

October 28, 1992

Docket No. 50-440

Mr. Michael D. Lyster, Vice President
Nuclear - Perry
The Cleveland Electric Illuminating
Company
10 Center Road
Perry, Ohio 44081

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Dear Mr. Lyster:

SUBJECT: AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-58
(TAC NO. M84061)

The Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications (TSs) in response to your application dated June 19, 1992, as supplemented by letter dated September 24, 1992.

This amendment revises the TSs by removing the component lists from Sections 3.6.4 and 3.8.4.1 in accordance with the guidelines set forth in NRC Generic Letter 91-08, "Removal of Component Lists from Technical Specifications." Other TS references to these tables and the associated Bases are also revised.

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

James R. Hall, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No.44 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

LA:DRPW
PKreutzer
10/28/92

PM:PD33:DRPW
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10/28/92

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10/28/92

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

October 28, 1992

Docket No. 50-440

Mr. Michael D. Lyster, Vice President
Nuclear - Perry
The Cleveland Electric Illuminating
Company
10 Center Road
Perry, Ohio 44081

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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,

A handwritten signature in cursive script that reads "James R. Hall".

James R. Hall, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 44 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Michael D. Lyster
Cleveland Electric Illuminating Company

Perry Nuclear Power Plant
Unit Nos. 1 and 2

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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. NPF-58

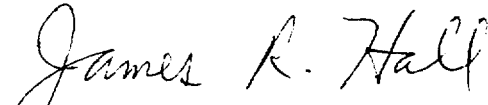
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated June 19, 1992, supplemented September 24, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-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. 44 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.

FOR THE NUCLEAR REGULATORY COMMISSION



James R. Hall, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: October 28, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 44

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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. Overleaf pages are provided to maintain document completeness.

Remove

1-6
3/4 3-11 through
3/4 3-14

3/4 3-16
3/4 3-22
3/4 3-78
3/4 6-1 through
3/4 6-5

3/4 6-28
3/4 6-29
3/4 6-30 through
3/4 6-39

3/4 8-21
3/4 8-22
3/4 8-23
3/4 8-24
B 3/4 6-6
B 3/4 8-3

Insert

1-6
3/4 3-11 through
3/4 3-14

3/4 3-16
3/4 3-22
3/4 3-78
3/4 6-1 through
3/4 6-5

3/4 6-28
3/4 6-29
-

3/4 8-21
3/4 8-22
-
-
B 3/4 6-6
B 3/4 8-3

DEFINITIONS

LIMITING CONTROL ROD PATTERN

1.22 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.23 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LIQUID RADWASTE TREATMENT SYSTEM

1.24 The LIQUID RADWASTE TREATMENT SYSTEM is any process or control equipment used to reduce the amount or concentration of liquid radioactive materials prior to their discharge to UNRESTRICTED AREAS. It involves all the installed and available liquid radwaste management system equipment, as well as their controls, power instrumentation, and services that make the system functional.

LOGIC SYSTEM FUNCTIONAL TEST

1.25 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MEMBER(S) OF THE PUBLIC

1.26 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.27 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.28 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the radiological environmental monitoring program.

DEFINITIONS

OPERABLE - OPERABILITY

1.29 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.30 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.31 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59 or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.32 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.33 PRIMARY CONTAINMENT INTEGRITY shall exist when:

- a. All containment penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE containment automatic isolation system, or
 2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that may be opened as permitted by Specification 3.6.4.
- b. The containment equipment hatch is closed and sealed.
- c. Each containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- d. The containment leakage rates are in compliance with the requirements of Specification 3.6.1.2.
- e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

TABLE 3.3.2-1
ISOLATION ACTUATION INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)</u>	<u>APPLICABLE OPERATIONAL CONDITION</u>	<u>ACTION</u>
1. <u>PRIMARY CONTAINMENT ISOLATION</u>			
a. Reactor Vessel Water Level - Low, Level 2	2	1, 2, 3 and #	20
b. Drywell Pressure - High	2	1, 2, 3	20
c. Containment and Drywell Purge Exhaust Plenum Radiation - High	2 ^(b)	1, 2, 3 and *	21
d. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3 and #	20
e. Manual Initiation	2 ^(c)	1, 2, 3 and *	22
2. <u>MAIN STEAM LINE ISOLATION</u>			
a. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3	20
b. Main Steam Line Radiation - High	2	1, 2	23
c. Main Steam Line Pressure - Low	2	1	24
d. Main Steam Line Flow - High	2/line	1, 2, 3	23
e. Condenser Vacuum - Low	2	1, 2**, 3**	23
f. Main Steam Line Tunnel Temperature - High	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	2	1, 2, 3	23
h. Turbine Building Main Steam Line Temperature - High	2	1, 2, 3	23
i. Manual Initiation	2	1, 2, 3	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)</u>	<u>APPLICABLE OPERATIONAL CONDITION</u>	<u>ACTION</u>
3. <u>SECONDARY CONTAINMENT ISOLATION</u>			
a. Reactor Vessel Water Level - Low, Level 2	2	1, 2, 3 and #	25
b. Drywell Pressure - High	2	1, 2, 3	25
c. Manual Initiation	2	1, 2, 3	22
	2	*	25
4. <u>REACTOR WATER CLEANUP SYSTEM ISOLATION</u>			
a. Δ Flow - High	1	1, 2, 3	27
b. Δ Flow Timer	1	1, 2, 3	27
c. Equipment Area Temperature - High	1	1, 2, 3	27
d. Equipment Area Δ Temperature - High	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low, Level 2	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temperature - High	1	1, 2, 3	27
h. SLCS Initiation	1	1, 2, 3	27
i. Manual Initiation	2	1, 2, 3	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)</u>	<u>APPLICABLE OPERATIONAL CONDITION</u>	<u>ACTION</u>
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>			
a. RCIC Steam Line Flow - High	1	1, 2, 3	27
b. RCIC Steam Supply Pressure - Low	1	1, 2, 3	27
c. RCIC Turbine Exhaust Diaphragm Pressure - High	2	1, 2, 3	27
d. RCIC Equipment Room Ambient Temperature - High	1	1, 2, 3	27
e. RCIC Equipment Room Δ Temperature - High	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temperature - High	1	1, 2, 3	27
h. Main Steam Line Tunnel Temperature Timer	1	1, 2, 3	27
i. RHR Equipment Room Ambient Temperature - High	1/Area	1, 2, 3	27
j. RHR Equipment Room Δ Temperature - High	1/Area	1, 2, 3	27
k. RCIC Steam Line Flow High Timer	1	1, 2, 3	27
l. Drywell Pressure - High	1	1, 2, 3	27
m. Manual Initiation	1	1, 2, 3	26

PERRY - UNIT 1

3/4 3-13

Amendment No. 44

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)</u>	<u>APPLICABLE OPERATIONAL CONDITION</u>	<u>ACTION</u>
6. <u>RHR SYSTEM ISOLATION</u>			
a. RHR Equipment Area Ambient Temperature - High	1/Area	1, 2, 3	28
b. RHR Equipment Area Δ Temperature - High	1/Area	1, 2, 3	28
c. RHR/RCIC Steam Line Flow - High	1	1, 2, 3	28
d. Reactor Vessel Water Level - Low, Level 3	2	1, 2, 3	28
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	2	1, 2, 3	28
f. Drywell Pressure - High	2	1, 2, 3	28
g. Manual Initiation	2	1, 2, 3	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
ACTION

- ACTION 20 - In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours. In OPERATIONAL CONDITION #, suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel.
- ACTION 21 - Close the affected system isolation valve(s) within one hour or:
a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel.
- ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or:
a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In OPERATIONAL CONDITION *, suspend CORE ALTERATIONS, operations with a potential for draining the reactor vessel, and handling of irradiated fuel in the primary containment.
- ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- ACTION 24 - Be in at least STARTUP within 6 hours.
- ACTION 25 - Verify SECONDARY CONTAINMENT INTEGRITY with the annulus exhaust gas treatment system operating within one hour.
- ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within 1 hour and declare the affected system inoperable.
- ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.
- ACTION 28 - Within one hour lock the affected system isolation valves closed, or verify, by remote indication, that the valve(s) is closed and electrically disarmed, or isolate the penetration(s) and declare the affected system inoperable.

NOTES

- * When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.
- ** When any turbine stop valve is greater than 90% open and/or the key locked Condenser Low Vacuum Bypass Switch is in the normal position.
- # During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
ACTION

NOTES (Continued)

- (a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
- (b) Containment and Drywell Purge System inboard and outboard isolation valves each use a separate two out of two isolation logic.
- (c) There is only one (1) RCIC manual initiation channel for RCIC system containment isolation valves.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

<u>TRIP FUNCTION</u>	<u>RESPONSE TIME (Seconds)#</u>
<u>1. PRIMARY CONTAINMENT ISOLATION</u>	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Containment and Drywell Purge Exhaust Plenum Radiation - High ^(b)	< 10 ^(a)
d. Reactor Vessel Water Level - Low, Level 1	NA
e. Manual Initiation	NA
<u>2. MAIN STEAM LINE ISOLATION</u>	
a. Reactor Vessel Water Level - Low, Level 1	< 1.0*/< 10 ^{(a)**}
b. Main Steam Line Radiation - High ^(b)	< 1.0*/< 10 ^{(a)**}
c. Main Steam Line Pressure - Low	< 1.0*/< 10 ^{(a)**}
d. Main Steam Line Flow - High	< 0.5*/< 10 ^{(a)**}
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Turbine Building Main Steam Line Temperature - High	NA
i. Manual Initiation	NA
<u>3. SECONDARY CONTAINMENT ISOLATION</u>	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Manual Initiation	NA
<u>4. REACTOR WATER CLEANUP SYSTEM ISOLATION</u>	
a. Δ Flow - High	NA
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low, Level 2	NA
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

<u>TRIP FUNCTION</u>	<u>RESPONSE TIME (Seconds)#</u>
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>	
a. RCIC Steam Line Flow - High	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Ambient Temperature - High	NA
e. RCIC Equipment Room Δ Temperature - High	NA
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Tunnel Temperature Timer	NA
i. RHR Equipment Room Ambient Temperature - High	NA
j. RHR Equipment Room Δ Temperature - High	NA
k. RCIC Steam Line Flow High Timer	NA
l. Drywell Pressure - High	NA
m. Manual Initiation	NA
6. <u>RHR SYSTEM ISOLATION</u>	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. RHR/RCIC Steam Line Flow - High	NA
d. Reactor Vessel Water Level - Low, Level 3	NA
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
g. Manual Initiation	NA

- (a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays.
- (b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each containment isolation valve shall be added to the isolation time for each valve to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>REQUIRED NUMBER OF CHANNELS</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE OPERATIONAL CONDITIONS</u>	<u>ACTION</u>
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water Level	2	1	1,2,3	80
3. Suppression Pool Water Level	2	1	1,2,3	80
4. Suppression Pool Water Temperature	16, 2/sector	8, 1/sector	1,2,3	80
5. Primary Containment Pressure	2	1	1,2,3	80
6. Primary Containment Air Temperature	2	1	1,2,3	80
7. Drywell Pressure	2	1	1,2,3	80
8. Drywell Air Temperature	2	1	1,2,3	80
9. Primary Containment and Drywell Hydrogen Concentration Analyzer and Monitor	2	1	1,2,3	80
10. Safety/Relief Valve Position Indicators**	2/valve	1/valve	1,2,3	80
11. Primary Containment/Drywell Area Gross Gamma Radiation Monitors	2*	1*	1,2,3	81
12. Offgas Ventilation Exhaust Monitor#	1	1	1,2,3	81
13. Turbine Building/Heater Bay Ventilation Exhaust Monitor#	1	1	1,2,3	81
14. Unit 1 Vent Monitor#	1	1	1,2,3	81
15. Unit 2 Vent Monitor#	1	1	1,2,3	81
16. Neutron Flux				
a. Average Power Range	2	1	1,2,3	80
b. Intermediate Range	2	1	1,2,3	80
c. Source Range	2	1	1,2,3	80
17. Primary Containment Isolation Valve Position***	2/valve	1/valve	1,2,3	82

* Each for primary containment and drywell.

** One channel consists of a pressure switch on the SRV discharge pipe, the other channel consists of a temperature sensor on the SRV discharge pipe.

*** One channel consists of the open limit switch, and the other channel consists of the closed limit switch for each automatic containment isolation valve.

High and intermediate range D19 system noble gas monitors.

PERRY - UNIT 1

3/4 3-78

Amendment No. 44

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.1.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

- a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 11.31 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.
- b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE primary containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that may be opened as permitted by Specification 3.6.4.
- c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- d. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, drywell, or the steam tunnel portion of the auxiliary building, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.6.1.1.2 PRIMARY CONTAINMENT INTEGRITY* shall be maintained.#

APPLICABILITY:

When irradiated fuel is being handled in the primary containment, and during CORE ALTERATIONS, and operations with a potential for draining the reactor vessel. Under these conditions, the requirements of PRIMARY CONTAINMENT INTEGRITY do not apply to normal operation of the inclined fuel transfer system.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, suspend handling of irradiated fuel in the primary containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.6.1.1.2 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

- a. At least once per 31 days by verifying that all primary containment penetrations not capable of being closed by OPERABLE primary containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that may be opened as permitted by Specification 3.6.4.#
- b. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

*The primary containment leakage rates in accordance with Specification 3.6.1.2 are not applicable.

#Except that six (6) 3/4" vent and drain line pathways may be opened for the purpose of performing containment isolation valve leak rate testing provided the plant has been subcritical for at least seven (7) days.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

- a. An overall integrated leakage rate of less than or equal to $0.75 L_a$, 0.20 percent by weight of the primary containment air per 24 hours^a at P_a , 11.31 psig.
- b. A combined leakage rate of less than or equal to $0.60 L_a$ for all penetrations and all valves, except for main steam line^a isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests when pressurized to P_a , 11.31 psig.
- c. Less than or equal to 25 scf per hour for any one main steam line through the isolation valves when tested at P_a , 11.31 psig.
- d. A combined leakage rate of less than or equal to $0.0504 L_a$ for all penetrations that are secondary containment bypass leakage paths when pressurized to the required test pressure.
- e. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at $1.10 P_a$, 12.44 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

With:

- a. The measured overall integrated primary containment leakage rate exceeding $0.75 L_a$, or
- b. The measured combined leakage rate for all penetrations and all valves except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests exceeding $0.60 L_a$, or
- c. The measured leakage rate exceeding 25 scf per hour for any one main steam line through the isolation valves, or
- d. The combined leakage rate for all penetrations that are secondary containment bypass leakage paths exceeding $0.0504 L_a$, or
- e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves:

*See Special Test Exception 3.10.1.

CONTAINMENT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore:

- a. The overall integrated leakage rate(s) to less than or equal to $0.75 L_a$, and
- b. The combined leakage rate for all penetrations and all valves, except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests to less than or equal to $0.60 L_a$, and
- c. The leakage rate to less than 25 scf per hour for any one main steam line through the isolation valves, and
- d. The combined leakage rate for all penetrations that are secondary containment bypass leakage paths to less than or equal to $0.0504 L_a$, and
- e. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972 and BN-TOP-1; test results shall also be reported based on the Mass Point Methodology described in ANSI/ANS N56.8-1981:

- a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a , 11.31 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.
- b. If any periodic Type A test fails to meet $0.75 L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet $0.75 L_a$, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet $0.75 L_a$, at which time the above test schedule may be resumed.
- c. The accuracy of each Type A test shall be verified by a supplemental test which:

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within $0.25 L_a$. The formula to be used is:
$$[L_o + L_{am} - 0.25 L_a] \leq L_c \leq [L_o + L_{am} + 0.25 L_a]$$
 where L_c = supplemental test result; L_o = superimposed leakage;
 L_{am} = measured Type A leakage.
2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
3. Requires the quantity of gas injected into the primary containment or bled from the primary containment during the supplemental test to be between $0.75 L_a$ and $1.25 L_a$.
- d. Type B and C tests shall be conducted with gas at P_a , 11.31 psig*, at intervals no greater than 24 months except for tests involving:
 1. Air locks,
 2. Main steam line isolation valves,
 3. Valves pressurized with fluid from a seal system,
 4. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and
 5. Purge supply and exhaust isolation valves with resilient material seals.
- e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.
- f. Main steam line isolation valves shall be leak tested at least once per 18 months.
- g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J of 10 CFR 50 Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least $1.10 P_a$, 12.44 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.
- h. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

*Unless a hydrostatic test is required.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.3. and 4.6.1.8.4.
- j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, and 4.6.1.2.d.

(Next page is 3/4 6-6.)

CONTAINMENT SYSTEMS

SUPPRESSION POOL MAKEUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.4 The suppression pool makeup system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

- a. With one suppression pool makeup line inoperable, restore the inoperable makeup line to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With the upper containment pool water level less than the limit, restore the water level to within the limit within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- c. With upper containment pool water temperature greater than the limit, restore the upper containment pool water temperature to within the limit within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.4 The suppression pool makeup system shall be demonstrated OPERABLE:

- a. At least once per 24 hours by verifying the upper containment pool water:
 1. Level to be greater than or equal to 22'10" above the reactor pressure vessel flange, and
 2. Temperature to be less than or equal to 100°F.
- b. At least once per 31 days by verifying that:
 1. The steam dryer storage/reactor well pool gate is removed and the fuel transfer pool gate is in place.
 2. Each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secure in position, is in its correct position.
- c. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual makeup of water to the suppression pool may be excluded from this test.

CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.#

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and **.

ACTION:

- a. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:
1. Restore the inoperable valve(s) to OPERABLE status, or
 2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or
 3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION a.2 or a.3 above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL CONDITION**, suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

**When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#The Containment Vessel and Drywell Purge system 42-inch inboard purge valves 1M14-F045 and -F085 are not required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2 and 3. The RCIC system containment isolation valves are not required to be OPERABLE in OPERATIONAL CONDITION **. The Fire Protection system manual hose reel containment isolation valves 1P54-F726 and -F727 may be opened as necessary to supply fire mains in OPERATIONAL CONDITION **. Locked or sealed closed isolation valves may be opened on an intermittent basis under administrative controls.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.4.2 Each automatic containment isolation valve shall be demonstrated OPERABLE at least once per 18 months by verifying that on an isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.4.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

Pages 3/4 6-30 through 3/4 6-39 are deleted.

ELECTRICAL POWER SYSTEMS

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 Primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary and backup containment penetration conductor overcurrent protective devices inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

1. For 13.8 kV circuit breakers, de-energize the 13.8 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.
2. For 120-volt circuit breakers, remove the inoperable circuit breaker(s) from service by racking out* the breaker within 72 hours and verify the inoperable breaker(s) to be racked out* at least once per 7 days thereafter.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 13.8 kV circuits which have their redundant circuit breakers tripped or to 120-volt circuits which have the inoperable circuit breaker racked out.*

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary and backup containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage 13.8 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing:
 - a) A CHANNEL CALIBRATION of the associated protective relays,
 - b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed, and

**Racking out may be accomplished by tripping the breaker under administrative control.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.
- 2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers' nominal setpoint including the instantaneous element setpoint and measuring the response time. The measured response time shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.
- b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with instructions prepared in conjunction with its manufacturer's recommendations.

Pages 3/4 8-23 and 3/4 8-24 are deleted.

CONTAINMENT SYSTEMS

BASES

DEPRESSURIZATION SYSTEMS (Continued)

In addition to the limits on temperature of the suppression pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, and (3) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve, where possible, to assure mixing and uniformity of energy insertion to the pool.

The containment spray system consists of two 100% capacity loops, each with three spray rings located at different elevations about the inside circumference of the containment. RHR pump A supplies one loop and RHR pump B supplies the other. RHR pump C cannot supply the spray system. Dispersion of the flow of water is effected by 345 nozzles in each loop, enhancing the condensation of water vapor in the containment volume and preventing overpressurization. Heat rejection is through the RHR heat exchangers. The turbulence caused by the spray system aids in mixing the containment air volume to maintain a homogeneous mixture for H₂ control.

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the system is to ensure containment integrity following a LOCA by preventing excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related component systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the emergency service water is accomplished in the RHR heat exchangers.

The suppression pool make-up system provides water from the upper containment pool to the suppression pool by gravity flow through two 100% capacity dump lines following a LOCA. The quantity of water provided is sufficient to account for all conceivable post-accident entrapment volumes, ensuring the long term energy sink capabilities of the suppression pool and maintaining the water coverage over the uppermost drywell vents. During refueling, there will be administrative control to ensure the make-up dump valves will not be opened.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

CONTAINMENT SYSTEMS

BASES

All required Containment Isolation Valves are listed in the PNPP Unit 1 Plant Data Book. The opening of normally locked or sealed closed containment isolation valves under administrative controls in accordance with footnote # includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment. The above considerations do not apply to the normally locked closed (LC) Fire Protection system manual hose reel containment isolation valves 1P54-F726 and -F727 when opened as necessary to supply fire mains when handling irradiated fuel in the primary containment, during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

3/4.6.5 VACUUM RELIEF

3/4.6.5.1 CONTAINMENT VACUUM RELIEF AND 3/4.6.5.2 CONTAINMENT HUMIDITY CONTROL

Vacuum breakers are provided on the containment to prevent an excessive vacuum from developing inside containment during an inadvertent or improper operation of the containment spray. Four vacuum breakers and their associated isolation valves are provided. Any two vacuum breakers provide 100% vacuum relief.

The containment vacuum relief system is designed to prevent an excessive vacuum from being created inside the containment following inadvertent initiation of the containment spray system. By maintaining temperature/relative humidity within the limits for acceptable operation shown on Figure 3.6.5.2-1, the maximum containment vacuum created by actuation of both containment spray loops will be limited to approximately -0.7 psig.

3/4.6.5.3 DRYWELL VACUUM BREAKERS

Drywell vacuum breakers are provided on the drywell to prevent drywell flooding due to differential pressure across the drywell and to equalize pressure between the drywell and containment.

Two drywell vacuum breakers and their associated isolation valves are provided. Any one vacuum breaker can provide full vacuum relief capability.

3/4.6.6 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Shield Building provides secondary containment during normal operation when the containment is sealed and in service. At other times, the containment may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the annulus with the annulus exhaust gas treatment system, along with the surveillance of the doors, hatches, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the annulus exhaust gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site

ELECTRICAL POWER SYSTEMS

BASES

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance. A list of required circuit breakers is contained in the PNPP Unit 1 Plant Data Book.

The surveillance requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-58

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

On June 19, 1992, as supplemented September 24, 1992, the Cleveland Electric Illuminating Company (the licensee) requested an amendment to Facility Operating License No. NPF-58, for the Perry Nuclear Power Plant (PNPP), Unit 1. The proposed amendment would remove certain Technical Specification (TS) tables that include lists of components referenced in individual specifications. In addition, the TS requirements have been modified such that all references to these tables have been removed. Finally, the TSs have been modified to state requirements in general terms that include the components listed in the tables removed from the TSs. Guidance on the proposed TS changes was provided by Generic Letter (GL) 91-08, dated May 6, 1991.

A "Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for a Hearing" for the proposed action was published in the Federal Register on August 19, 1992 (57 FR 37560). The additional information provided by the licensee on September 24, 1992 did not affect the staff's determination as documented in the referenced notice.

2.0 EVALUATION

The licensee proposes to relocate the component lists for Containment Isolation Valves (Table 3.6.4-1) and Containment Penetration Conductor Overcurrent Protection Devices (Table 3.8.4.1-1) from the PNPP Unit 1 TSs to the PNPP Operations Manual, Volume 18 (the Plant Data Book). In addition, TSs referencing those tables will also be revised, and footnotes in those tables that modify the TS requirements will be retained in new Limiting Conditions for Operation, as appropriate.

2.1 Removal of Table 3.6.4-1

The licensee has proposed the removal of Table 3.6.4-1, "Containment Isolation Valves," referenced in TS 3/4.6.4. With the removal of this table, the

statement of the Limiting Condition for Operation (LCO) for TS 3.6.4 will read as follows:

Each containment isolation valve shall be OPERABLE.

The licensee has also revised the Action requirements under TS 3.6.4, and the Surveillance Requirements of TS 4.6.4.1 through 4.6.4.3 to remove all references to Table 3.6.4-1. Footnote "#" is added to the revised LCO, to retain certain exceptions currently allowed in footnotes to Table 3.6.4-1. These include:

- a. A provision stating that Containment Vessel and Drywell Purge system 42-inch inboard purge valves 1M14-F045 and -F085 are not required to be OPERABLE in OPERATIONAL CONDITION 1, 2 and 3.
- b. A provision stating that the Reactor Core Isolation Cooling system containment isolation valves are not required to be OPERABLE in OPERATIONAL CONDITION ** (when handling irradiated fuel in the primary containment, during CORE ALTERATIONS and operations with a potential for draining the reactor vessel).
- c. A provision allowing Fire Protection system manual hose reel containment isolation valves 1P54-F726 and -F727 to be opened as necessary to supply fire mains in OPERATIONAL CONDITION **.

In addition, footnote # includes a statement that locked or sealed closed isolation valves may be opened on an intermittent basis under administrative controls. This provision will apply to all containment isolation valves not addressed above that have the locked or sealed closed feature consistent with General Design Criteria 55, 56 and 57 of Appendix A to 10 CFR Part 50. The licensee has also revised TS Bases Section 3/4.6.4, "Containment Isolation Valves," to better define the intent of opening valves under administrative controls, as recommended by the staff in GL 91-08.

The information deleted from Table 3.6.4-1 will be retained in the PNPP Plant Data Book, which will be controlled via the requirements for procedures in TS Section 6.0, "Administrative Controls." The proposed changes to TS 3.6.4, including the removal of Table 3.6.4-1 and changes to the associated Bases, are consistent with the guidance of Generic Letter 91-08 and are therefore acceptable.

2.2 References to Table 3.6.4-1

As a result of the proposed removal of Table 3.6.4-1, numerous references to the table throughout the TS also need to be modified for consistency. TS definition 1.33, "Primary Containment Integrity" and Surveillance Requirements 4.6.1.1.1.b and 4.6.1.1.2.a, "Primary Containment Integrity-Operating" and "Shutdown," respectively, have been modified to delete the reference to Table 3.6.4-1, and to allow the opening of certain valves as specified in new TS 3.6.4. Although the words "under administrative control" are omitted, the revised statements are consistent with the intent of GL 91-08 because new TS 3.6.4 does specify that locked or sealed closed isolation valves may be opened

on an intermittent basis under administrative controls. Therefore, this requirement is implicit in the revised statements, and these changes are acceptable.

TS Tables 3.3.2-1, "Isolation Actuation Instrumentation", 3.3.2-3, "Isolation Actuation Instrumentation Response Time" and 3.3.7.5-1, "Accident Monitoring Instrumentation" are revised to remove explicit references to valve groups and isolation times listed in Table 3.6.4-1. Similarly, TS 3.6.1.2, "Primary Containment Leakage" and the associated Surveillance requirements are revised to remove all references to Table 3.6.4-1, consistent with GL 91-08. These changes are editorial and do not revise existing requirements; therefore, they are acceptable.

2.3 Removal of Table 3.8.4.1-1

The licensee has proposed the removal of Table 3.8.4.1-1, "Containment Penetration Conductor Overcurrent Protective Devices" and revisions to the associated TS 3/4.8.4.1. The revised statements in the LCO and Surveillance Requirements are consistent with the guidance of Generic Letter 91-08 and do not change any existing requirements. The Bases are also revised to indicate that the list of devices is included in the PNPP Plant Data Book.

2.4 Summary

On the basis of the above review, the staff finds that the proposed changes to the TS for the Perry Nuclear Power Plant, Unit 1, are primarily administrative in nature and do not alter the requirements set forth in the existing TS. Overall, these changes will allow the licensee to make corrections and updates to the lists of components for which these TS requirements apply, under the provisions that control plant procedures as specified in Section 6.0, Administrative Controls, of the PNPP Unit 1 TS. These controls include requirements for the review of changes in accordance with 10 CFR 50.59, review and approval by the Plant Operations Review Committee, and approval by the plant General Manager. Therefore, the staff finds that the proposed TS changes are 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 a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves 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 this amendment involves no significant hazards

consideration and there has been no public comment on such finding (57 FR 37560). 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.R. Hall

Date: October 28, 1992