March 31, 1989

Docket No. 50-440

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

Dear Mr. Kaplan:

SUBJECT: AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-58
PRIMARY CONTAINMENT INTEGRITY-SHUTDOWN (TAC NO. 71767)

The Commission has issued the enclosed Amendment No. 19 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 December 29, 1988.

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 one or two 3/4-inch vent and drain lines open on those penetrations that otherwise would not be testable when that specification is applicable. This amendment is applicable to the first refueling outage only.

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,

/s/

Timothy G. Colburn, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 19 to License No. NPF-58
2. Safety Evaluation
3. Notice of issuance

DF01
1/1

cc w/enclosures:
See next page

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Perry Nuclear Power Plant
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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. 19
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 December 29, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 19 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

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: March 31, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 19

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
xx	xx
3/4 6-2	3/4 6-2
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
----	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.

*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.

c. Whenever 1 or 2 vent and drain line pathways are open for the purpose of performing valve leak rate testing, verify primary containment to secondary containment differential pressure is within the limits of T.S 3.6.1.6 at least once per 12 hours.

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

#Except that two (2) 3/4" vent and drain line pathways may be opened for the purpose of performing containment isolation valve leak rate testing provided primary containment to secondary containment differential pressure is maintained within the limits of T.S. 3.6.1.6. This provision is in effect for the first refueling outage only.

3/4.6 CONTAINMENT SYSTEMS

BASES

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 two vent and drain line pathways for the purpose of performing containment isolation valve leak rate surveillance testing. Offsite doses have been calculated for the postulated fuel handling accident inside primary containment assuming a 100 cfm flowrate through the open vent and drain pathways. Although more than two vent/drain pathways could be open while remaining within the bounds of the analysis, the number of open pathways was limited to two to avoid the need for constant monitoring of the containment pressure through administrative controls. By adding the requirement to periodically check primary containment to secondary containment differential pressure while the 3/4" vent/drain pathways are open, there is assurance that sufficient primary containment to outside atmosphere differential pressure does not exist to create a 100 cfm flow through the open vent/drain pathways.

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 +0.1 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. During refueling conditions, this differential pressure is monitored when opening 3/4" vent/drain valves to perform leak rate testing per Technical Specification 3.6.1.1.2. See Bases for Technical Specification 3/4.6.1.1.

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. 19 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 December 29, 1988, the Cleveland Electric Illuminating Company (CEI), the licensee for Perry Nuclear Power Plant, Unit 1, proposed changes to plant technical specifications (TS) to revise the Primary Containment Integrity requirements during fuel handling. The proposed changes are requested to allow opening of one or two 3/4-inch vent and drain lines for Type C local leak rate testing (LLRT) of up to two penetrations during refueling activities. These changes were proposed to reduce the refueling outage duration, while still maintaining control to reestablish containment integrity.

The current Perry TS 3/4.6.1, requires the suspension of handling irradiated fuel in the primary containment if its integrity is not maintained. As a result, most Type C local leak rate surveillance testing required by 10 CFR Part 50, Appendix J, cannot be performed while refueling is in progress.

The licensee has performed an evaluation to show that opening of up to 50 3/4-inch vent and drain line pathways with a resulting flow of 100 CFM assuming 0.25-inch water gauge differential between the primary containment and the auxiliary building would not result in unacceptable offsite doses in the event of a fuel handling accident. Also, at no time during the testing process will the containment isolation valves be disabled. Therefore, the containment isolation function provided by the valves could be available if called upon to perform their isolation function.

2.0 EVALUATION

The staff has performed independent calculations which confirmed the licensee's calculated flow rate of 100 CFM based on 50 3/4-inch vent and drain line pathways open at an expected 0.25-inch water differential pressure between the primary containment and the auxiliary building. Although a pressure producing transient during refueling is not expected, the maximum containment pressure limited by plant TS Section 3.6.1.6 is less than 1 PSID between the primary containment and the auxiliary building. At 1 PSID, opening of four 3/4-inch vent and drain lines would result in a flow rate of 84 CFM.

The staff has reviewed the licensee's offsite dose rate calculations and concluded that the calculated offsite doses based on a 100 CFM flow rate through the vent and drain paths are acceptable assuming 15 days of decay time. This analysis is only valid for the first refueling. Each penetration into the primary containment is typically provided with vent and drain valves to facilitate Type C leak rate test for the inboard and outboard containment isolation valves.

In the Perry Safety Evaluation Report (NUREG-0887) dated May 1982, the staff previously evaluated a postulated fuel handling accident using assumptions contained in Positions C.1.a through C.1.k of Regulatory Guide 1.25 and the procedures specified in Standard Review Plan (SRP) Section 15.7.4 (NUREG-0800). In addition, the specified assumptions postulate a single dropped fuel assembly, the kinetic energy of which is expended with perfect mechanical efficiency in breaking open the maximum possible number of fuel rods. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods occurs as gas bubbles pass up through the water covering the fuel. All radioactivity reaching the primary containment atmosphere is exhausted within 2 hours.

In evaluating the proposed amendment, the staff performed the independent offsite dose calculations using the same assumptions previously used (including atmospheric diffusion parameters) for a postulated fuel handling accident at Perry with the following four exceptions:

(1) A decay period of 15 days was assumed (from February 22, 1989 to March 9, 1989).

(2) An unmitigated release of 100-cubic-feet per minute (CFM) occurred during the entire 30 days from the primary containment through up to two open vent and drain lines in addition to the maximum allowable unidentified primary containment leakage of 0.2 percent per day,

    The licensee stated in their proposal that the leak rate test procedure (Type C test) 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 drainings 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, the containment isolation function provided by 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) Leak rate 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.

(3) No credit is given for mixing of airborne radioactive material with primary containment atmosphere prior to release to the environment for the entire 30-day period after the accident. The staff assumed that the only air exhausted from the containment is the air directly above the pool (24 inches).

The licensee stated in the proposal that they conservatively assumed isolation of the Containment Vessel and Drywell Purge Systems (CVDPS's) to occur within 20 seconds after the accident. This isolation time includes the travel time of radioactive materials to the radiation monitor, receipt of isolation signal by the isolation damper, and the closing time of the damper (the current Perry Technical Specifications require the maximum damper closure time of 4 seconds in Table 3.6.4-1).

The staff assumed the CVDPS is in operation at the time of the accident at the design flow rate capacity of 10,000 CFM. Therefore, approximately 3,300 cubic feet of contaminated containment air is released during the first 20-second period after the accident. No filtration credit (removal efficiency) was given for the CVDPS for containment air that was released before containment isolation occurs, since the CVDP is not an engineered safety feature (ESF) system.

(4) After 20 seconds from the time of a postulated fuel handling accident, throughout the entire 30-day period, there will be two release pathways remaining from the primary containment to the environment; (i) unmitigated release of 100 CFM through up to two open vent and drain lines and (ii) the

maximum allowable unidentified primary containment leakage of 0.2 percent per day.

The Perry primary containment is a Mark III Pressure suppression containment with a free standing steel structure with a total free air volume of approximately 1.2×10^6 feet. The refueling floor is at elevation 682' and most of the penetrations subject to Type C local leak rate testing are located between elevations 599' and 666'. Therefore, in the event of a fuel handling accident, airborne radioactive material would have to travel a minimum of 16 feet downward through refueling floor open areas to reach the first possible open penetration to escape into the auxiliary or fuel building. However, the staff conservatively assumed that the 100 CFM air exhausted from the containment is the air directly above the pool (96' x 36' x 2') throughout the entire 30-day period.

Any unidentified leakage from the primary containment into the annulus building (0.2 percent per day) is collected and processed by the annulus exhaust gas treatment system (AEGTS). The AEGTS is a redundant engineered safety features system. Each AEGTS unit has a design capacity of 2,000 CFM and includes, among other things, one 4-inch deep carbon absorber and two high efficiency particulate air filters. The staff assumed 99 percent iodine removal efficiency by this ESF system. The staff finds that the offsite dose contribution from this pathway is negligible compared to that from the 100 CFM leakage pathway.

The offsite doses computed for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary using the above four 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 Plan 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 23 rem to the thyroid and less than 1 rem to the whole body at EAB and 3 rem to the thyroid and less than 1 rem to the whole body at LPZ. The staff, therefore, finds that the proposed Perry Unit 1 TS changes are acceptable on an interim basis for the first refueling only.

The 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, at no time during the testing process are the containment isolation valves disabled and, therefore, the containment isolation function provided by the 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 two, control room operators will be aware of the openings, pressure differential will be monitored, 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 size of vent and drain valves and the low offsite dose consequences based on a 15-day decay period, that the proposed changes to Perry Technical Specifications concerning the primary containment integrity shutdown during fuel handling is acceptable on an interim basis for the first refueling only. Final acceptance is contingent upon a favorable finding by the staff of the analysis without reliance on the 15-day decay time.

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 March 22, 1989 (54 FR 11830). 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 Contributors: R. Goel
J. Lee

Dated: March 31, 1989

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. 19 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 one or two 3/4-inch vent and drain lines for Type C local leak rate testing during refueling activities. This amendment is effective during the first refueling outage only.

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 Prior Hearing in connection with this action was published in the FEDERAL REGISTER on January 19, 1989 (54 FR 2246). 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 amendments dated December 29, 1988, (2) Amendment No. 19 to License No. NPF-58, (3) the Commission's related Safety Evaluation dated March 31, 1989 and (4) the Environmental Assessment dated March 15, 1989. All of these items are available for public inspection at the Commission's Public Document Room, Gelman Building, 2120 L Street NW, 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 31st day of March 1989.

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