Docket File

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

November 5, 1992

Docket Nos. 50-352
and 50-353

Mr. George J. Beck
Manager-Licensing, MC 52A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Beck:

SUBJECT: CLARIFICATION OF RESIDUAL HEAT REMOVAL SYSTEM SURVEILLANCE REQUIREMENTS, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TSCR 92-09-0) (TAC NOS. M84308 AND M84309)

The Commission has issued the enclosed Amendment No. 57 to Facility Operating License No. NPF-39 and Amendment No. 23 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 11, 1992.

These amendments change surveillance requirement (SR) 4.6.2.3.b and the associated Bases of the TSs to clarify that the intent of this specific SR is to confirm Residual Heat Removal (RHR) pump performance in the Suppression Pool Cooling (SPC) mode of operation. The TS change revises the SR to include the RHR heat exchanger bypass valve as well as the shell side of the RHR heat exchanger in the overall flow path. On October 1, 1992, during the quarterly test of the Unit 2 "B" RHR pump, you determined that leakage through the bypass butterfly valve around the 2B RHR heat exchanger had increased since the previous quarterly test on June 25, 1992, such that the measured flow through the heat exchanger was less than 10,000 gpm. On October 1, 1992, at your request, we granted a verbal temporary waiver of compliance from SR 4.6.2.3.b for Unit 2. This was confirmed by our letter of October 5, 1992, and was in response to your letters of October 2 and 5, 1992. This temporary waiver of compliance expires with the issuance of these amendments.

While we agree with your position that SR 4.6.2.3.b was intended to be an Inservice Test (IST) of the RHR pumps, as discussed in the enclosed safety evaluation, we requested that you submit the subject TS application to focus your attention on the high percentage of RHR pump flow that could be bypassing the RHR heat exchanger and thus could reduce the heat transfer capability of the heat exchanger. The request for the TS application was also prompted by our perception that increased attention should be given to performance of not only the valves in the RHR system but to the performance of valves in the RHR

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Service Water (RHRSW) and Emergency Service Water (ESW) systems, the fouling, pitting and corrosion of the RHR heat exchanger tubes, the corrosion and leaks in the service water systems, the need to utilize freeze-seals to work on components and piping, and the issues being evaluated by your Raw Water Task Force.

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. 57 to
   License No. NPF-39
   Amendment No. 23 to
   License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	MO'Brien(2)	CGrimes, 11E21	CAnderson, RGN-I
NRC & Local PDRs	RClark(3)/JShea	CMoon	
PDI-2 Reading	OGC	ACRS(10)	
SVarga	DHagan, 3206	OPA	
JCalvo	GHill(8), P1-22	OC/LFMB	
CMiller	Wanda Jones, 7103	EWenzinger, RGN-I	

*Previously Concurred

OFC	:PDI-2/LA	:PDI-2/PM	:OTSB*	:C/OTSB*	:OGC*	:PDI-2/D
NAME	:MO'Brien	:RClark:rb	:CMoon	:CGrimes	:EHolier	:CMiller
DATE	:11/6/92	:11/03/92	:09/30/92	:09/30/92	:10/02/92	:11/4/92

Service Water (RHRSW) and Emergency Service Water (ESW) systems, the fouling, pitting and corrosion of the RHR heat exchanger tubes, the corrosion and leaks in the service water systems, the need to utilize freeze-seals to work on components and piping, and the issues being evaluated by your Raw Water Task Force.

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

Sincerely,

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. 57 to
   License No. NPF-39
   Amendment No. 23 to
   License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. George J. Beck
Philadelphia Electric Company

Limerick Generating Station,
Units 1 & 2

cc:

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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. 57
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 August 11, 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-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. 57 , 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 as of its date of issuance and is to be implemented within 14 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

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: November 5, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 57

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 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
3/4 6-15*	3/4 6-15*
3/4 6-16	3/4 6/16
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-3a
-	B 3/4 6-4*
-	-

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by 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.

b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and the suppression pool spray sparger when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by 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.

b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the flow path including the RHR heat exchanger and its associated closed bypass valve, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5.

* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 55 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the suppression chamber air space must not exceed 55 psig. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, suppression pool pressure during the design basis accident is approximately 30 psig which is below the design pressure of 55 psig. Maximum water volume of 134,600 ft^3 results in a downcomer submergence of 12'3" and the minimum volume of 122,120 ft^3 results in a submergence approximately 2'3" less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown through safety/relief valves assuming an initial suppression chamber water temperature of 95°F results in a bulk water temperature of approximately 136°F immediately following blowdown which is below the 190°F bulk temperature limit used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

3/4.6.2 DEPRESSURIZATION SYSTEMS (cont.)

One of the surveillance requirements for the suppression pool cooling (SPC) mode of the RHR system is to demonstrate that each RHR pump develops a flow rate $\geq$10,000 gpm while operating in the SPC mode with flow through the heat exchanger and its associated closed bypass valve, ensuring that pump performance has not degraded during the cycle and that the flow path is operable. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component operability, trend performance and detect incipient failures by indicating abnormal performance. The RHR heat exchanger bypass valve is used for adjusting flow through the heat exchanger, and is not designed to be a tight shut-off valve. With the bypass valve closed, a portion of the total flow still travels through the bypass, which can affect overall heat transfer. However, no heat transfer performance requirement of the heat exchanger is intended by the current Technical Specification surveillance requirement. This is confirmed by the lack of any flow requirement for the RHRSW system in Technical Specifications Section 3/4.7.1. Verifying an RHR flowrate through the heat exchanger does not demonstrate heat removal capability in the absence of a requirement for RHRSW flow. LGS does perform heat transfer testing of the RHR heat exchangers as part of its response to Generic Letter 89-13, which verified the commitment to meet the requirements of GDC 46.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief valves are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. Two pairs of valves are required to protect containment structural integrity. There are four pairs of valves (three to provide minimum redundancy) so that operation may continue for up to 72 hours with no more than two pairs of vacuum breakers inoperable in the closed position.

Each vacuum breaker valve's position indication system is of great enough sensitivity to ensure that the maximum steam bypass leakage coefficient of

$$\frac{A}{\sqrt{k}} = 0.05 \text{ ft}^2$$

for the vacuum relief system (assuming one valve fully open) will not be exceeded.

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated August 11, 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-85 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. 23 , 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 as of its date of issuance and is to be implemented within 14 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

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: November 5, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 23

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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
3/4 6-15*	3/4 6-15*
3/4 6-16	3/4 6/16
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-3a
-	B 3/4 6-4*
-	-

CONTAINMENT SYSTEMS

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by 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.

b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and the suppression pool spray sparger when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by 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.

b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the flow path including the RHR heat exchanger and its associated closed bypass valve, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5.

* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 55 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the suppression chamber air space must not exceed 55 psig. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, suppression pool pressure during the design basis accident is approximately 30 psig which is below the design pressure of 55 psig. Maximum water volume of 134,600 ft^3 results in a downcomer submergence of 12'3" and the minimum volume of 122,120 ft^3 results in a submergence approximately 2'3" less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown through safety/relief valves assuming an initial suppression chamber water temperature of 95°F results in a bulk water temperature of approximately 136°F immediately following blowdown which is below the 190°F bulk temperature limit used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

3/4.6.2 DEPRESSURIZATION SYSTEMS (cont.)

One of the surveillance requirements for the suppression pool cooling (SPC) mode of the RHR system is to demonstrate that each RHR pump develops a flow rate $\geq$10,000 gpm while operating in the SPC mode with flow through the heat exchanger and its associated closed bypass valve, ensuring that pump performance has not degraded during the cycle and that the flow path is operable. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component operability, trend performance and detect incipient failures by indicating abnormal performance. The RHR heat exchanger bypass valve is used for adjusting flow through the heat exchanger, and is not designed to be a tight shut-off valve. With the bypass valve closed, a portion of the total flow still travels through the bypass, which can affect overall heat transfer. However, no heat transfer performance requirement of the heat exchanger is intended by the current Technical Specification surveillance requirement. This is confirmed by the lack of any flow requirement for the RHRSW system in Technical Specifications Section 3/4.7.1. Verifying an RHR flowrate through the heat exchanger does not demonstrate heat removal capability in the absence of a requirement for RHRSW flow. LGS does perform heat transfer testing of the RHR heat exchangers as part of its response to Generic Letter 89-13, which verified the commitment to meet the requirements of GDC 46.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200^{o}F during any period of relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief valves are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. Two pairs of valves are required to protect containment structural integrity. There are four pairs of valves (three to provide minimum redundancy) so that operation may continue for up to 72 hours with no more than two pairs of vacuum breakers inoperable in the closed position.

Each vacuum breaker valve's position indication system is of great enough sensitivity to ensure that the maximum steam bypass leakage coefficient of

$$\frac{A}{\sqrt{k}} = 0.05 \text{ ft}^2$$

for the vacuum relief system (assuming one valve fully open) will not be exceeded.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 57 AND 23 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated August 11, 1992, the Philadelphia Electric Company (PECo, the licensee) submitted a request for changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications (TS). The requested changes would change the TSs to clarify the flow Surveillance Requirement (SR) for the Suppression Pool Cooling (SPC) Mode of the Residual Heat Removal (RHR) System.

2.0 BACKGROUND

The RHR system is described in Section 5.4.7 of the Updated Final Safety Analysis Report (UFSAR) for the Limerick Generating Station (LGS), Units 1 and 2. The RHR system is comprised of four independent loops. Each loop contains a motor-driven pump, piping, valves, instrumentation, and controls. Each loop takes suction from the suppression pool and is capable of discharging water to the reactor vessel via a separate vessel nozzle loop or back to the suppression pool via a full flow test line. In addition, loops A and B have heat exchangers that are cooled by the Residual Heat Removal Service Water (RHRSW) system. These two loops can also take suction from the reactor recirculation system suction and can discharge into the reactor recirculation system discharge.

The RHR system has five subsystems or modes of operation: 1) the Residual Heat Removal or shutdown cooling mode, 2) the Low Pressure Coolant Injection (LPCI) mode, 3) the Suppression Pool Cooling (SPC) mode, 4) the Containment Spray Cooling mode and 5) the Reactor Steam Condensing mode. The latter has been abandoned at both Limerick and Susquehanna. The functional design basis for the LPCI mode is to pump 10,000 gpm of water per loop, using the separate loop pumps, from the suppression pool into the core region of the reactor vessel. In this mode, the RHR pump recirculates the suppression pool water directly to the reactor vessel via the RHR heat exchanger bypass line without going through the RHR heat exchanger. When the RHR system operates in the SPC mode, the suppression pool water is pumped from the pool through the shell-side of the RHR heat exchanger and returned to the suppression pool. The heat in the suppression pool water is transferred to the RHRSW which flows through the tube-side of the RHR heat exchanger. In the SPC mode of operation, the butterfly bypass valve around the heat exchanger is closed. A simplified diagram of the SPC mode for one heat exchanger is shown in the attached Figure 1.

The RHR heat exchangers are vertical shell and tube units with 530 type 304L stainless steel "U" bend tubes, nominally one-inch diameter and 0.049 inch wall. The tube sheet is also stainless steel clad. The shell is carbon steel. The effective surface area of the tubes was initially about 6281 ft^2. The heat exchangers were sized on the basis of duty for the shutdown cooling mode. All other modes, such as SPC, require less heat transfer.

During normal plant operation, the RHR system is shut down and thus the RHR heat exchangers are not in operation. They normally are filled with demineralized water (wet layup) and a biocide and/or corrosion inhibitor. Shutdown cooling would normally be expected to be the most common mode of operation for the RHR heat exchangers. As discussed previously, the RHR heat exchangers also provide suppression pool cooling. Any time the suppression pool temperature exceeds a pre-set limit (e.g., 90°F during normal operation) flow commences through one of the RHR heat exchangers in order to cool the suppression pool water. During the Unit 1 cycle four operation, the suppression pool water had to be cooled periodically due to leaking safety relief valves (SRVs). During June, July and August of 1991, the IA heat exchanger was used for several hours, nearly every day, to cool the suppression pool water. Between April 1991 and March 1992, the IA RHR heat exchanger was run for over 1200 hours to cool the suppression pool water. In early 1992, the licensee noted a decline in the heat transfer performance of the IA RHR heat exchanger and operations personnel began to use the 1B heat exchanger to augment the IA unit.

Limerick Unit 1 shutdown for the fourth refueling outage on March 20, 1992. Because of the noted decline in the IA RHR heat exchanger performance near the end of the cycle, in May 1992 the licensee tested both the 1A and 1B units in accordance with Generic Letter (GL) 89-13 for heat transfer duty. This testing (May 8, 1992) revealed that the 1A heat exchanger exhibited a fouling factor in excess of design limitations. The 1B unit passed, but some degradation was noted. To restore performance, the 1A unit was chemically cleaned with a relatively mild mixed organic acid solution primarily designed to remove suspected hardness scale. The subsequent testing showed little improvement in heat transfer. The licensee decided to remove the bottom head and perform a visual examination of the tubes in the 1A RHR heat exchanger. There are inlet and outlet isolation valves on the service water (Spray Pond water) lines into and out of each RHR heat exchanger. The licensee has experienced many problems with these 20" Anchor Darling valves, such as broken internals, sticking, leakage, etc. (See the Resident Inspector's reports for November 18 to December 31, 1990, 50-352/90-27 and 50-353/90-27; September 1, to October 5, 1991, 50-352/91-18 and 50-353/91-19; January 5 to February 15, 1992, 50-352/92-03 and 50-353/92-03; et al). Because the isolation valves were not leak-tight, the licensee had to establish and maintain freeze seals on the inlet and outlet lines to the 1A (and later the 1B) heat exchanger to remove the bottom head. Upon removing the bottom head of the 1A heat exchanger on May 27, 1992, an examination of the lower part of the tubes disclosed large quantities of a black, slimy, tar-like substance that remained after chemical cleaning. The licensee contracted to hydrolyze the 530 tubes, including the U-Bend areas. There is no estimate of the total amount of the black gunk removed, but from the writer's observation, there was a significant volume of black crud on the floor from hydrolyzing a single leg of each tube. Following cleaning, the licensee performed a 100% eddy current (ECT) inspection of all 530 tubes, including the "U" bend area where possible. The ECT revealed defects (pits) in all of the tubes examined; 72 tubes (13.6%) exhibited indications greater than 90% through wall. The ECT indicated that over half the tubes contained defects of more than 60% through-wall. Straight sections of two tubes were removed for examination. As a result of various tests and analysis, the licensee plugged 37 tubes in the 1A heat exchanger. In view of what they found in the 1A heat exchanger, the licensee removed the head from the 1B heat exchanger, and inspected the tubes. The same black, slimy, tar-like substance was on the inside of the tubes. The tubes were hydrolyzed, and inspected by eddy current testing. Six of the U-Bends with the deepest measured pits(>89%) were evaluated by EPRI with an MRPC (Pancake Probe) to assess the characteristics of the pits. All of the 530 tubes showed some pitting; 70% of the tubes indicated pit depths greater than 50% through-wall. A total of 35 tubes (6.60%) were plugged. The general guideline adopted by PECo Engineering was to plug any tube in which the ECT measurements in either the inlet, outlet or U-Bend areas of the tubes indicated a maximum pit depth of 80% through wall or greater.

On June 25, 1992, representatives from PECo met with the NRC staff at the NRC Region I Headquarters to discuss the results of their testing and proposed actions to address the corrosion problems. PECo also presented their safety assessments to confirm that there was no reduction in margin of safety with the plugged and pitted tubes. (See the Resident Inspector's reports for June 7-July 18, 1992, and July 19-August 29, 1992, 50-352/92-17 and 50-353/92-17 and 50-352/92-23 and 50-353/92-23, respectively.)

Because of the number of plugged tubes in each RHR heat exchanger, in the later part of June 1992 and early July 1992, PECo conducted comprehensive flow and heat transfer tests on the RHR and the RHRSW systems in both Units 1 and 2. TS SR 4.6.2.3.b currently states that the SDC mode of RHR shall be demonstrated to be operable, "By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the RHR heat exchanger, the suppression pool, and the full flow test line when tested pursuant to Specification 4.0.5." TS Section 4.0.5 invokes the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI In-Service Testing (IST) of pumps and valves, indicating that the intent of this SR is to confirm the performance of the RHR pumps when aligned in the SPC flowpath. Pump performance is therefore one of the parameters surveillance in order to determine operability of the SPC mode of the RHR System. The initial flow tests showed that the RHR pumps were putting out more than the specified 10,000 gpm (e.q., 10,500 gpm) but that about 2500 gpm was going through the supposedly closed bypass valve rather than through the heat exchanger. The bypass valve is a butterfly type valve that was not intended by design to be leak tight. By adjusting the limit switch, the licensee reduced leakage through the valve to about 1000 gpm, which is still about 10% of the total flow. There was no safety concern with the amount of flow being bypassed. As discussed previously, the 10,000 gpm requirement on the pumps was based on the LPCI mode of operation. The SPC mode only requires about 7500 gpm. Nevertheless, in view of about 25% of the flow going through a supposedly closed valve, the performance of the valves in the RHRSW, the corrosion and plugging of small diameter lines in the RHRSW, the relative priority given to the "Raw Water Issues Task Force" to develop a plan of action to address the problems in the service water systems, etc., the Resident Inspectors raised the question about interpretation of 4.6.2.3.b -- specifically, did "through the RHR heat exchanger" mean that the specified 10,000 gpm pump output was to be measured at the pump discharge, as PECo has been measuring it quarterly as part of the In-Service Test (IST) program or should it be measured at the outlet of the heat exchanger. There is no in-line instrumentation to measure flows at the latter locations. The question was referred to NRR for interpretation. On June 24, 1992, we had a conference call with PECo's Engineering and Plant staff on the issue, with the NRC Resident Inspectors, the NRC's Region I Project staff and NRR's Project Manager and Standard Technical Specification staff. The NRC's position was that there was no safety issue, but a literal interpretation of "through the heat exchanger" would require that the specified flow should go through the heat exchanger shell. We suggested that PECo submit a TS application to clarify the basis (intent) of SR 4.6.2.3.b. On August 11, 1992, PECo submitted the subject application. By letter dated July 20, 1992, PECo also submitted Licensee Event Report (LER) No. 1-92-013 discussing the surveillance tests.

PECo was able to demonstrate that there was 10,000 gpm of flow through the RHR heat exchangers in both Units 1 and 2 in the SPC mode. To do so, they had to remove a restricting orifice in each of the LPCI test return lines at locations downstream of the globe valves. The orifices were added under Modification Nos. 86-0024 and 5791 for Unit 1 and were part of the original design for Unit 2. The orifices are needed so that adequate RHR pump discharge pressure can be achieved (during use of the LPCI test mode or SPC mode) without excessively throttling the globe valves. Without these orifices, the globe valves could be subject to significant damage as a result of cavitation. The modifications in 1990 and 1991 reduced suppression pool temperature stratification by promoting better mixing. From the standpoint of plant safety, it is desirable to reinstall these orifices as soon as possible. Implementation of the proposed TS change will permit reinstallation of the orifices.

On October 1, 1992, PECo conducted the quarterly flow test of the Unit 2 'B' RHR pump and the 2B loop of SFC mode of the RHR system in accordance with TS SR 4.6.2.3.b. Using special flow monitoring instrumentation, the plant staff determined that there was an increase in flow of about 300 gpm through the closed Unit 2 'B' RHR heat exchanger bypass valve, HV-C-51-2F048B, since the previous test on June 25, 1992. The surveillance test procedure was repeated several times and the actual flow through the Unit 2 RHR heat exchanger was between 9800 and 9950 gpm, slightly less than the 10,000 gpm in SR 4.6.2.3.b. The licensee requested a temporary waiver of compliance from SR 4.6.2.3.b on October 1, 1992, pending issuance of the subject amendments. The licensee followed-up the verbal request with written requests on October 2 and 5, 1992. The licensee's initial request was verbally approved by the NRC on October 1, 1992, and documented by letter dated October 5, 1992. The temporary waiver of compliance expires with issuance of these amendments. Even though the increase in bypass flow was not consequential, it was another indication that flow through the butterfly bypass valves can change, even if the valve has not been operated. In their letter of October 2, 1992, the licensee's explanation for the slight increase in bypass flow was that the "data showed a slight increase in the degradation of the valve." We understand that the licensee is evaluating the design and operation of the bypass valves to determine if leakage through the valves can be controlled within a predictable range. As a result of the tube plugging and fouling in the 1A and 1B RHR heat exchangers, the licensee reanalyzed the minimum required flows of service water and suppression pool water through the heat exchangers to perform the design functions. In their letter of October 5, 1992, the licensee reported that the revised calculations showed that an RHR system flow rate of approximately 7500 gpm is needed to pass through the RHR heat exchangers to remove the design heat load, assuming the design maximum value for the heat exchanger fouling factors, a nominal number of plugged tubes, and the maximum RHRSW inlet temperature. In their response of January 29, 1990, to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," the licensee committed to periodically test the heat transfer performance of the RHR and other heat exchangers. This commitment was reiterated in the licensee's letters of August 11, 1992 and October 2, 1992. This is the important consideration, more so than flows through the heat exchanger, although there is a close relationship.

3.0 <u>EVALUATION</u>

TS SR 4.6.2.3.b currently states that the suppression pool cooling (SPC) mode of RHR operation shall be demonstrated to be operable, "By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the RHR heat exchanger, the suppression pool, and the full flow test line when tested pursuant to Specification 4.0.5."

Since the TS Bases for this surveillance requirement did not address the LGS design which included an RHR heat exchanger bypass valve, this proposed change provides clarification of this TS SR. PECo's position is that the purpose of this TS SR is to confirm the RHR pump performance while operating in the SPC mode, pursuant to the IST requirement of TS Section 4.0.5. Specifically, the purpose of this TS SR is to confirm that each RHR pump develops a flow rate of 10,000 gpm through the most restrictive flow path. This includes the RHR heat exchanger and its associated closed bypass valve, the suppression pool, and the full flow test line. This TS SR is not intended to confirm the heat transfer capability of the RHR heat exchanger since there is no equivalent TS SR for the flow of RHRSW through the RHR heat exchanger. Periodic heat transfer testing of the RHR heat exchanger is required by 10 CFR 50, Appendix A, GDC 40, and implemented by administrative controls as committed to in PECo's response to NRC Generic Letter 89-13. Accordingly, PECo proposes to change TS SR 4.6.2.3.b to clarify its purpose as follows:

"b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the flow path including the RHR heat exchanger and its associated closed bypass valve, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5."

This proposed clarification does not change the operation of the RHR system in the SPC mode, the heat transfer capability of the system, or the existing heat transfer testing requirements. The proposed TS changes do not involve any physical changes to the RHR system components. These proposed TS changes only clarify the fact that the purpose of TS SR 4.6.2.3.b is to confirm the RHR pump performance while operating in the SPC mode, i.e., flow through the most restrictive conditions of the flow path. The RHR heat exchanger performance will continue to be verified by periodic testing as described above. Therefore, the pressure suppression function of the suppression pool is unaffected by these TS changes.

We agree with PECo's interpretation. The proposed changes to the SR and Bases will clarify the interpretation as we suggested in our June 24, 1992 telecon with the PECo staff. The proposed changes are acceptable.

4.0 <u>STATE CONSULTATION</u>

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

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

The amendments change the surveillance requirements. The NRC 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 that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 40218). Accordingly, the amendments meet 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 amendments.

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

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Date: November 5, 1992

Attachment: Figure 1

FIGURE 1
SUPPRESSION POOL COOLING
MODE OF 1A RHR
(TYPICAL)