January 22, 1988

Docket No. 50-440

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Mr. Alvin Kaplan
Vice President
The Cleveland Electric
Illuminating Company
10 Center Road
Perry, Ohio 44081

Dear Mr. Kaplan:

SUBJECT: TECHNICAL SPECIFICATION CHANGE REQUEST TO EXTEND SPECIFIED VALUE LOCAL LEAK RATE TESTS (LLRTs)

RE: Perry Nuclear Power Plant, Unit No. 1

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 11, 1987, as amended September 18, 1987, and supplemented January 8, 1988.

This amendment revises Technical Specification Sections 4.4.3.2.2 and 4.6.1.2 to extend specified valve local leak rate tests (LLRTs) until the first refueling outage, currrently scheduled to begin in January 1989. These tests will become overdue beginning January 24, 1988.

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

Sincerely,

Original signed by

Timothy G. Colburn, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 10 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Alvin Kaplan
The Cleveland Electric
Illuminating Company

Perry Nuclear Power Plant
Unit 1

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

CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. NPF-58

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

   A. The application for amendment by Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated September 11, 1987, as amended September 18, 1987, and supplemented January 8, 1988, 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:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 10 , are hereby incorporated into this license. 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

Martin J. Virgilio, Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 10

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 page(s) provided to maintain document completeness.

Remove	Insert
3/4 4-11	3/4 4-11
3/4 6-5	3/4 6-5
	3/4 6-5a

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate or gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment sump flow rate at least once per 12 hours,

c. Monitoring the drywell upper drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months,**

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate, and

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

**P.I.V. leak test extension to the first refueling outage is permissible for each Reactor Coolant System P.I.V. listed in Table 3.4.3.2-1, which are identified in letter PY-CEI/NRR-0714L (dated September 11, 1987) as needing a plant outage to test. For this one time test interval, the provisions of Specification 4.0.2 are not applicable.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
1C41-F006	Standby Liquid Control
1C41-F007	Standby Liquid Control
1E12-F008	Residual Heat Removal
1E12-F009	Residual Heat Removal
1E12-F041A	Residual Heat Removal
1E12-F041B	Residual Heat Removal
1E12-F041C	Residual Heat Removal
1E12-F042A	Residual Heat Removal
1E12-F042B	Residual Heat Removal
1E12-F042C	Residual Heat Removal
1E12-F550	Residual Heat Removal
1E21-F005	Low Pressure Core Spray
1E21-F006	Low Pressure Core Spray
1E22-F004	High Pressure Core Spray
1E22-F005	High Pressure Core Spray
1E51-F065	Reactor Core Isolation Cooling
1E51-F066	Reactor Core Isolation Cooling

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 per Table 3.6.4-1 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 per Table 3.6.4-1 which penetrate the primary containment shall be leak tested at least once per 18 months.#

*Unless a hydrostatic test is required per Table 3.6.4-1.

**Except for valves 1B21-F022A and 1B21-F028A, which shall be leak tested prior to July 12, 1987. This exception expires on July 12, 1987.

#A Type C test interval extension to the first refueling outage is permissible for primary containment isolation valves listed in Table 3.6.4-1, which are identified in letter PY-CEI/NRR-0714 L (dated September 11, 1987) as needing a plant outage to test. For this one time test interval, the provisions of Specification 4.0.2 are not applicable.

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.)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 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 September 11, 1987, as amended September 18, 1987, and supplemented January 8, 1988, Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) requested an amendment to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. The proposed amendment would extend the interval for 16 specified containment isolation or pressure isolation valve local leak rate tests (LLRTs) until the first refueling outage, currently scheduled for January 1989. The LLRTs for these valves would otherwise begin to become overdue on January 24, 1988. By a separate letter dated September 11, 1987, the licensees also requested a one-time exemption from the schedular requirements of Section III.D.3 of Appendix J to 10 CFR Part 50 concerning LLRT testing intervals for all of the containment isolation valves (14 of these valves). The exemption would defer testing of the valves until the first refueling outage.

2.0 EVALUATION

Technical Specification 4.6.1.2.d requires LLRTs (Type C tests) on the primary containment isolation valves listed in Table 3.6.4-1 to be performed at intervals no greater than 24 months except for containment isolation valves in hydrostatically tested lines penetrating the primary containment, which shall be leak tested at least once per 18 months per Technical Specification 4.6.1.2.h. The Commission's regulations (10 CFR 50, Appendix J, Section III.D.3) require LLRTs (Type C tests) to be performed during each reactor shutdown for refueling, but in no case at intervals greater than two years. Technical Specification 4.4.3.2.2.a requires leak rate tests for Reactor Coolant System pressure isolation valves (PIVs) listed in Table 3.4.3.2-1 to be performed at least once per 18 months. An additional 25 percent may be added to the 18-month interval per Specification 4.0.2.a. The licensees have requested that the initial 18 and 24-month testing intervals for 16 valves be extended on a one-time basis until the first refueling outage presently scheduled for January 1989. These valves would otherwise become overdue for testing between January 24 and June 15, 1988.

Testing of these valves requires one or more of the following plant conditions:

1) Drywell head removal.
2) Both Residual Heat Removal (RHR) shutdown cooling loops rendered inoperable.
3) Potentially reducing the number of Emergency Core Cooling System (ECCS) and/or shutdown cooling loops below the Technical Specification required systems (when taken in conjunction with other planned necessary outage work).

The licensees have currently scheduled drywell head removal to occur during the first refueling outage. To render both loops of RHR shutdown cooling inoperable, the licensees would either be required to remove the drywell and reactor heads and flood the vessel, or wait until decay heat is reduced such that ambient losses are sufficient to maintain cold shutdown. No planned outages of this duration will occur until the first refueling outage.

The requested extension became necessary as a result of delays in attaining full power operation common to initial startup activities and no planned or unplanned outages occurred during the startup test interval of sufficient duration to allow testing of these valves without significantly extending the outages for the sole purpose of conducting these tests.

The 2-year and 18-month ± 25% testing intervals for containment isolation, and reactor coolant pressure isolation valves, respectively, are intended to be often enough to prevent significant deterioration from occurring and long enough to permit LLRTs to be performed during plant outages. This provides added assurance of Reactor Coolant System valve integrity thereby reducing the probability of gross valve failure and consequent intersystem loss of coolant accident. It also provides assurance that the overall containment leakage limits will not exceed the value assumed in the accident analysis even accounting for possible degradation of the leakage barriers between leakage tests.

A normal reactor fuel load is designed to provide an 18-month cycle with approximately 16 months of full power operations. Consequently, the primary containment/pressure isolation valves are normally exposed to 18 months of rated temperature conditions between each leak rate test. Since the initial leak rate tests at the Perry Nuclear Power Plant, these valves will have been subjected to rated temperature conditions approximately equal to one 18-month operating cycle by the first refueling outage. An extension of the isolation valve test interval to the first refueling outage would not result in exposure of these valves to temperature/pressure profiles of greater length than will be expected during subsequent refueling outage intervals.

The licensees have stated that the isolation valves which are the subject of this amendment request and the related exemption request to Appendix J of 10 CFR Part 50 were all tested successfully in early 1986. The total of the Type C leakage rates for these valves is not a significant portion (4.13%) of the allowable leakage limit (0.6 L_a). Reactor Coolant System PIV leakage rates are all less than 5% of their allowable leakage rates. Deterioration in the overall integrity of non-flow modulating (either full open or full closed) isolation valves is normally expected to be a gradual process. By letter dated January 8, 1988, in response to staff requests for additional information (via telecon in December 1987 and January 1988), the licensees provided information relative to industry performance of these valves as a comparative check to the Perry Nuclear Power Plant valve performance history.

There are a total of 12 different valve groupings for which an extension was requested. The licensees' review of INPO's Nuclear Plant Reliability Data System (NPRDS) to determine how many leak rate test failures had been reported on the 12 types of valves indicated that five of the valve groups did not experience any leak rate failures. For the seven valve groups which had experienced at least one leak rate failure, the specific valve manufacturer was contacted to try to determine an approximate number of each type of valve that was being used in the nuclear industry. Based on the information from the valve manufacturers, the licensees concluded that the valves experiencing leak rate failure is a very small percentage of the total valve population (on the order of 1 to 2% failure rate). This is without considering multiple successful testing of most of these valves in the industry. It should be also noted that the valves in the NPRDS data base for the most part have been in service for significant periods whereas the valves in the Perry plant will have experienced only a part of the first fuel cycle power operation time by the date of the proposed testing. The NPRDS data do not suggest that these valves will experience excessive leakage during this time period.

The second industry review performed by the licensees was to contact three BWR plants which had been granted extensions of leak test intervals on valves to try to determine if the valves experienced a higher leak rate due to the testing extension. The three utilities reviewed a total of 49 valves with the following results:

Leak rate <u>stayed the same</u> as previous test result -10
Leak rate <u>decreased</u> from previous test result - 18
Leak rate <u>increased</u> from previous test result - 21

The test interval extensions for the plants ranged from 16 days to 100 days. Several of the increased leak rates reported were very small, and a number of others were due to crud deposits found under the seating surfaces, not due to valve malfunctions. One utility with extensive testing history reported that the valves that showed an increase after the test interval extension were ones which had also shown increased leak rates in previous normal interval testing. The increased leak rates on the valves granted an extension did not result in exceeding the 10 CFR 50, Appendix J, limits at any of the plants. From the information collected, a direct relationship between test intervals and increased leak rates could not be determined. Thus, there was nothing found in the information which would indicate that the extended test intervals would have a sudden detrimental effect on the overall leak rates of the valves involved.

The licensees have further committed to performing the LLRTs for the valves which are the subject of this amendment request if an unplannned outage of sufficient duration were to occur prior to the refueling outage with the exception of three valves which require removal of the drywell head for testing. The licensees have stated that testing of these valves can only be performed during a refueling outage.

The NRC staff has determined that the licensees' request for extension of the requested Reactor Coolant System pressure isolation valve and containment isolation valve LLRTs until the first refueling outage will not present a significant safety concern and is therefore acceptable based on the following considerations:

1. The integrated temperature/pressure profiles seen by the valves which are the subject of the extension request are not significantly greater than expected for subsequent refueling cycle test intervals.

2. The favorable results of previous LLRTs performed at the Perry Nuclear Power Plant coupled with the small contribution to allowable leakage, confirmatory industry experience and expected gradual deterioration of valves of these types provide reasonable assurance that the granting of the requested extension will not result in a significant decrease in the integrity of the penetrations.

3. The 24-month interval requirement for Type B and C penetrations is intended to be often enough to prevent significant deterioration from occurring and long enough to permit the LLRTs to be performed during plant outages. Leak testing of the penetrations during plant shutdown is preferable because of the lower radiation exposures to plant personnel. Moreover, some penetrations, because of their intended functions, cannot be tested at power operation. For penetrations that cannot be tested during power operation or those that, if tested during plant operation would cause a degradation in the plant's overall safety (e.g., the closing of a redundant line in a safety system), the increase in confidence of containment integrity following a successful test is not significant enough to justify a plant shutdown specifically to perform the LLRTs within the 24-month time period, as long as the penetrations are in compliance with Items 1 and 2 above.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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.

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.

4.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and 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: Timothy Colburn

Dated: January 22, 1988