February 9, 1989

Docket No.: 50-352

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Mr. George A. Hunger, Jr.
Director-Licensing
Philadelphia Electric Company
Correspondence Control Desk
P. O. Box 7520
Philadelphia, Pennsylvania 19101

Dear Mr. Hunger:

SUBJECT: LPCI INJECTION VALVE DIFFERENTIAL PRESSURE INSTRUMENT LOOPS (TAC NO. 64260)

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 5, 1986, as supplemented by your letter of November 5, 1987.

This amendment revises the Technical Specifications (TSs) relating to the trip setpoints and allowable values for the low pressure coolant injection valve differential pressure instrument loops. Issuance of this amendment closes out item 6.4 in Inspection Report 50-352/87-09.

One of the three key entries in our Safety Issues Management System (SIMS) is the Licensee Implementation Date. We request that you advise us when Modification No. 85-0522 is implemented.

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

Sincerely,
Original signed by
Richard J. Clark
Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 16 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

[HUNGER]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
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See next page

[HUNGER]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:tr	RBachmann	WButler
/ /89	01/24/89	1/27/89	/ /89

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

February 9, 1989

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Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 16 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Troy B. Conner, Jr., Esquire
Conner and Wetterhahn
1747 Pennsylvania Ave., N.W.
Washington, D. C. 20006

Mr. Rod Krich S7-1
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. David Honan N2-1
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. Graham M. Leitch, Vice President
Limerick Generating Station
Post Office Box A
Sanatoga, Pennsylvania 19464

Mr. James Linville
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Thomas Kenny
Senior Resident Inspector
US Nuclear Regulatory Commission
P. O. Box 596
Pottstown, Pennsylvania 19464

Mr. Joseph W. Gallagher
Vice President, Nuclear Services
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. John S. Kemper
Senior Vice President-Nuclear
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. Robert Gramm
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 596
Pottstown, Pennsylvania 19464

Mr. Ted Ullrich
Manager - Unit 2 Startup
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. John Doering
Superintendent-Operations
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Thomas Gerusky, Director
Bureau of Radiation Protection
PA Dept. of Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Single Point of Contact
P. O. Box 11880
Harrisburg, Pennsylvania 17108-1880

Mr. Philip J. Duca
Superintendent-Technical
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated November 5, 1986, as supplemented by letter dated November 5, 1987, 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-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 16 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective 30 days after date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 9, 1989

Previously concurred*

PDI-2/LA	PDI-2/PM*	OGC*	PDI-2/D*	WB
MO'Brien	RClark:tr	RBachmann	WButler	
2/4/89	01/24/89	01/29/89	2/8/89	

3. This license amendment is effective ~~as of its date of~~ 30 days after date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:tr	RBachmann	WButler
/ /89	01/24/89	1/29/89	/ /89

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FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 9, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change. The overleaf page is provided to maintain document completeness.*

Remove	Insert
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38*

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure - Low	≥ 455 psig,(decreasing)	≥ 435 psig, (decreasing)
d. Manual Initiation	N.A.	N.A.
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure - Low	≥ 455 psig,(decreasing)	≥ 435 psig, (decreasing)
d. Injection Valve Differential Pressure - Low	≥ 74 psid, (decreasing)	≥ 64 psid and ≤ 84 psid
e. Manual Initiation	N.A.	N.A.
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - (Low Low, Level 2)	≥ -38 inches*	≥ -45 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Condensate Storage Tank Level - Low	≥ 167.8 inches**	≥ 164.3 inches
d. Suppression Pool Water Level - High	≤ 24 feet 1.5 inches	≤ 24 feet 3 inches
e. Reactor Vessel Water Level - High, Level 8	≤ 54 inches	≤ 60 inches
f. Manual Initiation	N.A.	N.A.
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Reactor Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. ADS Timer	≤ 105 seconds	≤ 117 seconds
d. Core Spray Pump Discharge Pressure - High	≥ 145 psig,(increasing)	≥ 125 psig, (increasing),
e. RHR LPCI Mode Pump Discharge Pressure-High	≥ 125 psig,(increasing)	≥ 115 psig, (increasing)
f. Reactor Vessel Water Level-Low, Level 3	≥ 12.5 inches	≥ 11.0 inches
g. Manual Initiation	N.A.	N.A.
h. ADS Drywell Pressure Bypass Timer	≤ 420 seconds	≤ 450 seconds

*See Bases Figure B 3/4.3-1.

**Corresponds to 2.25 feet indicated.

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT		ALLOWABLE VALUE
5. LOSS OF POWER	RELAY			
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	127-11X	NA		NA
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	RELAY			
	127-11X0X and 102-11X0X	a.	4.16 kV Basis 2905 ± 115 volts	2905 ± 145 volts
		b.	120 V Basis 83 ± 3 volts	83 ± 4 volts
		c.	≤ 1 second time delay	≤ 1.5 second time delay
	127Y-11X0X** and 127Y-1-11X0X	a.	4.16 kV Basis 3640 ± 91 volts	3640 ± 182 volts
		b.	120 V Basis 104 ± 3 volts	104 ± 5.2 volts
		c.	≤ 52 second time delay	≤ 60 second time delay
	127Z-11X0X and 162Y-11X0X	a.	4.16 kV Basis 3745 ± 94 volts	3745 ± 187 volts
		b.	120 V Basis 107 ± 3 volts	107 ± 5.4 volts
		c.	≤ 10 second time delay	≤ 11 second time delay
	127Z-11X0X and 162Z-11X0X	a.	4.16 kV Basis 3745 ± 94 volts	3745 ± 187 volts
		b.	120 V Basis 107 ± 3 volts	107 ± 5.4 volts
		c.	≤ 61 second time delay	≤ 64 second time delay

*This is an inverse time delay voltage relay. The voltages shown are the maximum that will not result in a trip. Some voltage conditions will result in decreased trip times.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NO. NPF-39

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1.0 INTRODUCTION

By letter dated November 5, 1986, as supplemented by letter dated November 5, 1987, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. The proposed amendment would revise the Technical Specifications (TSs) relating to the trip setpoints and allowable values for the low pressure coolant injection (LPCI) valve differential pressure instrument loops. The Trip Setpoint (SP) and Allowable Values (AV) are being revised as a result of the licensee's proposed increase in the instrument loop range and corresponding increase in specified allocations for instrument loop accuracy and calibration accuracy.

2.0 DISCUSSION

The LPCI injection valve differential pressure instrument loops are provided to protect the low pressure Residual Heat Removal (RHR) piping from high reactor pressure by interlocking the LPCI injection valves so that they cannot be opened if the pressure on the reactor side of the injection valve is greater by a given amount than the pressure in the RHR pump discharge piping. The differential pressure setpoint is established so that the injection valves are permitted to open in sufficient time to establish LPCI flow into the reactor to satisfy reactor water level requirements during a design basis Loss of Coolant Accident (LOCA), but not until reactor pressure has decreased to the allowable rated pressure for the low pressure RHR piping. This interlock, which prevents the motor operated LPCI injection valves from opening until the differential pressure across the valves is below a specified value, has been evaluated in the U.S. Nuclear Regulatory Commission's Safety Evaluation Report for the Limerick Generating Station, Units 1 and 2, NUREG-0991, (page 7-47, August 1983).

The instrument loops are presently calibrated to monitor a range of 0 to 800 psid. The specified allocations for instrument loop accuracy and calibration accuracy are 6 psid and 2 psid, respectively. The upper analytical limit for the differential pressure across the valves is 95 psid. The trip units are calibrated for a Trip Setpoint of $\geq$ 78 psid (decreasing) and Allowable Values of $\geq$ 68 psid and $\leq$ 88 psid as denoted in TS Table 3.3.3.2, Item 2d. The RHR pump discharge piping is normally pressurized to a nominal 125 psig via the condensate transfer system. The licensee has stated that during normal plant shutdowns, reactor pressure is reduced to below the RHR pump discharge piping pressure and the measured differential pressure relative to the RHR discharge piping becomes negative. Under these circumstances, the differential pressure is no longer within the present calibrated range of the instrument loop (0 to 800 psid), causing the transmitter output to go below the normal minimum of 4 milliamperes (mA). Eventually, the measured differential pressure will become sufficiently negative to drive the transmitter output to a value which causes the associated trip unit to sense a low signal/gross failure condition, thus actuating a RHR loop out-of-service annunciation in the control room.

This false out-of-service annunciation can be corrected by increasing the range of the differential pressure instrument loop to - 200 to 800 psid. With this range, the transmitter output will remain above 4 mA as long as the sensed differential pressure remains above - 200 psid. The new range is sufficient to envelope the expected differential pressures during plant shutdowns, thereby eliminating the conditions which presently cause the trip units to sense a low signal/gross failure condition.

As a result of increasing the instrument loop range, the specified allocations for instrument loop accuracy and calibration accuracy must be increased to 10 psid and 3 psid, respectively, thereby requiring a change to Table 3.3.3-2 for the instrument loop SP and AV. The licensee has proposed a revised Trip Setpoint of $\geq$ 74 psid (decreasing) and Allowable Values of $\geq$ 64 psid and $\leq$ 84 psid consistent with the increase in calibration range of - 200 to 800 psid for the LPCI injection valve differential pressure loop.

3.0 EVALUATION

There are two safety functions associated with the differential pressure setpoint of the LPCI injection valve differential pressure instrument loops:

a. to protect the low pressure RHR piping from high reactor pressure by interlocking the LPCI injection valves to prevent them from opening when reactor pressure is greater than the allowable pressure for the low pressure RHR piping.

b. to enable the LPCI injection valves to open in sufficient time to establish LPCI flow into the reactor in order to satisfy reactor water level requirements during a design basis LOCA.

The LPCI injection valve differential pressure interlock is designed to prevent opening the LPCI injection valves during all normal operational and accident event scenarios when the reactor pressure is above the maximum design pressure of the low pressure RHR piping. The maximum design pressure is based upon the most limiting component in the RHR piping system. The licensee states that this limiting component is the RHR heat exchanger which corresponds to a differential pressure of 106.2 psid across the injection valve. The licensee's original design maintains a margin of 11.2 psid to the maximum design pressure providing for an upper analytical limit of 95 psid. The licensee has determined that an increase in calibration range (-200 to 800 psid) for the LPCI valve differential pressure loop would result in only a small increase (4 psid) in the instrument loop accuracy, corresponding to a decrease in the upper Allowable Value from 88 psid to 84 psid and in the Trip Setpoint from 78 psid to 74 psid. The upper analytical limit of 95 psid, established in the original design, remains unchanged. Based on the staff's review of the licensee's proposed request as it relates to protecting the RHR pump discharge piping from high reactor pressure we concluded that the existing level of overpressure protection is not reduced. With respect to the first safety function discussed above, the proposed modification to the calibration range and the proposed changes to the TSs are acceptable.

The LPCI injection valve differential pressure interlock is also designed to enable the LPCI injection valves to open in sufficient time to establish LPCI flow into the reactor during a postulated LOCA. The licensee states that the corresponding ECCS analysis assumes the differential pressure interlock is cleared and the injection valves are signaled to open when the reactor steam dome pressure drops to 300 psig. The staff has reviewed the licensee's comparison of current and proposed setpoint data as it relates to the ECCS analysis. The licensee has determined that an increase in calibration range (-200 to 800 psid) would decrease the lower Allowable Value from 68 to 64 psid, corresponding to a reactor steam dome pressure of 368.9 psid. As discussed below, the licensee has determined that the total change in margin is only 5.3 psid, out of total pressure margins of approximately 80 psi in analysis input assumptions.

The licensing basis ECCS analysis, which is described in FSAR Section 6.3.3, was performed using the SAFE code for the LOCA pipe rupture events described in FSAR, Section 15.2.8, 15.6.4, and 15.6.5. The limiting event is the postulated guillotine break of the reactor recirculation system suction line. In the ECCS analysis it is assumed that the interlock is cleared and the injection valve receives a signal to begin to open when the reactor steam dome pressure drops to 300 psig. Further, the analysis conservatively assumes that flow through the valve does not occur until the valve is fully open (a 26 second delay time after the interlock is cleared and the valve is signaled to open).

The interlock design analysis has been performed using a proposed Lower Analytical Limit (LAL) of 53 psid. The Analytical Limit is the value of the sensed process variable prior to or at the point which a desired action is to be initiated to prevent the process variable from reaching the associated safety limit. The value of 53 psid corresponds to a reactor steam dome pressure of 368.9 psig. This reactor steam dome pressure of 368.9 psig provides a margin of 68.9 psi above the value of reactor steam dome pressure of 300 psig, at which the ECCS analysis assumes the injection valve differential pressure interlock is cleared and the injection valve receives a signal to begin to open.

The proposed Lower Allowable Value (LAV), 64 psid, corresponds to a reactor steam dome pressure of 379.9 psig, and includes allowances above the LAL for instrument loop accuracy and calibration errors. The Allowable Valve is the limiting value of the sensed process variable at which the trip setpoint may be found during instrument surveillance.

The proposed setpoint of 74 psid, corresponds to a reactor steam dome pressure of 389.9 psig, and incorporates an allowance for instrument loop drift.

The current setpoint of 78 psid for the injection valve differential pressure interlock corresponds to a reactor steam dome pressure of 393.0 psig. The value of reactor steam dome pressure at which the ECCS analysis assumes the injection valve differential pressure interlock is cleared and the injection valve receives a signal to begin to open is 300 psig. The difference between these two values of reactor steam dome pressure is 93.9 psi, and is allocated in the setpoint determination process as follows:

Total Loop Accuracy	- 7 psi
Drift	- 10 psi
Excess Unassigned Pressure Margin	- 76.9 psi

As noted above, the proposed setpoint of 74 psid for the injection valve differential pressure interlock corresponds to a reactor steam dome pressure of 389.9 psig. Again, the value of reactor steam dome pressure at which the ECCS analysis assumes the injection valve differential pressure interlock is cleared and the injection valve receives a signal to begin to open is 300 psig. The difference between these two values of reactor steam dome pressure is 89.9 psi, and is allocated in the setpoint determination process as follows:

Total Loop Accuracy	- 11 psi
Drift	- 10 psi
Excess Unassigned Pressure Margin	- 68.9 psi

As a result of increasing the calibration range of the LPCI injection valve differential pressure loop by 200 psi, the accuracy allowance for the instrument loop must be increased by 4 psi (from 7 to 11 psi). In order to maintain the upper analytical limit of 95 psid, the upper allowable value will be decreased from 88 to 84 psid to account for the increased accuracy allowances. Likewise, the setpoint will be decreased to 74 psig from 78 psig to account for the unchanged drift allowance of 10 psi.

As noted above, the lower allowable value will be decreased from 68 psig to 64 psig to account for the unchanged drift allowance of 10 psi for the proposed setpoint of 74 psig. The lower Allowable Limit will be decreased from 61 psid to 53 psid to primarily account for the increased accuracy allowance (11 psi) of the differential pressure loop. As stated previously, the proposed lower allowable limit of 53 psid corresponds to a reactor steam dome pressure of 368.9 psig. The setpoint of 74 psig and the lower allowable value of 64 psid were established, considering the drift and accuracy allowances of the differential pressure instrument loop to clear the interlock prior to reaching the lower allowable limit. The ECCS analysis assumes the differential pressure interlock is cleared and the injection valves are signaled to open when reactor steam dome pressure drops to 300 psig; much lower than the reactor steam dome pressure of 368.9 psig associated with the lower allowable limit of 64 psid.

Therefore, the injection valve differential pressure interlock will continue to clear sooner than was assumed in the ECCS analysis. The signals to open the LPCI injection valves originate from different instrument loops which are not affected by this modification. The timing of initial ECCS flow into the reactor remains unchanged from that previously analyzed.

As discussed in item 6.4 of Inspection Report 50-352/87-09, the importance of the low pressure permissive calibrations for RHR-LPCI injection is emphasized in the Limerick Unit 1 PRA. The dominant failure mode for low pressure injection leading to core damage involves postulated miscalibration of the above pressure channels which causes failure of the differential pressure permissive; thereby preventing LPCI flow into the reactor vessel. The PRA also assumes that the miscalibration is combined with a failure of control room operators to recognize that the injection valves HV-51-017A thru D have not automatically opened.

As can be indicated by our discussion of the proposed setpoints, calibration accuracy and potential instrument drift, the staff has reviewed all aspects to assure that there is adequate margins (aside from the conservative margins in the ECCS codes) so that the differential pressure interlocks will be cleared and the injection valve is signaled to open if needed.

Based upon our review of the licensee's proposed request to change the trip setpoint and allowable values for the LPCI injection valve differential pressure instrument loops, we find that the proposed request 1) continues to maintain an acceptable level of overpressure protection to the low pressure piping and 2) continues to provide adequate margins and response to allow the LPCI injection valves to open in the event of a LOCA. The proposed modification to increase the instrument loop range is approved. The proposed changes to the Trip Setpoint and Allowable Values in Table 3.3.3-2 of the TSs are acceptable.

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 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. 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 48335) on November 30, 1988 and consulted with the State of Pennsylvania. No public comments were received and the State of Pennsylvania did not have any comments.

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 the security nor to the health and safety of the public.

Principal Contributor: Richard J. Clark, D. Yue

Dated: February 9, 1989