 85

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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

Remove	Insert
ITS 3.1-20	ITS 3.1-20
ITS 3.3-19	ITS 3.3-19
ITS 3.3-23	ITS 3.3-23
ITS 3.3-41	ITS 3.3-41
ITS 3.3-54	ITS 3.3-54
ITS 3.3-55	ITS 3.3-55
ITS 3.3-56	ITS 3.3-56
ITS 3.3-59	ITS 3.3-59
ITS 3.4-1	ITS 3.4-1
ITS 3.4-25	ITS 3.4-25
ITS 3.4-30	ITS 3.4-30
ITS 3.6-6	ITS 3.6-6
ITS 3.6-19	ITS 3.6-19
ITS 3.6-35	ITS 3.6-35
ITS 3.6-48	ITS 3.6-48
ITS 3.6-65	ITS 3.6-65
ITS 3.6-66	ITS 3.6-66
ITS 3.6-67	ITS 3.6-67
ITS 3.7-4	ITS 3.7-4
ITS 3.7-8	ITS 3.7-8
ITS 3.7-18	ITS 3.7-18
ITS 3.8-6	ITS 3.8-6
ITS 3.8-8	ITS 3.8-8
ITS 3.8-10	ITS 3.8-10
ITS 3.8-15	ITS 3.8-15
ITS 3.8-28	ITS 3.8-28
ITS 3.8-29	ITS 3.8-29
ITS 3.8-30	ITS 3.8-30
ITS 3.8-31	ITS 3.8-31
ITS 4.0-2	ITS 4.0-2
ITS 5.0-1	ITS 5.0-1
ITS 5.0-4	ITS 5.0-4
ITS 5.0-9	ITS 5.0-9
ITS 5.0-12	ITS 5.0-12
ITS 5.0-20	ITS 5.0-20

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Rod Pattern Control System			
a. Rod withdrawal limiter	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.6 SR 3.3.2.1.9
	(b)	2	SR 3.3.2.1.2 SR 3.3.2.1.5 SR 3.3.2.1.7 SR 3.3.2.1.9
b. Rod pattern controller	1(c), 2	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.5 SR 3.3.2.1.7 SR 3.3.2.1.9
2. Reactor Mode Switch – Shutdown Position	(d)	2	SR 3.3.2.1.8

(a) THERMAL POWER > 70% RTP.

(b) THERMAL POWER > 35% RTP and ≤ 70% RTP.

(c) With THERMAL POWER ≤ 20% RTP.

(d) Reactor mode switch in the shutdown position.

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Steam Dome Pressure	2	E
2. Reactor Vessel Water Level—Wide Range	2	E
3. Reactor Vessel Water Level—Fuel Zone	2	E
4. Suppression Pool Water Level	2	E
5. Suppression Pool Sector Water Temperature	2[c]	E
6. Drywell Pressure	2	E
7. Drywell Air Temperature	2	E
8. Primary Containment/Drywell Area Gross Gamma Radiation Monitors	2	F
9. Penetration Flow Path, PCIV Position	2 per penetration flow path[a][b]	E
10. Primary Containment and Drywell H_2 Concentration Analyzer and Monitor	2	E
11. Primary Containment Pressure	2	E
12. Primary Containment Air Temperature	2	E

(a) Not required for isolation valves whose associated penetration flow path is isolated.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) Monitoring each of eight sectors.

Table 3.3.5.1-1 (page 3 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI B and LPCI C Subsystems (continued)					
e. LPCI Pump B and LPCI Pump C Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}$,$5^{(a)}$	1 per pump	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 1450 gpm
f. Manual Initiation	1,2,3, $4^{(a)}$,$5^{(a)}$	1	C	SR 3.3.5.1.6	NA
3. High Pressure Core Spray (HPCS) System					
a. Reactor Vessel Water Level – Low Low, Level 2	1,2,3, $4^{(a)}$,$5^{(a)}$	$4^{(e)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 127.6 inches
b. Drywell Pressure – High	1,2,3	$4^{(e)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 1.88 psig
c. Reactor Vessel Water Level – High, Level 8	1,2,3, $4^{(a)}$,$5^{(a)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 221.7 inches
d. Condensate Storage Tank Level – Low	1,2,3, $4^{(c)}$,$5^{(c)}$	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 59,700 gallons
e. Suppression Pool Water Level – High	1,2,3	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 18 ft 6 inches

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(c) When HPCS is OPERABLE for compliance with LCO 3.5.2, "ECCS – Shutdown," and aligned to the condensate storage tank while tank water level is not within the limits of SR 3.5.2.2.

(e) Also required to initiate the associated diesel generator.

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 14.3 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 795.0 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 191 psid
d. Condenser Vacuum – Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 7.6 inches Hg vacuum
e. Main Steam Line Pipe Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F
f. Main Steam Line Turbine Building Temperature—High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 138.9°F
g. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2[(b)]	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches

(continued)

(a) With any turbine stop valve not closed.

(b) Required to initiate the associated drywell isolation function.

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level—Low Low, Level 2 (continued)	(c)	2(b)	L	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches
b. Drywell Pressure – High	1,2,3	2(b)	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
c. Reactor Vessel Water Level – Low Low Low, Level 1 (ECCS Divisions 1 and 2)	1,2,3	2(b)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 14.3 inches
	(c)	2(b)	L	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 14.3 inches
d. Drywell Pressure – High (ECCS Divisions 1 and 2)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
e. Reactor Vessel Water Level – Low Low, Level 2 (HPCS)	1,2,3	4	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches
	(c)	4	L	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches
f. Drywell Pressure – High (HPCS)	1,2,3	4	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
g. Containment and Drywell Purge Exhaust Plenum Radiation – High	1,2,3	2(b)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 4.0 mR/hr above background

(continued)

(b) Required to initiate the drywell isolation function.

(c) During CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation					
g. Containment and Drywell Purge Exhaust Plenum Radiation – High (continued)	(d)	2	K	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 4.0 mR/hr above background
h. Manual Initiation	1,2,3	2(b)	G	SR 3.3.6.1.5	NA
	(d)	2	K	SR 3.3.6.1.5	NA
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 298.5 inches water
b. RCIC Steam Line Flow Time Delay	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 3 seconds and ≤ 13 seconds
c. RCIC Steam Supply Line Pressure – Low	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 55 psig
d. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 20 psig
e. RCIC Equipment Area Ambient Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 145.9°F
f. Main Steam Line Pipe Tunnel Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F

(continued)

(b) Required to initiate the drywell isolation function.

(d) During CORE ALTERATIONS, operations with a potential for draining the reactor vessel, and movement of irradiated fuel assemblies in primary containment.

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RHR System Isolation					
a. RHR Equipment Area Ambient Temperature – High	$2^{(e)}, 3^{(e)}$	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 159.9°F
b. Reactor Vessel Water Level-Low, Level 3	$1, 2^{(g)}, 3^{(g)}$	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 177.1 inches
	$2^{(e)}, 3^{(e)}, 4, 5$	$2^{(f)}$	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 177.1 inches
c. Reactor Vessel Steam Dome Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.6.1.1.3 SR 3.6.1.1.4 SR 3.6.1.1.5	≤ 150 psig
d. Drywell Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
e. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA

(e) With reactor vessel steam dome pressure less than the RHR cut in permissive pressure.

(f) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

(g) With reactor vessel steam dome pressure greater than or equal to the RHR cut in permissive pressure.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Either:

a. Two recirculation loops shall be in operation with:

1. Matched flows; and
2. Total core flow and THERMAL POWER within limits.

OR

b. One recirculation loop shall be in operation with:

1. Thermal power ≤ 2500 MWt;
2. Total core flow and THERMAL POWER within limits;
3. Required limits modified for single recirculation loop operation as specified in the COLR; and
4. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power – High) Allowable Value of Table 3.3.1.1-1 reset for single loop operation.

------------------------NOTE------------------------
Required limit and setpoint modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Recirculation loop jet pump flow mismatch not within limits.	A.1	Shut down one of the recirculation loops.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND	
	B.2 Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.8	-------------------NOTE------------------- Only required to be met in single loop operation during increases in THERMAL POWER or recirculation loop flow with the operating recirculation loop jet pump flow ≤ 50% of rated core flow or THERMAL POWER ≤ 30% of RTP, and with reactor vessel steam dome pressure ≥ 25 psig. -- Verify the difference between the bottom head coolant temperature and the RPV coolant temperature is ≤ 100°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow
SR 3.4.11.9	-------------------NOTE------------------- Only required to be met in single loop operation during increases in THERMAL POWER or recirculation loop flow with the operating recirculation loop jet pump flow ≤ 50% of rated core flow, or THERMAL POWER ≤ 30% of RTP, and the idle recirculation loop not isolated from the RPV. -- Verify the difference between the reactor coolant temperature in the recirculation loop not in operation and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow
SR 3.4.11.10	The reactor vessel material surveillance specimens shall be removed and examined to determine changes in reactor pressure vessel material properties.	In accordance with the schedule required by 10 CFR 50, Appendix H

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours
E. Required Action and associated Completion Time of Condition A, B, or C not met during movement of recently irradiated fuel assemblies in the primary containment, or during OPDRVs.	E.1	Suspend movement of recently irradiated fuel assemblies in the primary containment.	Immediately
	AND		
	E.2	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify combined leakage rate of 1 gpm times the total number of PCIVs through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at $\geq 1.1\ P_a$.	-----NOTE----- SR 3.0.2 is not applicable -------------- - In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions
SR 3.6.1.3.12	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify each outboard 42 inch primary containment purge valve is blocked to restrict the valve from opening > 50°.	18 months
SR 3.6.1.3.13	------------------NOTE------------------ Not required to be met when the Backup Hydrogen Purge System isolation valves are open for pressure control, ALARA or air quality considerations for personnel entry, or Surveillances or special testing of the Backup Hydrogen Purge System that require the valves to be open. -- Verify each 2 inch Backup Hydrogen Purge System isolation valve is closed.	31 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met or in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the primary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1	Suspend movement of irradiated fuel assemblies in the primary containment.	Immediately
	<u>AND</u>		
	C.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENT

SURVEILLANCE		FREQUENCY
SR 3.6.1.12.1	Verify containment average temperature-to-relative humidity to be within limits.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.2.4	Verify each required igniter in accessible areas develops a surface temperature of ≥ 1700°F.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.5.3 Drywell Isolation Valves

LCO 3.6.5.3 Each drywell isolation valve, except for Drywell Vacuum Relief System valves, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths, except for the 24 inch and 36 inch purge supply and exhaust valve penetration flow path, may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by drywell isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.5.1, "Drywell," when drywell isolation valve leakage results in exceeding overall drywell bypass leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one drywell isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	8 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	 Prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days
B. One or more penetration flow paths with two drywell isolation valves inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours
C. Required Action and associated Completion Time not met.	C.1 <u>AND</u> C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.3.1	Verify each 24 inch and 36 inch drywell purge supply and exhaust isolation valve is sealed closed.	31 days
SR 3.6.5.3.2	Deleted.	
SR 3.6.5.3.3	-------------------NOTES------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for drywell isolation valves that are open under administrative controls. -- Verify each drywell isolation manual valve and blind flange that is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4, if not performed in the previous 92 days
SR 3.6.5.3.4	Verify the isolation time of each power operated and each automatic drywell isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.5.3.5	Verify each automatic drywell isolation valve actuates to the isolation position on an actual or simulated isolation signal.	18 months

3.7 PLANT SYSTEM

3.7.3 Control Room Emergency Recirculation (CRER) System

LCO 3.7.3 Two CRER subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the primary containment or fuel handling building,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRER subsystem inoperable.	A.1 Restore CRER subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.7 PLANT SYSTEMS

3.7.4 Control Room Heating, Ventilating, and Air Conditioning (HVAC) System

LCO 3.7.4 Two control room HVAC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the primary containment or fuel handling building,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room HVAC subsystem inoperable.	A.1 Restore control room HVAC subsystem to OPERABLE status.	30 days
B. Two control room HVAC subsystems inoperable.	B.1 Verify control room air temperature is $\leq 90°F$.	Once per 4 hours
	<u>AND</u>	
	B.2 Restore one control room HVAC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	C.2 Be in MODE 4.	36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each FHB ventilation exhaust subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.7.9.2	Perform FHB ventilation exhaust filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Perform a system functional test.	18 months
SR 3.7.9.4	Perform a CHANNEL FUNCTIONAL TEST of the FHB ventilation exhaust radiation monitor (noble gas)	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-------------------NOTES------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by, and immediately follow, without shutdown, a successful performance of SR 3.8.1.2 or SR 3.8.1.7. --- Verify each DG operates for ≥ 60 minutes at a load ≥ 5600 kW and ≤ 7000 kW for Division 1 and 2 DGs, and ≥ 2600 kW for Division 3 DG.	As specified in Table 3.8.1-1
SR 3.8.1.4	Verify each day tank contains ≥ 316 gal of fuel oil for Divisions 1 and 2 and ≥ 279 gal for Division 3.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to automatically transfer fuel oil from the storage tank to the day tank.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	-----------------NOTES----------------- 1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. --- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load. Following load rejection, engine speed is maintained less than nominal plus 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is less.	18 months
SR 3.8.1.10	-----------------NOTE----------------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. --- Verify each DG operating at a power factor ≤ 0.9 does not trip and voltage is maintained ≤ 4784 V for Division 1 and 2 DGs and ≤ 5000 V for Division 3 DG during and following a load rejection of a load ≥ 5600 kW for Division 1 and 2 DGs and ≥ 2600 kW for Division 3 DG.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------NOTES------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. -- Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each DG auto-starts from standby condition and: a. In ≤ 10 seconds for Divisions 1 and 2, and ≤ 13 seconds for Division 3 after auto-start and during tests, achieves voltage ≥ 3900 V and ≤ 4400 V; b. In ≤ 10 seconds for Divisions 1 and 2, and ≤ 13 seconds for Division 3 after auto-start and during tests, achieves frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and c. Operates for ≥ 5 minutes.	18 months
SR 3.8.1.13	------------------NOTE-------------------- This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify each DG's automatic trips are bypassed on an actual or simulated ECCS initiation signal except: a. Engine overspeed; and b. Generator differential current.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.20	------------------NOTE------------------ All DG starts may be preceded by an engine prelube period. -- Verify, when started simultaneously from standby condition, the Division 1 and 2 DGs achieve voltage ≥ 3900 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz in ≤ 10 seconds, and the Division 3 DG achieves a frequency ≥ 58.8 Hz in ≤ 10 seconds, and a voltage ≥ 3900 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz in ≤ 13 seconds .	10 years

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources – Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division 1 or 2 onsite Class 1E electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems - Shutdown";

b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem when required by LCO 3.8.8; and

c. The Division 3 DC electrical power subsystem capable of supplying the Division 3 onsite Class 1E DC electrical power distribution subsystem, when the Division 3 onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the primary containment or fuel handling building.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1	Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.2.2	Suspend movement of irradiated fuel assemblies in the primary containment and fuel handling building.	Immediately
	<u>AND</u>		
	A.2.3	Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>		
	A.2.4	Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	--------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs

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4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the USAR;

b. A nominal fuel assembly center to center storage spacing of 7 inches within rows and 12 inches between rows in the storage racks in the upper containment pool; and

c. A nominal fuel assembly center to center storage spacing of 6.625 inches, with a neutron poison material between storage spaces, in the high density storage racks in the fuel handling building.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the USAR; and

b. A nominal 7 inch center to center distance between fuel assemblies placed in storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 594 ft 6 inches.

4.3.3 Capacity

4.3.3.1 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4020 fuel assemblies.

4.3.3.2 No more than 190 fuel assemblies may be stored in the upper containment pool.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager, or his designee, shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety, and all administrative procedures.

5.1.2 The shift supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the unit is in MODE 1, 2, or 3, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 4 or 5, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions as modified by Specification 5.2.2.f, except for the radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the licensed Reactor Operators and Senior Reactor Operators, who shall comply with the requirements of 10 CFR 55.

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary as follows:

   1. for noble gases: ≤ 500 mrem/yr to the total body and ≤ 3000 mrem/yr to the skin, and

   2. for iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days: ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 3.9.1.1, cyclic and transient occurrences to ensure that the reactor vessel is maintained within the design limits.

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:

ESF Ventilation System	Delta P	Flowrate
a) Control Room Emergency Recirculation	4.9" H_2O	30,000 scfm
b) Fuel Handling Building	4.9" H_2O	15,000 scfm
c) Annulus Exhaust Gas Treatment	6.0" H_2O	2,000 scfm

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below ± 10% when corrected to nominal input voltage when tested in accordance with ANSI N510-1980:

ESF Ventilation System	Wattage
a) Control Room Emergency Recirculation	100 kW
b) Fuel Handling Building	50 kW
c) Annulus Exhaust Gas Treatment	20 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the main condenser offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the main condenser offgas treatment system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

(continued)

5.7 High Radiation Area

5.7.2 (continued)

Individuals qualified in radiation protection procedures (e.g., health physics technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 3000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

5.7.3 In addition to the requirements of Specification 5.7.1, for individual high radiation areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose ≥ 1000 mrem that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that are not continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

5.7.4 In addition to the requirements and exemptions of Specifications 5.7.1 and 5.7.2 for individual areas accessible to personnel such that a major portion of the body could receive in 1 hour a dose > 3000 mrem, entry shall require an approved RWP which will specify dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over activities within the areas.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. NPF-58

CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated April 26, 1996, the Cleveland Electric Illuminating Company et al. (the licensees), requested changes to the Improved Technical Specifications (ITS) for the Perry Nuclear Power Plant, Unit 1 (PNPP). The proposed amendment would correct minor technical and administrative errors in the ITS prior to its implementation expected in July 1996.

2.0 EVALUATION

A requested change would create a new footnote in Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation." The current footnote is incorrect as applied to the high pressure core spray (HPCS) system because it indicates that an interface exists between the annulus exhaust gas treatment (AEGT) system and the HPCS system. Therefore, the existing footnote is incorrect for two listed functions in the table. The NRC staff finds that the new footnote is acceptable because it does not reference the AEGT system. The existing footnote will continue to apply to appropriate system instrumentation that interfaces with the AEGT system.

Another requested change would remove the backup hydrogen purge system (BHPS) valves from the ITS section on drywell isolation valves and place the BHPS valves into the ITS section for primary containment isolation valves. This is correct because the BHPS valves are primary containment valves. The NRC staff finds this change acceptable.

Another requested change would increase the minimum volumes required for the diesel generator fuel oil day tank. The licensee performed new calculations to account for instrument uncertainties and to ensure that the diesel generator secondary fuel oil transfer pump does not reach its shutoff point. Increasing the required minimum tank volumes is conservative and appropriate; therefore, the NRC staff finds this change acceptable.

Also, a change is proposed for Specification 5.3.1, "Unit Staff Qualifications." By license Amendment No. 70, a change was made to the current Technical Specifications to reflect that reactor operators shall comply with the requirements of 10 CFR Part 55. This change was not included in ITS which were issued by license Amendment No. 69. The NRC staff has reviewed this change and finds it acceptable for inclusion in the ITS.

Finally, the remaining changes are typographical and administrative in nature. These changes were reviewed by the NRC staff, and were found acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 21213). Accordingly, this amendment meets 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 this amendment.

5.0 CONCLUSION

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

Principal Contributor: J. Hopkins

Date: June 18, 1996