September 28, 1990

Docket No. 50-440

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

Dear Mr. Lyster:

SUBJECT: AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. NPF-58 PRIMARY CONTAINMENT INTEGRITY-SHUTDOWN, PERRY UNIT 1 (TAC NO. 76390)

The Commission has issued the enclosed Amendment No. 35 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications in response to your application dated March 16, 1990.

This amendment revises the Technical Specification for Primary Containment Integrity-Shutdown to allow Type C containment isolation valve local leak rate tests to be performed with up to six 3/4-inch vent and drain lines open on those penetrations that otherwise would not be testable when that specification is applicable. This provision is applicable only after the reactor has been subcritical for at least seven days.

Copies of the Safety Evaluation and of the notice of issuance are also enclosed. The notice of issuance has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original Signed By:

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

Enclosures:
1. Amendment No. 35 to
License No. NPF-58
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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Date:	9/11/90	9/21/90	9/21/90	9/24/90

DOCUMENT NAME: 76390 AMD

DF01

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The Cleveland Electric
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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. 35
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, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated March 16, 1990 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.

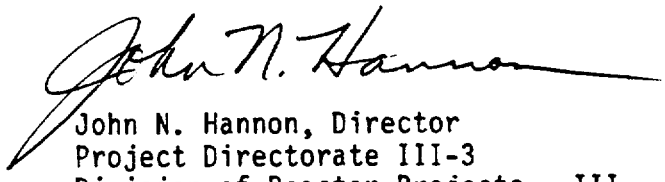
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 35 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



John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: September 28, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 35

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	Insert
3/4 6-2	3/4 6-2
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2a	B 3/4 6-2a

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 as provided in Table 3.6.4-1 of 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.

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*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 as provided in Table 3.6.4-1 of 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.

3/4.6.1 CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During shutdown when irradiated fuel is being handled in the primary containment, and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel, the # footnote permits the opening of six vent and drain pathways for the purpose of performing containment isolation valve leak rate surveillance testing provided the reactor has been subcritical for at least seven days. Offsite doses were calculated assuming the postulated fuel handling accident inside primary containment after a seven day decay time, and assuming all the airborne activity existing inside containment after the accident is immediately discharged directly to the environment (i.e., no containment). Although this analysis would indicate that no restriction on the number of vent and drain pathways was required, the number of open pathways was restricted to six for conservatism.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 11.31 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Overall integrated leakage rate means the leakage rate which obtains from a summation of leakage through all potential leakage paths. Where a leakage path contains more than one valve, fitting, or component in series, the leakage for that path will be that leakage of the worst leaking valve, fitting, or component and not the summation of the leakage of all valves, fittings, or components in that leakage path.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

3/4.6.1 CONTAINMENT (Continued)

3/4.6.1.2 CONTAINMENT LEAKAGE (Continued)

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemptions granted for testing the airlocks after each opening.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

The air supply to the containment air lock and seal system is the service and instrument air system. The system consists of two 100% capacity air compressors per unit and can be cross-connected. This system is redundant and extremely reliable and provides system pressure indication in the control room.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steam line isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIV's when isolation of the primary system and containment is required.

3/4.6.1.5 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment to secondary containment differential pressure ensure that the primary containment peak pressure of 11.31 psig does not exceed the design pressure of 15.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of +0.8 psid. The limit of -0.1 to +1.0 psid for initial positive primary containment to secondary containment pressure will limit the primary containment pressure to 11.31 psid which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 185°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND CONTAINMENT PURGE SYSTEM

The use of the drywell and containment purge lines is restricted to the 42-inch outboard and 18-inch purge supply and exhaust isolation valves. These valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The term sealed closed as used in this context means that the valve is secured in its closed position by deactivating the valve motor operator, and does not pertain to injecting seal water between the isolation valves by a seal water system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 35 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

By letter dated March 16, 1990, the Cleveland Electric Illuminating Company (CEI), et al. the licensees for the Perry Nuclear Power Plant, Unit 1, proposed changes to the plant technical specifications (TSs) to revise the Primary Containment Integrity requirements during fuel handling. The proposed changes would allow the opening of up to six 3/4-inch vent and drain line pathways during refueling activities, for the purpose of performing Type C local leak rate testing (LLRT) as required by 10 CFR Part 50, Appendix J. Under the proposed changes, these lines could be opened to perform Type C testing only after the reactor has been subcritical for at least 7 days. These changes were proposed to reduce the duration of refueling outages, by allowing limited Type C testing in parallel with fuel handling activities, while maintaining adequate control for establishing containment integrity.

2.0 EVALUATION

In Amendment Number 19 to Facility Operating License No. NPF-58, dated March 31, 1989, the staff approved changes to the Perry TSs to allow up to two vent and drain pathways to be opened during fuel movement for the performance of Type C tests. That approval was granted for the first refueling outage only and was based, in part, on the fact that the reactor had been subcritical for 15 days prior to issuance of the amendment. The current Perry TSs would require the suspension of fuel handling activities if primary containment integrity is not maintained. As a result, most Type C testing required by Appendix J could not be performed while refueling is in progress.

The licensees have submitted an analysis to demonstrate that following a 7-day decay period, offsite doses due to a fuel handling accident, assuming no containment (i.e., a direct release to the environment), would fall within the guidelines of NRC Standard Review Plan (SRP) Section 15.7.4 (less than 25% of the limits specified in 10 CFR Part 100). Although this analysis supports an unlimited number of open vent and drain lines from a dose consequence perspective, the licensees have proposed that the number of open vent and drain pathways be limited to six, to assure adequate administrative control for closing these pathways in the event of an accident.

In evaluating the proposed amendment, the staff performed independent offsite dose calculations using the same assumptions used to evaluate a postulated fuel handling accident as described in the Perry Safety Evaluation Report (NUREG-0887) dated May 1982, with the following exceptions:

1) A decay period of 7 days was assumed, following reactor shutdown, and,

2) All activity inside containment following the postulated accident was assumed to be immediately and directly released to the environment. This extremely conservative assumption does not take credit for containment, filtration or dilution of the release (and bounds the case of all 3/4-inch vent and drain lines being open).

The offsite doses computed for the Exclusion Area boundary (EAB) and Low Population Zone (LPZ) boundary using the above assumptions, assumptions contained in Regulatory Guide 1.25, and the procedures specified in SRP Section 15.7.4, are within the dose reference values of 10 CFR Part 100. The dose reference values of 10 CFR Part 100 are 300 rem to the thyroid and 25 rem to the whole body at EAB and LPZ. SRP Section 15.7.4 provides additional guidance by defining "well within" as 25 percent of the 10 CFR Part 100 dose reference values or 75 rem to the thyroid and 6 rem to the whole body at EAB and LPZ. The doses calculated were 47 rem to the thyroid and less than 1 rem to the whole body at EAB and 5.3 rem to the thyroid and less than 1 rem to the whole body at LPZ.

These values are in close agreement with the offsite dose calculations performed by the licensee (46 rem and 5 rem to the thyroid at the EAB and LPZ, respectively, and less than 1 rem whole body).

As discussed in Amendment 19 to the Perry License, Type C leak rate testing typically involves the following:

(a) Draining and refilling of the liquid from the test volume (between inboard and outboard manual isolation valves). Each of these operations will take less than 8 hours on the average. During these draining and refilling operations, a water seal will exist which would prevent (due to the lack of differential pressure) release of airborne radioactivity from the primary containment.

(b) Connection and disconnection of test apparatus to and from the vent valves. The potential for airborne radioactivity releases from the primary containment does exist through a vent valve, a drain valve, and two inboard and outboard containment isolation valves during the time intervals between (i) completion of test volume drain and connection of test apparatus, and (ii) disconnection of test apparatus and start of refill. At no time during the entire testing process are any open containment isolation valves disabled. Therefore, these valves would remain available if called upon to perform their isolation function. The plant operators would also be able to close any open automatic containment isolation valve from the main control room.

(c) Actual testing. The leak rate test itself will take less than 4 hours on the average. No airborne radioactivity release pathways exist during actual leak rate testing.

Any flow from the containment through the open vent and drain pathways would occur only if the inboard containment isolation valve is open. When utilizing the proposed change, the containment isolation valves would remain available if called upon to close. Additionally, administrative controls will ensure that the number of 3/4-inch vent and drainline pathways opened at any one time will be limited to six, control room operators will be aware of the openings, and test engineers will make reasonable attempts to isolate vent/drain lines prior to evacuating if evacuation is announced over the PA system. These administrative controls will ensure that timely action will be taken to close open vent and drain valves and the isolation valves in the event of a fuel handling accident.

The staff has determined, based on the operability of the containment isolation valves during the testing phase, administrative controls, the small size of the vent and drain lines and the low offsite dose consequences based on a 7-day decay period, that the proposed changes to the Perry Technical Specifications concerning the primary containment integrity requirements during fuel handling are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35 an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on September 28, 1990 (55 FR 39767). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 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, 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: J. R. Hall
J. Lee

Dated: September 28, 1990

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

THE CLEVELAND ELECTRICAL ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 35 to Facility Operating License No. NPF-58, issued to The Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company and Toledo Edison Company (the licensees), which revised the Technical Specifications for operation of the Perry Nuclear Power Plant, Unit No. 1, located in Lake County, Ohio. The amendment was effective as of the date of issuance.

The amendment modified the Technical Specifications to allow opening of up to six 3/4-inch vent and drain lines for Type C local leak rate testing during refueling activities, after the reactor has been subcritical for at least 7 days.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on April 5, 1990 (55 FR 12756). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated March 16, 1990, (2) Amendment No. 35 to License No. NPF-58, (3) the Commission's related Safety Evaluation dated September 28, 1990 and (4) the Environmental Assessment dated September 20, 1990. All of these items are available for public inspection at the Commission's Public Document Room, Gelman Building 2120 L Street N.W., Washington, D.C. and at the Perry Public Library, 3753 Main Street, Perry, Ohio 44081. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this 28th day of September 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation