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

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. 86
License No. NPF-58

1. The U.S. 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, OES Nuclear, Inc., Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated January 31, 1997, supplemented August 6, 1997, 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. 86 , 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
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Attachment: Changes to the Technical Specifications

Date of Issuance: September 9, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 86

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 contains vertical lines indicating the area of change.

Remove	Insert
1.0-3	1.0-3
3.6-2	3.6-2
3.6-7	3.6-7
3.6-18	3.6-18
3.6-19	3.6-19
3.6-60	3.6-60
5.0-15	5.0-15
- -	5.0-15a

1.1 Definitions (continued)

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. Exceptions are stated in the individual surveillance requirements.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or the turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. Exceptions are stated in the individual surveillance requirements.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.2.1	------------------NOTES------------------ 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. During MODES 1, 2, and 3, results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. -- Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.2.2	Verify primary containment air lock seal air header pressure is ≥ 90 psig.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.9	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.0504 L_a when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.10	------------------NOTE------------------ Only required to be met in MODES 1, 2, and 3. -- Verify leakage rate through each main steam line is ≤ 25 scfh when tested at ≥ P_a. Until the end of Operating Cycle 6, the leakage rate through one main steam line is limited to ≤ 35 scfh when tested at ≥ P_a, as long as the total leakage rate through all four main steam lines is ≤ 100 scfh.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

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 ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.1	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit.	------NOTE----- The performance of the drywell bypass leakage test is extended to the sixth refueling outage and need not be performed during the fifth refueling outage. --------------- 18 months
SR 3.6.5.1.2	Visually inspect the exposed accessible interior and exterior surfaces of the drywell.	Three times during each 10-year service period, at approximately equal intervals.

(continued)

5.5 Programs and Manuals

5.5.10 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases for these TS.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the USAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that meet the criteria of Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- BN-TOP-1 methodology may be used for Type A tests.
- The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996, are considered to be an integral part of NEI 94-01.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 7.80 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.20% of primary containment air weight per day at the calculated peak containment pressure (P_a).

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $< 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with this Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq$ 2.5 scfh when tested at $\geq P_a$.

   2) For each door, leakage rate is $\leq$ 2.5 scfh when the gap between the door seals is pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.