May 28, 1993

Docket No. 50-440

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Mr. Robert A. Stratman, Vice President
Nuclear - Perry
The Cleveland Electric Illuminating
Company
10 Center Road
Perry, Ohio 44081

Dear Mr. Stratman:

SUBJECT: AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. NPF-58 (TAC NO. M84012)

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications (TSs) in response to your application dated June 30, 1992.

This amendment revises Technical Specifications 3.3.1, "Reactor Protection System Instrumentation," and 6.9.1.9, "Core Operating Limits Report (COLR)," to transfer the simulated thermal power time constant from the TSs to the COLR.

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,

Original signed by Jon Hopkins

for Robert J. Stransky, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 48 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD3-3/DRPW	PM/PD3-3/DRPW	PD/PD3-3/DRPW	OGC-WF1
PKreutzer	RStransky/rjs/bj	JHannon	
3/26/93	3/22/93 JBH for 5/28/93	3/26/93	5/24/93

DOCUMENT NAME: G:\PERRY\PER84012.AMD

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

May 28, 1993

Docket No. 50-440

Mr. Robert A. Stratman, Vice President
Nuclear - Perry
The Cleveland Electric Illuminating
Company
10 Center Road
Perry, Ohio 44081

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Sincerely,

for Jon B. Hopkins

Robert J. Stransky, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 48 to
   License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Robert A. Stratman
Cleveland Electric Illuminating Company

Perry Nuclear Power Plant
Unit Nos. 1 and 2

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Perry, Ohio 44081

The Honorable Robert V. Orosz
Mayor, Village of North Perry
North Perry Village Hall
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North Perry Village, Ohio 44081

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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
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, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated June 30, 1992, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 48 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

for Jon B. Hopkins

Robert J. Stransky, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: May 28, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 48

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
B 2-7	B 2-7
3/4 3-6	3/4 3-6
3/4 3-8	3/4 3-8
6-21	6-21

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-High setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-High setpoint, a time constant specified in the COLR is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine control valve fast closure and turbine stop valve closure trips are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Reactor Vessel Water Level-High

A reactor scram from high reactor water level, approximately two feet above normal operating level, is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. An excess of feedwater entering the vessel would be detected by the level increase in a timely manner. This scram feature is only effective when the reactor mode switch is in the Run position because at THERMAL POWER levels below 10% to 15% of RATED THERMAL POWER, the approximate range of power level for changing to the Run position, the safety margins are more than adequate without a reactor scram.

6. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

7. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤ 0.09**
c. Neutron Flux - High	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.35
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Reactor Vessel Water Level - High, Level 8	≤ 1.05
6. Main Steam Line Isolation Valve - Closure	≤ 0.06
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	NA
10. Turbine Stop Valve - Closure	≤ 0.06
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤ 0.07#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including the simulated thermal power time constant specified in the COLR.

#Measured from start of turbine control valve fast closure.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1 and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.9.1, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) This function is automatically bypassed when turbine first stage pressure is less than the value of turbine first stage pressure corresponding to 40%** of RATED THERMAL POWER.

---

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**The Turbine First Stage Pressure Bypass Setpoints and corresponding Allowable Values are adjusted based on Feedwater temperatures (see 3/4.2.2 for definition of ΔT). The Setpoints and Allowable Values for various ΔTs are as follows:

T(°F)	Setpoint (psig)	Allowable Value (psig)
$0 = T$	≤ 212	≤ 218
$0 < \Delta T \le 50$	≤ 190	≤ 196
$50 < \Delta T \le 100$	≤ 168	≤ 174
$100 < \Delta T \le 170$	≤ 146	≤ 152

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
10. Turbine Stop Valve - Closure	NA	M	R	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	M	R	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Deleted
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reaching 25% of RATED THERMAL POWER.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 MWD/T using the TIP system.
(g) Calibrate trip unit setpoint at least once per 31 days.
(h) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow).
(i) This calibration shall consist of verifying that the simulated thermal power time constant is within the limits specified in the COLR.
(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(l) This function is not required to be OPERABLE when Drywell Integrity is not required.
(m) The CHANNEL CALIBRATION shall exclude the flow reference transmitters, these transmitters shall be calibrated at least once per 18 months.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b) S	W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U,S,(b) S	W W	SA SA	2 3, 5
b. Flow Biased Simulated Thermal Power - High	S,D(h)	W	W(d)(e), SA(m), R(i)	1
c. Neutron Flux - High	S	W	W(d), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R(g)	1, 2(j)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R(g)	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R(g)	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2(j)
8. Drywell Pressure - High	S	M	R(g)	1, 2(l)
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R(g)	1, 2, 5(k)
b. Float Switches	NA	M	R	1, 2, 5(k)

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively; and description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Resource Management, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

(2) The Minimum Critical Power Ratio (MCPR) for Technical Specification 3.2.2.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.3.

(4) The Simulated Thermal Power Time Constant for Technical Specification 3.3.1.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel. (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report.

6.9.3 Safety-relief valve failures will be reported to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report system within 30 days.

6.9.4 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report system within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering the interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 48 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 June 30, 1992, the Cleveland Electric Illuminating Company, et al. (licensees), proposed changes to the Technical Specifications (TSs) for the Perry Nuclear Power Plant, Unit No. 1. The proposed changes would modify TSs 3.3.1, "Reactor Protection System Instrumentation," and 6.9.1.9, "Core Operating Limits Report (COLR)," to transfer the specific value of the simulated thermal power time constant from the Technical Specifications to the COLR. These changes are being made in accordance with guidance provided in NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," and supplement similar changes to the Technical Specifications made in Amendment No. 33.

2.0 EVALUATION

In Amendment No. 33, the Commission approved the licensee's transfer of certain cycle-specific limits from the Technical Specifications to a Core Operating Limits Report. The licensee's establishment of the COLR included appropriate Technical Specification references to the COLR, as well as requirements for the submittal of the COLR to the NRC prior to operation with the new parameter limits. The currently proposed changes to the Technical Specifications would add the simulated thermal power time constant to the list of values included in the COLR.

Advances in fuel designs have led to the construction of fuel bundles with more individual fuel rods located within the same cross-sectional area. In order to place additional fuel rods into the same area, the diameter of the fuel rods and fuel pellets must be decreased. The decrease in fuel pellet diameter reduces the simulated thermal power time constant, which is related to the amount of time needed for heat to travel from the center of the fuel pellet to the outer edge.

The Perry Nuclear Power Plant currently uses General Electric 8x8 array fuel bundles, although the licensee is contemplating the use of 9x9 or 10x10 array fuel bundles in upcoming cycles. Since the simulated thermal power time constant is different for the various fuel designs, the use of new fuel

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designs in the future would necessitate changes to the simulated thermal power time constant on a cycle-specific basis. Therefore, the staff finds that the simulated thermal power time constant is a cycle-specific limit, and is eligible for relocation to the COLR.

The licensee proposes to make several changes to the TS in order to transfer the simulated thermal power time constant to the COLR. The changes would modify (1) the bases for the Reactor Protection System instrumentation setpoints, (2) the footnote to TS Table 3.3.1-2, and (3) the footnote to TS Table 4.3.1.1-1, to reference the simulated thermal power time constant as located in the COLR. The amendment would also change TS 6.9.1.9, "Core Operating Limits Report," to specify that the COLR will contain the Simulated Thermal Power Time Constant. No other changes to the TSs are made by this amendment.

On the basis of information previously reviewed by the Commission in the Safety Evaluation for Amendment No. 33, the staff found that the licensee's implementation of the COLR adequately addressed the guidance provided in NRC Generic Letter 88-16. The currently proposed changes to the Technical Specifications are also consistent with the intent of GL 88-16. Furthermore, because plant operation will continue to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 37561). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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: R. Stransky

Date: May 28, 1993