February 9, 1994

Docket No. 50-388

Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Byram:

SUBJECT: CHANGES TO HPCI PUMP SUCTION TRANSFER LOGIC, SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 (PLA-3838 and PLA-3955) (TAC NO. M84331)

The Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your letter dated August 19, 1992, as supplemented by letters dated May 18, and October 7, 1993.

This amendment changes the Technical Specifications to revise the logic which controls the automatic transfer of the High Pressure Coolant Injection (HPCI) pump suction source on high suppression pool level.

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, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 101 to License No. NPF-22
2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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See next page

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated August 19, 1992, as supplemented by letters dated May 18, and October 7, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 101 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L 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 and is to be implemented prior to startup in Cycle 7, currently scheduled for May 20, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 9, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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

REMOVE	INSERT
3/4 3-27	3/4 3-27*
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29*
3/4 3-29a	3/4 3-29a
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6*

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(a)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(a)	1, 2, 3	30
c. Reactor Vessel Steam Dome Pressure - Low (Permissive)	2(a)	1, 2, 3, 4*, 5*	31 32
d. Manual Initiation	1/subsystem	1, 2, 3, 4*, 5*	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(a)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(a)	1, 2, 3	30
c. Reactor Vessel Steam Dome Pressure - Low (Permissive)			
1) System Initiation	2(a)	1, 2, 3 4*, 5*	31 32
2) Recirculation Discharge Valve Closure	2(a)	1, 2, 3 4*, 5*	31 32
d. Manual Initiation	1/subsystem	1, 2, 3, 4*, 5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Water Level - Low Low, Level 2	2(a)	1, 2, 3	30
b. Drywell Pressure - High	2(a)	1, 2, 3	30
c. Condensate Storage Tank Level - Low	2(a)(b)	1, 2, 3	34
d. Suppression Pool Water Level - High ***	2(a)	1, 2, 3	34
e. Reactor Vessel Water Level - High, Level 8	2(c)	1, 2, 3	31
f. Manual Initiation	1/system	1, 2, 3	33

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM ##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(f)	1, 2, 3	30
b. Drywell Pressure - High	2(f)	1, 2, 3	30
c. ADS Timer	1(f)	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	2(d)(f)	1, 2, 3	31
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	2(d)(e)(f)	1, 2, 3	31
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1(f)	1, 2, 3	31
g. ADS Drywell Pressure Bypass Timer	2(f)	1, 2, 3	31
h. Manual Inhibit	1	1, 2, 3	33
i. Manual Initiation	1/valve	1, 2, 3	33

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
a. 4.16 kv ESS Bus Under-voltage (Loss of Voltage, < 20%)	1/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	35
b. 4.16 kv ESS Bus Under-voltage (Degraded Voltage, < 65%)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	36
c. 4.16 kv ESS Bus Under-voltage (Degraded Voltage, < 93%)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	36
d 480V ESS Bus 08565 Under-voltage (Degraded Voltage < 65%) ###	2/bus	1/bus	2/bus	1, 2, 3, 4**, 5**	36
e 480V ESS Bus 08466 Under-voltage (Degraded Voltage < 92%) ###	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	36

See footnotes on next page.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

(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) One trip system. Provides signal to HPCI pump suction valves only.

(c) Two out of two logic.

(d) Either 4d or 4e must be satisfied. The ACTION is required to be taken only if neither is satisfied. A channel is not OPERABLE unless its associated pump is OPERABLE per Specification 3.5.1.

(e) Within an ADS Trip System there are two logic subsystems, each of which contains an overall pump permissive. At least one channel associated with each of these overall pump permissives shall be OPERABLE.

(f) A channel may be placed in an inoperable status for up to 2 hours for required surveillance testing provided that all channels in the other trip system are OPERABLE.

* When the system is required to be OPERABLE per Specification 3.5.2

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

** Required when ESF equipment is required to be OPERABLE.

\#\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

\#\#\# Required to be OPERABLE only when Diesel Generator E is either aligned to the Class 1E system or not aligned to the Class 1E system but operating on the Test Facility.

*** The automatic transfer of HPCI pump suction from the condensate storage tank to suppression pool on high suppression pool water level occurs only when HPCI injection valve is open.

SURVEILLANCE REQUIREMENTS (Continued)

2.# For the HPCI system, verifying that the system develops a flow of at least 5000 gpm against a test line pressure of greater than or equal to 245 psig when steam is being supplied to the turbine at 150 ± 15 psig*.

3. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ 1 psid.

4. Verifying that the suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber either on a suppression chamber water level-high signal when HPCI injection valve is open, or on a condensate storage tank water level - low signal.

5. Performing a CHANNEL CALIBRATION of the condensate transfer pump discharge low pressure alarm instrumentation and verifying the low pressure alarm setpoint to be ≥ 113 psig.

d. For the ADS:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

2. At least once per 18 months:

a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b) Manually** opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig* and observing that either:

1) The control valve or bypass valve position responds accordingly, or
2) There is a corresponding change in the measured steam flow.

---

* 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.

** ADS solenoid energization shall be used alternating between ADS Division 1 and ADS Division 2.

For the startup following the Third Refueling and Inspection Outage, this surveillance shall read as follows:

For the HPCI System, verifying that the system develops a flow of at least 4850 gpm against a test line pressure of 600 psig when steam is being supplied to the turbine at 150 ± psig.*

SURVEILLANCE REQUIREMENTS (Continued)

c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm systems and verifying air alarm setpoint of 2070 ± 35 psig on decreasing pressure.

e. At least every 18 months the following shall be accomplished by any series of sequential, overlapping or total channel steps such that the entire channel is tested:

1. A functional test of the interlocks associated with LPCI and CS pump starts in response to an automatic initiation signal in Unit 1 followed by a "False" automatic initiation signal in Unit 2.

2. A functional test of the interlocks associated with LPCI and CS pump starts in response to an automatic initiation signal in Unit 2 followed by a "False" automatic initiation signal in Unit 1.

3. A functional test of the interlocks associated with LPCI and CS pump starts in response to simultaneous occurrence of an automatic initiation signal in both Unit 1 and Unit 2 and a Loss-of-Offsite-Power condition affecting both Unit 1 and Unit 2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER AND LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

By letter dated August 19, 1992, as supplemented by letters dated May 18, 1993 and October 7, 1993, the Pennsylvania Power and Light Company (PP&L or the licensee) submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Unit 2, Technical Specifications (TS). The requested changes would reflect a pending modification to Unit 2 that will revise the logic which controls the automatic transfer of the High Pressure Coolant Injection (HPCI) pump suction source on high suppression pool level.

The May 18, 1993, letter provided a minor revision in the wording to describe the position of the HPCI injection valve. The revision was for clarification and to make the wording the same as the amendment application submitted for Susquehanna, Unit 1. The supplemental change was administrative in nature and did not change the intent of the initial application and did not affect the staff's No Significant Hazards Consideration Determination.

The October 7, 1993, letter documented responses to two questions raised by the NRC staff. The information was confirmatory in nature and did not change the amendment application and did not affect the staff's No Significant Hazards Consideration Determination.

The same changes to the HPCI logic were approved for Susquehanna, Unit 1 by Amendment No. 130, issued on October 19, 1993.

On July 31, 1991, SSES Unit 1 scrammed from full power when a switchyard fault at a fossil plant resulted in de-energization of one of Susquehanna's offsite AC power supplies to a transformer. The de-energization of the transformer resulted in actuation of the Unit 1 'A' Reactor Protection System (RPS) and Main Steam Isolation Valve (MSIV) A/C channels isolation logic. Likewise, on Unit 2, the de-energization resulted in a similar half scram and containment isolations associated with the 'A' RPS power. On Unit 1, the 'B' main steam line (MSL) radiation monitor had failed earlier that morning, resulting in a 'B' RPS (Division II) actuation (half-scram) and a MSIV B/D Logic isolation signal. With the 'B' RPS half-scram and MSIV B/D isolation logic signals already present, actuation of the 'A' channels caused a RPS scram and MSIV closure. The void collapse caused by closure of the MSIVs resulted in a

reactor water level transient. The Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) systems initiated within seconds and injected into the reactor vessel.

The events that transpired following the scram are described in PP&L's Licensee Event Report (LER) 91-008, submitted August 30, 1991. In accordance with the Emergency Operating Procedures (EOPs), the operators used the RCIC system to control level and the safety relief valves (SRVs) to control pressure. Delays were encountered in reestablishing vacuum to the main condenser due to problems with the auxiliary boilers. Controlling pressure with the SRVs is difficult. Shortly after the MSIV closure, the 'E' SRV cycled open and closed twice automatically to control reactor pressure. During the next 7 hours, three additional RPS actuations occurred, one when the high reactor pressure setpoint (1037 psig) was reached and two actuations when reactor vessel level 3 (+13") was reached. Since all control rods were already fully inserted, no rod movement occurred. During this transient, problems were encountered in restoring the Reactor Water Cleanup (RWCU) system and placing the Residual Heat Removal (RHR) system in the shutdown cooling mode of operation. Overall, it took about 30 hours to stabilize the plant. The low reactor water level scram setpoint (+13") was reached 10 times.

One of the causes for delay was the inability to use the HPCI system for pressure control during part of the restoration. Normally, the HPCI pump draws suction from the condensate storage tank (CST). However, if a low CST tank level (3' 7 1/2") or a high suppression pool water level (23' 9") occur, the HPCI suction supply will automatically transfer to the suppression pool. Section 3.6.2.1 of the TSs requires that a certain minimum and maximum volume of water be maintained in the suppression pool, equivalent to a level between 22'0" and 24'0")". Keeping the water level below 24' 0" ensures that there is still adequate space in the suppression pool to accommodate the large volume of water that could conceivably be released into containment from a postulated Loss of Coolant Accident (LOCA) without creating structural concerns. If the 24' 0" limit is exceeded, the plant has to be in cold shutdown within 36 hours. Prior to the reactor scram, the suppression pool water level was 23' 2". Within the first 1 1/2 hours, the suppression pool water level increased to 23' 9", (primarily due to added inventory from SRVs being cycled open to control reactor pressure) which automatically transferred the HPCI suction from the CXS to the suppression pool. The reason for this auto transfer is to keep the water level from exceeding 24' 0" in the event HPCI initiates automatically. However, because of the relatively poor quality of water in the suppression pool (possible rust, et al.), compared to the primary coolant, it is not desirable to pump water from the suppression pool into the reactor unless necessary. The HPCI system remained available for emergency core cooling if needed.

As a result of the complications encountered in coping with a transient that is within the design basis, that is an analyzed event in Chapter 15 of the Final Safety Analysis report (FSAR) and is reanalyzed for each reload, a management meeting was held with the licensee on November 14, 1991, in the NRC's Region I offices. The licensee's presentation was attached to NRC

combined inspection report 50-387/91-21 and 50-388/91-21 issued January 22, 1992. The scram was particularly complicated and challenging to operators, procedures, and hardware. At the meeting, the licensee discussed various actions they proposed to preclude and to improve response to possible future pressurization transients. The actions were documented in the licensee's letter of December 30, 1991, to the NRC (PLA-3707). The licensee agreed to: 1) improve training, communication, and coordination between the plant and the power control center, 2) upgrade the affected emergency operating procedures, 3) evaluate use of a mechanical vacuum pump to pull condenser vacuum when the MSIVs are closed and auxiliary steam is not available, 4) revise the Emergency Action Levels (EALs) on Emergency Core Cooling System (ECCS) initiation, 5) revise the operating procedure for restart of the RWCU with request to reactor vessel differential temperature limit requirements and 6) to pursue five possible design modifications to improve the operator's ability to use the HPCI in pressure control, to manage suppression pool inventory/enthalpy and to recover the RWCU system in the post-transient environment.

One of the key modifications which the licensee committed to implement was a revision to the HPCI suction transfer logic, which is the reason for the subject amendment application.

Pennsylvania Power and Light Company had proposed to complete the modifications of the HPCI suction transfer logic in Susquehanna, Unit 2 during the refueling outage, which began September 11, 1992. The amendment application to effect the logic change for Unit 2 was submitted August 19, 1992. However, the NRC staff did not complete the review of the Unit 2 amendment application in time for the licensee to implement the modification during the fall 1992 refueling outage. The licensee proposes to install the modification in Unit 2 during the refueling outage scheduled for March 1994.

The proposed logic will require that the HPCI injection valve F006 be open in addition to the present requirement of a high suppression pool water level in order for the automatic transfer of the HPCI pump suction from the CST to the suppression pool to take place. This automatically prevents the pump suction transfer when HPCI is not required for injection to the reactor vessel. The automatic transfer of HPCI pump suction from CST to suppression pool on low CST water level is unaffected by this logic change. The physical change to the unit involves a relay being added to the HPCI injection valve (F006) control logic to permit transfer of the HPCI pump suction from the CST to the suppression pool on high suppression pool level only when the F006 valve is open. The relay will be energized by an existing limit switch on the F006 valve that closes as the valve begins to open.

2.0 EVALUATION

The purpose of the automatic transfer of HPCI suction on high suppression pool level is to preserve the containment loading assumptions in the existing safety analysis. Therefore, the impacts on these assumptions as well as HPCI's safety function were evaluated by the licensee as summarized below:

2.1 HPCI Function

The safety function of the HPCI system is to maintain reactor vessel inventory following a Loss of Coolant Accident (LOCA) which does not permit the use of the low pressure Emergency Core Cooling Systems (ECCS). The proposed change is designed to ensure that this function will not be affected since the automatic transfer will occur when HPCI injection is required, based on injection valve position. Various failures associated with the new design were evaluated, and it was determined that failure of the new logic would affect the proper alignment of the suppression pool suction valve (F042) or the F006 valve. However, in the unlikely event of these failures or previously evaluated ones, the Automatic Depressurization System (ADS) will function to ensure that low pressure ECCS can provide adequate core cooling. Further, the new postulated failures were evaluated probabilistically, and the predicted failure rate of each valve was determined not to change significantly.

In response to a staff question, the licensee indicated that in the event of failure of automatic transfer of suction from the CST to the suppression pool, the transfer can still be made either manually by flipping a switch in the control room or, alternatively, can be accomplished by an operator opening the valve. The licensee also indicated that the failure of the new logic will not interfere with the manual transfer operation.

2.2 Containment Analysis

The effect (following an MSIV closure event) of the existing TS is to require suppression pool level control by HPCI if HPCI is used for Reactor Pressure Vessel (RPV) pressure control and suppression pool water level reaches 24 feet. This is due to the TS requirement that HPCI suction automatically switches to the suppression pool if suppression pool level reaches 24 feet. Since a high suppression pool level is a likely occurrence following an MSIV closure event, the licensee does not use HPCI for pressure control because low quality suppression pool water would thereby be pumped into the (clean) condensate storage tank. The result is that in the current configuration, the containment conditions are not impacted by the HPCI system since water is neither taken from the suppression pool nor added.

In the revised configuration, automatic switchover is blocked unless the HPCI injection valve is open. In the pressure control mode, however, the injection valve is closed and HPCI takes its suction from the CST and returns the water to the CST. Therefore, the containment conditions are not impacted in the revised configuration either, since no water is taken or added to the suppression pool. Therefore, the impact of the proposed change on containment is no different than that of the existing TS, if a pipe break is postulated to maximize containment loading during the pressure control mode.

If the suppression pool water level exceeds the limit defined in TS 3.6.2.1, the action statements require restoration of level within one hour or placing the reactor in hot shutdown within the next 12 hours and cold shutdown within

the following 24 hours. In the pressure control mode, the reactor is already scrammed and the unit is headed for cold shutdown. No additional operator actions are required due to the logic modification.

The HPCI turbine has a specified limit on exhaust line backpressure and a vacuum breaker which prevents siphoning water into the exhaust line. These features preclude turbine operation at water levels above 26'-0". In response to a staff question, the licensee estimated that the highest suppression pool water level that may be reached during the pressure control mode will not exceed 25'.

3.0 SUMMARY

The HPCI system has the capacity in its test mode alignment to control reactor pressure. This function and the proposed changes to modify HPCI pump suction transfer logic do not conflict with the primary HPCI function as an ECCS, do not adversely impact plant design parameters or safe operation of other systems, and are not detrimental to the HPCI system components. The staff, therefore, finds the proposed change to be acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 42778). Accordingly, the 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 the amendment.

6.0 CONCLUSION

The Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: M. Razzaque
R. Clark
R. Goel
R. Lobel
I. Ahmed

Date: February 9, 1994