June 26, 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. 22 TO FACILITY OPERATING LICENSE NO. NPF-58 (TAC NO. 68377)

The Commission has issued the enclosed Amendment No. 22 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 April 8, 1988 as amended July 21, 1988.

This amendment makes numerous editorial changes, typographical error corrections and title changes for personnel. In addition, various changes to the size and composition of the Plant Operations Review Committee (PORC) have been made.

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,

/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. 22 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

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Date:	6/7/89	6/9/89	6/9/89 6/26	6/13/89

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Perry Nuclear Power Plant
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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. 22
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 April 8, 1988, as amended July 21, 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. 22 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: June 26, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 22

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
xxvi	xxvi
1-7	1-7
3/4 6-5	3/4 6-5
3/4 7-6	3/4 7-6
3/4 7-6a	-----
3/4 7-17	3/4 7-17
3/4 8-5	3/4 8-5
3/4 8-6	3/4 8-6
6-7	6-7
6-8	6-8
6-9	6-9
6-10	6-10
6-14	6-14
6-15	6-15
6-16	6-16
6-17	6-17

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

1.34 The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71 and Federal and State regulations, burial ground requirements and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.35 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.36 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3579 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.37 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.38 A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

ROD DENSITY

1.39 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.40 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations terminating in the annulus and 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, as applicable secured in its closed position.

b. The containment equipment hatch is closed and sealed and the shield blocks are installed adjacent to the Shield Building.

c. The door in each access to the annulus is closed, except for normal entry and exit.

d. The sealing mechanism associated with each Shield Building penetration, e.g., welds, bellows or O-rings, is OPERABLE.

e. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.6.1.a., except for normal entry and exit to the annulus.

f. The Annulus Exhaust Gas Treatment System is in compliance with the requirements of Specification 3.6.6.2.

SHUTDOWN MARGIN

1.41 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.42 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOLIDIFICATION

1.43 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.44 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.45 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

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.
#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.

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.)

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the heaters dissipate 100 Kw ± 10% when tested in accordance with ANSI N510-1980.

4. Verifying that leakage through the outside air intake dampers (M25-F010A and M25-F020B for one train and M25-F010B and M25-F020A for the other train) is limited to less than 20 scfm.

5. Verify that leakage through the exhaust dampers M25-F130A and M25-F130B is limited to less than 20 scfm.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less thán 0.05% in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52 Revision 2, March 1978, while operating the system at a flow rate of 30000 scfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and testing acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52 Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30000 scfm ± 10%.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3* with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.
3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of greater than or equal to 700 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1020 + 25 - 100 psig (steam dome pressure).*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3/4.7.7 FUEL HANDLING BUILDING

FUEL HANDLING BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 At least three Fuel Handling Building (FHB) ventilation exhaust subsystems shall be OPERABLE.

APPLICABILITY: When irradiated fuel is being handled in the Fuel Handling Building.

ACTION:

With one FHB ventilation exhaust subsystem inoperable, restore the inoperable system to OPERABLE status within 7 days or suspend handling of irradiated fuel in the FHB. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each of the required FHB ventilation exhaust subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housing, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the subsystem flow rate is 15000 scfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1% when tested at a temperature of 30°C and a relative humidity of 70% in accordance with ASTM D3803; and

   3. Verifying a subsystem flow rate of 15000 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1% when tested at a temperature of 30°C and a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the fuel handling accident.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4.9 inches water gauge while operating the filter train at a flow rate of 15000 scfm ± 10%.

   3. Verifying that the filter train starts on manual initiation from the control room.

   4. Verifying that the heaters dissipate 50 kW ± 10% when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the subsystem at a flow rate of 15000 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the subsystem at a flow rate of 15000 scfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

7. Verifying the pressure in all air start receivers for each diesel generator to be greater than or equal to 210 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.

c. At least once per 92 days by checking for and removing accumulated water from the fuel oil storage tanks.

d. At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:

1) A water and sediment content of less than or equal to 0.05 volume percent;

2) A saybolt universal viscosity at 100°F of greater than or equal to 32.6 sus, but less than or equal to 40.1 sus;

3) An API gravity as specified by the manufacturer at 60°F of greater than or equal to 26 degrees, but less than or equal to 36 degrees;

4) An impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70; analysis shall be completed within 7 days after obtaining the sample but may be sampled and analyzed after the addition of new fuel oil; and

5) The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977; analysis shall be completed within 14 days after obtaining the sample but may be sampled and analyzed after the addition of new fuel oil.

e. At least once per 18 months*, ** during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with instructions prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to 1400 kw (LPCS pump) for diesel generator Div 1, greater than or equal to 729 kw (RHR B pump or RHR C pump) |

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* For any start of a diesel, the diesel must be loaded in accordance with the manufacturer's recommendations.

**Except 4.8.1.1.2.e.1 to be performed every refueling outage, for the Div 1 and Div 2 diesel generators.

for diesel generator Div 2, and greater than or equal to 2400 kw (HPCS pump) for diesel generator Div 3 while maintaining speed less than nominal speed plus 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 5800 kw for diesel generators Div 1 and Div 2 and 2600 kw for diesel generator Div 3 without tripping. The generator voltage shall not exceed 4784 volts for Div 1 and Div 2 and 5000 volts for Div 3 during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

   a) For divisions 1 and 2:

      1) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the load sequence (individual load timers) and operates for greater than or equal to 5 minutes while its generator is so loaded. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

   b) For division 3:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency bus with the permanently connected loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is so loaded. After energization, the steady

*All diesel generator starts for the purpose of this Surveillance Requirement may be preceded by an engine prelube period. The diesel generator start (10 sec)/load (60 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Director, Nuclear Engineering Department.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers or technically oriented individuals located onsite. Each shall have either (1) a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field, or (2) equivalent work experience as described in Section 4.1 of ANSI/ANS 3.1, December 1981.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Director, Nuclear Engineering Department.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Plant Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Perry Training Section Manager, and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the General Manager, Perry Plant Operations Department (PPOD) and the Director, Perry Plant Technical Department (PPTD), on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Director, Perry Plant Technical Department
Vice-Chairman/Member:	Manager, Operations Section
Member:	Manager, Technical Section
Member:	Manager, Maintenance Section
Member:	Reactor Engineer
Member:	Manager, Radiation Protection Section
Member:	Plant Health Physicist
Member:	Manager, Instrumentation and Control Section
Member:	Manager, Licensing and Compliance Section

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of all Administrative Procedures;

b. Review of the safety evaluations for (1) proposed procedures/instructions, (2) changes to procedures/instructions, equipment, systems or facilities, and (3) tests or experiments performed under the provisions of 10 CFR 50.59 to verify that such actions do not constitute an unreviewed safety question;

c. Review of proposed procedures/instructions and changes to procedures/instructions, equipment, systems or facilities which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Review of proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

e. Review of proposed changes to Technical Specifications or the Operating License;

f. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Group and to the Nuclear Safety Review Committee;

g. Review of all REPORTABLE EVENTS;

h. Review of the plant Security Plan and Security Contingency Instructions;

i. Review of the Emergency Plan and implementing instructions;

j. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems;

k. Review of any accidental, unplanned or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the General Manager, Perry Plant Operations Department (PPOD) and the Director, Perry Plant Technical Department (PPTD), the Nuclear Safety Review Committee and the Vice President - Nuclear Group;

l. Review of Unit operations to detect potential hazards to nuclear safety;

m. Investigations or analysis of special subjects as requested by the Chairman of the Nuclear Safety Review Committee; and

n. Review of the Fire Protection Program and implementing procedures.

RESPONSIBILITIES (Continued)

6.5.1.7 The PORC shall:

a. Recommend in writing to the General Manager, PPOD/Director, PPTD, approval or disapproval of items considered under Specifications 6.5.1.6a. through e., h., i., j., and k., above prior to their implementation;

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6b. through e., above, constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Vice President - Nuclear Group and the Nuclear Safety Review Committee of disagreement between the PORC and the General Manager, Perry Plant Operations Department; however, the General Manager, Perry Plant Operations Department, shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President - Nuclear Group and the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The NSRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering,

h. Quality assurance practices and administrative controls, and

i. Nondestructive testing.

The NSRC shall report to and advise the Vice President - Nuclear Group on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

AUDITS (Continued)

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified corporate licensee fire protection engineer(s) or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least every third year;

g. The radiological environmental monitoring program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

k. Any other area of unit operation considered appropriate by the NSRC or the Vice President - Nuclear Group.

RECORDS

6.5.2.9 Records of NSRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved, and forwarded to the Vice President - Nuclear Group within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Vice President - Nuclear Group within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Vice President - Nuclear Group and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures/instructions required by Specification 6.8 and other procedures/instructions which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure/instruction or procedure/instruction change shall be reviewed by a qualified individual(s) other than the individual(s) which prepared the procedure/instruction or procedure/instruction change, but who may be from the same section as the individual(s) which prepared the procedure/instruction or procedure/instruction change. Instructions shall be approved by appropriate management personnel as designated in writing by PORC, and approved by the appropriate managers, Perry Plant Departments. Both the General Manager, PPOD, and the Director, PPTD, shall approve Administrative Procedures.

b. Proposed modifications to plant structures, systems and components that affect nuclear safety shall be reviewed by individuals designated by the Director, Nuclear Engineering Department. Each such modification shall be reviewed by a qualified individual(s) other than the individual(s) which designed the modification, but who may be from the same section as the individual(s) which designed the modifications. Proposed modifications to plant structures, systems and components that affect nuclear safety shall be reviewed by PORC and approved prior to implementation by both the General Manager, PPOD/Director, PPTD.

c. Proposed tests and experiments which affect plant nuclear safety shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by a qualified individual(s) other than the individual(s) which prepared the proposed test or experiment. Proposed tests and experiments shall be approved before implementation by both the General Manager, PPOD/Director PPTD.

d. Sections responsible for reviews, including cross-disciplinary reviews, performed in accordance with Specifications 6.5.3.1a. and 6.5.3.1c., shall be designated in writing by PORC and approved by the General Manager, PPOD, or the Director, PPTD, as appropriate. The individual(s) performing the review shall meet or exceed the qualification requirements of appropriate section(s) of ANSI N18.1-1971;

e. Each review shall include a determination pursuant to 10 CFR 50.59 of whether or not the potential for an unreviewed safety question exists. If such a potential does exist, a safety evaluation per 10 CFR 50.59 to determine whether or not an unreviewed safety question is involved shall be performed. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions shall be obtained prior to implementation; and

ACTIVITIES (Continued)

f. The Plant Security Plan and Emergency Plan, and implementing instructions, shall be reviewed at least once per 12 months. Recommended changes to the implementing instructions shall be approved by the Director, Nuclear Support Department, or Director, Perry Plant Technical Department, as appropriate. Recommended changes to the Plans shall be reviewed pursuant to the requirements of Specification 6.5.1.6 and approved by the General Manager, Perry Plant Operations Department, and either the Director, Nuclear Support Department, or the Director, Perry Plant Technical Department, as appropriate. NRC approval shall be obtained as appropriate.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of the review submitted to the NSRC and the Vice President - Nuclear Group.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear Group and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC, and the Vice President - Nuclear Group within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS

6.8.1 Written procedures/instructions shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS (Continued)

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Security Plan implementation.

d. Emergency Plan implementation.

e. PROCESS CONTROL PROGRAM implementation.

f. OFFSITE DOSE CALCULATION MANUAL implementation.

g. Radiological Environmental Monitoring Program implementation.

h. Fire Protection Program implementation.

6.8.2 Each administrative procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the PORC and shall be approved by the General Manager, PPOD, and the Director, PPTD, prior to implementation. All procedures/instructions shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes. Temporary changes to procedures/instructions which do not change the intent of the approved procedures/instructions shall be approved for implementation by two members of the plant management staff, at least one of whom holds a Senior Operator license. These temporary changes shall be documented. The temporary changes shall be approved by the original approval authority within 14 days. For changes to procedures/instructions which may involve a change in intent of the procedures/instructions, the original approval authority shall approve the change prior to implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the HPCS, RHR, RCIC, LPCS, feedwater leakage control system, the hydrogen analyzer portion of Combustible Gas Control, and post-accident sampling systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report Subsection 14.2.12.2 and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions; and

b. Documentation of all challenges to safety/relief valves.

c. Annual reports shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 22 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 April 8, 1988, as amended July 21, 1988, the Cleveland Electric Illuminating Company (CEI), et al., the licensees, submitted a proposed amendment to Facility Operating License No. NPF-58 for Perry Nuclear Power Plant, Unit 1. The proposed amendment included a reorganization of the Plant Operations Review Committee (PORC), Section 6.5.1 and numerous other administrative changes to the Technical Specifications (TS).

Partial response to the licensees' April 8, 1988 application was contained in Amendment No. 13 to Facility Operating License NPF-58 for the Perry Nuclear Power Plant, Unit 1, issued on June 30, 1988. This amendment addresses the remaining items from the April 8, 1988 application as amended on July 21, 1988.

A Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity For Hearing related to the requested action was published in the Federal Register on May 31, 1988 (53 FR 19832) and again on March 22, 1989 (54 FR 11845). No requests for hearing and no public comments were received.

2.0 DISCUSSION AND EVALUATION

The following are descriptions and evaluations of each of the proposed changes to TS for Perry Nuclear Power Plant, Unit 1. None of these changes involved physical modifications to the facility.

Description of Change, Technical Specifications Pages 1-7 and 6-15

The proposed change would revise the definition of a REPORTABLE EVENT by deleting the reference to 10 CFR 50.72 immediate notification requirements.

Evaluation

The modified definition would be consistent with the recommendations of Generic Letter 83-43, dated December 19, 1983 which specified only conditions identified in 10 CFR 50.73 shall be defined as a reportable event. Since the events described in 10 CFR 50.72 are included in 10 CFR 50.73, the staff has determined that the proposed change is acceptable.

Description of Change, Technical Specification Page 3/4 6-5

The proposed change would delete the footnote accompanying surveillance requirement 4.6.1.2.f which extended the leak testing of the main steam line isolation valves.

Evaluation

This change is administrative in nature. The footnote was added by Amendment 5 to the Facility Operating License to extend the leak testing of main steam line isolation valves 1B21-F022A and 1B21-F028A until July 12, 1987. The footnote has expired and is no longer applicable; therefore the staff has determined that the change is acceptable.

Description of Change, Technical Specification 3.7.3

The proposed change would delete the footnote referenced in TS 3.7.3 which suspended the operability requirement for automatic initiation of the reactor core isolation cooling (RCIC) system for the period April 10, 1987 through May 31, 1987.

Evaluation

The original request to suspend the RCIC operability requirements was granted through Amendments 1 and 4 of the Facility Operating License. The suspension expired on May 31, 1987 and is no longer applicable. The deletion of the footnote is acceptable to the staff.

Description of Change, Technical Specification, Section 4.7.7.1

The proposed change would correct the word "system" to "subsystem" in TS 4.7.7.1.b.1.

Evaluation

The word replacement corrects a typographical error and is administrative in nature. The staff has determined that the change is acceptable.

Description of Change, Technical Specification 4.8.1.1.2

The proposed change corrects the values corresponding to the electrical loads of one residual heat removal (RHR) and one high pressure core spray (HPCS) pump on their associated emergency diesel generators. The values would be changed from 725kw to 729kw and from 2200kw to 2400kw, respectively.

Evaluation

The proposed load values reflect actual loads listed in Table 8.2-1 in the Updated Safety Analysis Report. These correct previous load data associated with the RHR and HPCS pumps. The proposed changes are also conservative in nature. The staff has determined that the changes are acceptable.

Description of Change, Technical Specification Section 6

The following changes are proposed:

- Section 6.3.1 would be changed to delete the exception to ANSI N18.1-1971 qualification requirements for the Senior Operations Coordinator.

- The title, Manager, Nuclear Engineering Department would be changed to Director, Nuclear Engineering Department in Sections 6.2.3 and 6.5.3.

- The Manager, Perry Plant Operations Department (PPOD) position would be renamed General Manager, PPOD in Sections 6.1, 6.2, and 6.5,

- The titles for all General Supervisors and General Supervising Engineers would be changed to Managers in Sections 6.4 and 6.5.

- In Sections 6.5.1.1, 6.5.1.6, 6.5.2.7, 6.5.3.1, and 6.8.2, the reference to Managers, Perry Plant Departments would be revised to General Manager, PPOD and Director, PPTD for clarification.

- The requirements for approval of temporary procedure changes would be moved from Section 6.5.3.1.a to a new section 6.8.3. The old Section 6.8.3 was renumbered as 6.8.4.

- In Section 6.8.4.a, the "CS" would be deleted and "hydrogen analyzer portion of Combustible Gas Control" would be added to clarify the portion of the CS system included in the Primary Coolant Sources Outside Containment Program.

- Due to a recent plant reorganization, the Site Protection Section now reports to the newly-created Nuclear Support Department (NSD) via the Director, Perry Plant Technical Department. Specification 6.5.3.1.f has been revised to indicate that Security Plan changes and Security Plan implementing instructions will now be approved by the Director, Nuclear Support Department, or Director, Perry Plant Technical Department, as appropriate.

Evaluation

All of these changes are administrative in nature. On Page 6-7, Specification 6.3.1 is being revised to delete the exception to ANSI N18.1-1971 qualification requirements for the Senior Operations Coordinator since these qualifications are now being met. Personnel title changes are being made to be consistent with the current plant organization titles. Moving the requirements for approval of temporary procedure changes from Section 6.5.3.1.a to 6.8.3 does not change the review process or the approval requirements. These proposed changes are considered acceptable by the staff.

Description of Change, Technical Specification Section 6.5.1

The proposed changes modify the composition and responsibility of the Plant Operations Review Committee (PORC). The General Manager, PPOD and the Technical Supervisor, PPTD have been removed from serving on the committee. The Director, PPTD and the Manager, Operations Section have been assigned as the Chairman and Vice Chairman, respectively. The Principal Nuclear Operations Engineer and General Supervising Engineer, Outage Planning Section, have been removed which eliminates an additional two positions in PORC. The definition of Quorum in 6.5.1.5 has been changed to require the Chairman or the designated alternate and at least four members present to correspond to the reduced size of the committee. The total committee size is reduced from 13 members to 9. Section 6.5.1.3 was revised to reduce the number of allowable participating alternates from three to two.

Section 6.5.1.6 h, i and n would be revised to delete the requirement to submit to the Nuclear Safety Review Committee a report recommended changes of the Security Plan, Emergency Plan, and Fire Protection Program.

Evaluation

The original PORC board consisted of the General Manager, PPOD as chairman, three vice chairman and nine regular members. A quorum was defined as the chairman and six members which included no more than three alternates. The proposed reorganization will reassign the Director, PPTD and the Manager, Operations Section to the positions of chairman and vice chairman respectively. The removal of the Manager, PPOD, Technical Superintendent, PPTD, Principal Nuclear Operations Engineer and General Supervisor, Engineer Outage Planning Section will reduce the board to nine permanent members. A quorum will require the presence of the chairman and at least four members with no more than two alternates. The ratio of permanent members to alternates is essentially unchanged, in that the majority of personnel attending a PORC meeting will always be permanent members.

The changes in Section 6.5.1.6 are administrative in nature. Deleting these reporting requirements will make this section consistent with the responsibilities of the Nuclear Safety Review Committee specified in TS 6.5.2.7. The review and approval of the PORC recommendations of the Security Plan, Emergency Plan and Fire Protection Program are the responsibilities of the General Manager, PPOD and/or Directors, PPTD or NSD as required by TS 6.5.3.1.

The reorganization of the PORC will not interfere with its intended function. The PORC members advise the General Manager, PPOD on matters related to nuclear safety. This individual is being removed from the PORC since it is felt that the role of the PORC could be better performed if this position were independent of the PORC. The Principal Nuclear Operations Engineer is being removed from the PORC since he is also functioning as the Nuclear Safety Review Committee Chairman. The General Supervising Engineer, Outage Planning Section, is being removed because the position is not required. The Technical Superintendent, PPTD, is being removed since the Director, PPTD, will now serve as chairman. The remaining PORC members have an adequate knowledge of the various aspects of nuclear power plan operations. The ANSI N18.7-1976 endorsed by Regulatory Guide 1.33 and Chapter 13.4 of the Standard Review Plan describe the general characteristics and responsibilities of the operations review committee. The proposed changes do not conflict with these standards, and are therefore considered acceptable by the staff.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment relates to changes in surveillance requirements and changes in recordkeeping, reporting or administrative procedures or requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusions set forth in 10 CFR 51.22(c)(9) and (10). 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

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.

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Dated: June 26, 1989