Docket Nos. 50-352
and 50-353

February 17, 1

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

Dear Mr. Hunger:

SUBJECT: SUPPRESSION POOL BYPASS LEAK TEST, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M88332 AND M88333)

The Commission has issued the enclosed Amendment No. 68 to Facility Operating License No. NPF-39 and Amendment No. 31 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 November 30, 1993.

These TS amendments make the following changes: 1) decrease the test frequency of the drywell-to-suppression chamber bypass leak test to coincide with the primary Containment Integrated Leak Rate Test interval, and 2) require an additional test to measure vacuum breaker leakage area for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

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

Sincerely,
/S/
Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 68 to License No. NPF-39
   Amendment No. 31 to License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	MO'Brien(2)	CGrimes, 11E21
NRC & Local PDRs	FRinaldi/JShea	RBarrett
PDI-2 Reading	OGC	ACRS(10)
SVarga	DHagan, 3206	OPA
JCalvo	GHill(4), P1-22	OC/LFDCB
CMiller	EWenzinger, RGN-I	CAnderson, RGN-I

*Previously Concurred

OFC	:PDI-2/LA	:PDI-2/PM	:SCSB*	:OGC*	:PDI-2/D
NAME	:MO'Brien	:FRinaldi:rb	:RBarrett	:EHoller	:CMiller
DATE	:2/4/94	:2/8/94	:02/02/94	:02/04/94	:2/17/94

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

Docket Nos. 50-352
and 50-353

February 17, 1994

Mr. George A. Hunger, Jr.
Director-Licensing, MC 52A-5
Philadelphia Electric Company
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P.O. Box No. 195
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SUBJECT: SUPPRESSION POOL BYPASS LEAK TEST, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M88332 AND M88333)

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These TS amendments make the following changes: 1) decrease the test frequency of the drywell-to-suppression chamber bypass leak test to coincide with the primary Containment Integrated Leak Rate Test interval, and 2) require an additional test to measure vacuum breaker leakage area for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

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

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Frank Rinaldi

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

Enclosures:
1. Amendment No. 68 to
   License No. NPF-39
   Amendment No. 31 to
   License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 68
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 30, 1993, 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. 68 , 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.

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: February 17, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 68

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 6-13	3/4 6-13*
3/4 6-14	3/4 6-14
B 3/4 6-3	B 3/4 6-3*
B 3/4 6-3a	B 3/4 6-3a
B 3/4 6-4	B 3/4 6-4

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.

d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

c. By verifying at least two suppression chamber water level indicators and at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the water level and temperature alarm setpoint for:

1. High water level ≤ 24′1½"

2. High water temperature:

   a) First setpoint ≤ 95°F

   b) Second setpoint ≤ 105°F

   c) Third setpoint ≤ 110°F

   d) Fourth setpoint ≤ 120°F

d. Drywell-to-suppression chamber bypass leak tests shall be conducted at 40 +/- 10 month intervals to coincide with the ILRT at an initial differential pressure of 4 psi and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.

e. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area $A/\sqrt{k}$ contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.d is not conducted.

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 rated conditions. 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 below the design pressure. 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 temperature 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 ≥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.

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.

During a LOCA, potential leak paths between the drywell and suppression chamber airspace could result in excessive containment pressures, since the steam flow into the airspace would bypass the heat sink capabilities of the chamber. Potential sources of bypass leakage are the suppression chamber-to-drywell vacuum breakers (VBs), penetrations in the diaphragm floor, and cracks in the diaphragm floor and/or liner plate and downcomers located in the suppression chamber airspace. The containment pressure response to the postulated bypass leakage can be mitigated by manually actuating the suppression chamber sprays. An analysis was performed for a design bypass leakage area of $A/\sqrt{k}$ equal to 0.0500 ft² to verify that the operator has sufficient time to initiate the sprays prior to exceeding the containment design pressure of 55 psig. The limit of 10% of the design value of 0.0500 ft² ensures that the design basis for the steam bypass analysis is met.

DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test in not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

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. 20555-0001

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
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 November 30, 1993, 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. 31 , 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.

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: February 17, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 31

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 6-13	3/4 6-13*
3/4 6-14	3/4 6-14
B 3/4 6-3	B 3/4 6-3*
B 3/4 6-3a	B 3/4 6-3a
B 3/4 6-4	B 3/4 6-4

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.

d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F, and

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

c. By verifying at least two suppression chamber water level indicators and at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the water level and temperature alarm setpoint for:

1. High water level ≤ 24'1½"

2. High water temperature:

a) First setpoint ≤ 95°F

b) Second setpoint ≤ 105°F

c) Third setpoint ≤ 110°F

d) Fourth setpoint ≤ 120°F

d. Drywell-to-suppression chamber bypass leak tests shall be conducted at 40 +/- 10 month intervals to coincide with the ILRT at an initial differential pressure of 4 psi and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.

e. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area $A/\sqrt{k}$ contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.d is not conducted.

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.

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 ≥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.

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.

During a LOCA, potential leak paths between the drywell and suppression chamber airspace could result in excessive containment pressures, since the steam flow into the airspace would bypass the heat sink capabilities of the chamber. Potential sources of bypass leakage are the suppression chamber-to-drywell vacuum breakers (VBs), penetrations in the diaphragm floor, and cracks in the diaphragm floor and/or liner plate and downcomers located in the suppression chamber airspace. The containment pressure response to the postulated bypass leakage can be mitigated by manually actuating the suppression chamber sprays. An analysis was performed for a design bypass leakage area of $A/\sqrt{k}$ equal to 0.0500 ft^2 to verify that the operator has sufficient time to initiate the sprays prior to exceeding the containment design pressure of 55 psig. The limit of 10% of the design value of 0.0500 ft^2 ensures that the design basis for the steam bypass analysis is met.

DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test in not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

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. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 68 AND 31 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 November 30, 1993, the Philadelphia Electric Company (PECo or the licensee) submitted a request for changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the TSs to:

1. Decrease the test frequency of the drywell-to-suppression chamber bypass test to coincide with the test frequency for the 10 CFR Part 50, Appendix J, Integrated Leakage Rate Test (ILRT). This test frequency would require that three low pressure bypass tests be conducted at 40±10-month intervals during each 10-year service period, and

2. Require an additional surveillance test to measure the vacuum breaker leakage area for those outages for which the above drywell-to-suppression chamber bypass test is not scheduled.

The current Technical Specifications specify that a drywell-to-suppression chamber bypass test be conducted at least once-per-18-months to verify an acceptable containment bypass effective leakage area, $A/\sqrt{K}$. Substitution of the suppression chamber bypass test with the individual vacuum breaker bypass tests would result in a significant economic benefit to the licensee while maintaining adequate safety as discussed below.

2.0 DISCUSSION AND EVALUATION

2.1 Proposed TS Change

The current Technical Specification, 4.6.2.1.d, states that:

"The suppression chamber shall be demonstrated OPERABLE:

d. At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 4.0 psi and verifying that the A/√K calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18-month test schedule may be resumed."

The proposed TS, 4.6.2.1.d and e, would state:

"The suppression chamber shall be demonstrated OPERABLE:

d. Drywell-to-suppression chamber bypass leak tests shall be conducted at 40 +/- 10-month intervals to coincide with the ILRT at an initial differential pressure of 4 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit at which time the test schedule may be resumed.

e. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area A/√k contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-supression chamber bypass leak test in Specification 4.6.2.1.d is not conducted.

2.2 LGS Mark II Pressure Suppression Containment Design

LGS incorporates a Mark II containment with the drywell located over the suppression chamber and separated by a diaphragm slab. The suppression chamber contains a pool of water having a depth that varies between 22' and 24'-3" during normal operation. Eighty-seven downcomers and 14 main steam safety/relief valve (SRV) discharge lines penetrate the diaphragm slab and terminate at a pre-designed submergence within the pool. During a loss of coolant accident (LOCA) inside containment, the containment design directs steam from the drywell to the suppression pool via the downcomers through the pool of water to limit the maximum containment pressure response to less than the design pressure of 55 psig. The effectiveness of the LGS pressure suppression containment design requires that the leak path from the drywell to the suppression chamber airspace be minimized. Steam that enters the suppression pool airspace through leak paths will bypass the suppression pool and can result in a rapid post-LOCA increase in containment pressure depending on the size of the bypass flow area.

2.3 Basis for Current TS Requirements

The licensee's architect/engineer calculated the containment pressure response based on a conservative design bypass flow area of $A/\sqrt{K}$ equal to 0.0500 ft^2 (7.20 in^2). The analysis assumed a small break LOCA with a differential pressure between the drywell and suppression chamber airspace equal to the static pressure due to downcomer submergence. The analysis showed that it takes over 30 minutes from the onset of the LOCA to reach the containment design pressure of 55 psig. The steam bypass analysis results were evaluated by the NRC and reported in the LGS Safety Evaluation Report (NUREG-0991). The criteria for the staff review were the requirements stipulated in the Standard Review Plan (SRP), Appendix I of Section 6.2.1.1.c, "Steam Bypass for Mark I, II, III Containments." The staff concluded that LGS's steam bypass capability is adequate, since the operator has sufficient time to actuate the suppression chamber sprays prior to reaching the containment design pressure.

TS 3.6.2.1.b conservatively specifies a maximum allowable bypass area of 10% of the design value of 0.0500 ft^2. The TS limit provides an additional factor of 10 safety margin above the conservatisms taken in the steam bypass analysis. The drywell-to-suppression chamber bypass test required by TS 4.6.2.1.d verifies that the actual bypass flow area is less than or equal to the TS limit. The bypass leakage test ensures that degradation in the measured bypass area is identified and corrected to ensure containment integrity during LOCA events.

The design value for leakage area is determined by analyzing a spectrum of LOCA break sizes. For each break size there is a limiting leakage area. In determining the limiting leakage area, credit is taken for the capability of operators to initiate drywell and suppression pool sprays after a period of time sufficient for them to realize that there is a significant bypass leakage flow. The effect of suppression pool bypass on containment pressure response is greatest with small breaks. The design value of 0.0500 ft^2 for LGS represents the maximum leakage area that can be tolerated for that break size that is most limiting with respect to suppression pool bypass.

2.4 Potential for Bypass Leakage

Several potential bypass leakage pathways exist:

Leakage through the diaphragm floor penetrations (SRV discharge line and downcomers), Cracks in the diaphragm floor/liner plate,

Cracks in the downcomers that pass through the suppression pool airspace,

Valve seat leakage in the four sets of drywell-to-suppression chamber containment vacuum breakers, and

Seat leakage of isolation valves in piping connecting the drywell and the suppression chamber air space.

Each potential flow pathway has been evaluated by the licensee with respect to the potential for suppression pool bypass leakage.

Several plant design features and the bypass leak test data measured to date confirm that the leakage from other than the vacuum breaker assemblies is negligible and indicates that this leakage will continue to be negligible for the proposed increased duration between tests. All pressure boundary penetrations between the drywell and the suppression chamber are welded except the vacuum breaker valves and the blind flanges closing 10 spare nozzles in the downcomers. All pressure boundary penetrations between the drywell-to-suppression chamber have been fabricated, erected, and inspected in accordance with the American Society of Mechanical Engineers Code, Section III, Subsection NC, 1971 Edition, with the exception of the tees supporting the vacuum breakers.

The downcomer and SRV discharge lines penetrate through the diaphragm slab and terminate in the suppression pool. A steel ring plate is welded to the outside of the downcomers. The downcomer/ring plate assemblies are embedded in the diaphragm slab with the top surface of the ring plate flush with the drywell side of the diaphragm slab. All connections are welded to form a continuous steel membrane between the liner plate and downcomer penetrations. The SRV discharge lines are routed through welded flued heads at the diaphragm floor. The flued head design and construction are similar to the downcomer penetrations and also provide a continuous steel barrier. The downcomer and SRV discharge lines are designed and constructed to safety-related requirements. In addition, they are designed for all postulated loading conditions, including seismic, hydrodynamic pressure, and temperature loads. The conservative design requirements ensure that the SRV discharge and the downcomer lines will not contribute to bypass leakage.

The diaphragm floor is a reinforced concrete slab approximately 3.5 feet thick. The drywell side surface of the diaphragm slab is capped with a 1/4-inch thick carbon steel liner plate. The liner plate and diaphragm slab provide a barrier against the potential for bypass leakage through the diaphragm floor. The structure integrity of the diaphragm floor and penetrations was demonstrated during the pre-operational test program. The drywell was pressurized to a drywell-to-suppression chamber differential pressure of above 30 psid, which envelopes the maximum drywell-to-suppression chamber differential pressure postulated to occur during LOCA conditions.

The most likely source of potential bypass leakage is the four sets of drywell-to-suppression chamber vacuum breakers. Each set consists of two vacuum breakers in series, flange mounted to a tee off the downcomers in the suppression chamber airspace. The drywell-to-suppression chamber bypass leak test is currently required by TS Surveillance Requirement 4.6.2.1.d to be completed during each refueling outage and the results are used to verify that the total bypass area, including that due to the vacuum breakers, meets the TS limit. If maintenance has been performed on the vacuum breakers, this test also serves as a post-maintenance vacuum breaker leakage area test.

The proposed TS changes decrease the frequency of the drywell-to-suppression chamber bypass leak test. The drywell-to-suppression chamber bypass leak test data obtained following vacuum breaker maintenance cannot be utilized to determine vacuum breakers leakage reliability over the duration of the proposed test interval extension. To address this concern and collect additional vacuum breaker leakage data, the proposed TS changes include an additional requirement to perform vacuum breaker leakage tests as described in 2.6 below.

There are several potential bypass flow paths between the drywell and suppression chamber air spaces via piping systems external to the containment. All flow paths have multiple in-series containment isolation valves. The piping systems include:

1) Containment vent and purge lines (20" and 24" diameter lines with two flow paths from the drywell to the suppression chamber),

2) Drywell and suppression chamber spray lines (18" and 6" diameter lines with two flow paths from the drywell to the suppression chamber),

3) Containment Integrated Leak Rate Test data acquisition system line (3/4" diameter lines with one flow path from the drywell to the suppression chamber),

4) Containment atmosphere sampling lines (1" and 2" diameter lines with two flow paths from the drywell to the suppression chamber),

5) Containment instrument gas line (1" diameter lines with two flow paths from the drywell to the suppression chamber).

The potential bypass leakage from the above cross-connected piping systems flow paths is considered to be negligible compared to the TS allowable drywell-to-suppression chamber bypass leakage (i.e., 0.720 in^2) based on the following:

The cross-connected piping is isolated from containment by drywell and suppression chamber containmetn isolation valves. All flow paths have multiple, in-series containment isolation valves that are designed to meet stringent leakage criteria as specified in 10 CFR Part 50, Appendix J.

The TS require performance of a periodic Local Leak Rate Test (LLRT) to ensure that the valves comply with the 10 CFR Part 50, Appendix J, Type C test criteria. Therefore, leakage from the drywell to the suppression chamber airspace can only occur through multiple isolation valves.

The licensee has performed a bounding analysis to determine the maximum potential bypass leakage area from the above sources. The leakage area was derived from the TS allowable leakage for the containment isolation valves located in the potential flow paths. The TS allowable leakage from the 10 CFR Part 50, Appendix J, Type B and Type C (i.e., LLRT) testing boundaries is 60% (0.6) of the allowed leakage, La (i.e., 94,964 scc/min). A conservative estimate of the potential leakage was determined by assuming that the total TS allowable leakage is bypassed to the suppression chamber airspace. The 0.6 La is a bounding leakage rate since it includes valves with the potential to bypass and includes all other valves and penetrations subject to Type B and Type C testing. The equivalent leakage area for a leakage rate of 94,964 scc/min at the safety analysis peak accident primary containment pressure of 44 psig is 0.00845 in. which is 1.17% of the TS allowable bypass leakage area of 0.720 in. The average total LLRT results for the previous six LGS refueling outages is 56,222 scc/min. The equivalent leakage area corresponding to this average leakage is 0.0050 in. or 0.7% of the TS allowable bypass leakage area of 0.720 in.

The LGS LLRT program is procedurally controlled and requires that program goals be set to define a target LLRT leakage rate for each isolation valve. The target leakage rates are based on the prior leakage history for each valve, coupled with a LLRT program philosophy that emphasizes the need to maintain LLRT leakage as low as practical. The program requires leakages that exceed the target values be investigated to determine if corrections must be made to LLRT totals.

2.5 Operational Experience

The licensee provided a discussion of past results of suppression pool bypass testing at LGS. Past testing has been performed following vacuum breaker maintenance and is therefore indicative of leakage of the non-vacuum breaker (i.e., passive) leakage sources such as the floor and penetrations. Based on the data from previous tests, bypass leakage through the floor and floor penetrations is consistently very low and it can be concluded that any significant pool bypass leakage under LOCA conditions would likely be via vacuum breaker disk leakage.

2.6 Substitution of Vacuum Breaker Leakage Tests for Suppression Pool Bypass Test During Non-Integrated Leakage Rate Test (ILRT) Outages

Analyses of the design and construction of potential leakage paths and of the historical results of suppression pool bypass tests, as discussed in 2.4 and 2.5 above, indicate that the drywell-to-suppression pool vacuum breakers constitute the only significant potential bypass leakage path. Based on this finding, the staff finds that there is sufficient basis to allow individual vacuum breaker leakage tests to substitute for the suppression pool bypass test during non-ILRT outages. This conclusion reflects findings of use of conservative margins with respect to vacuum breaker leakage test acceptance criteria, and assurance that the passive containment structure, liner and penetrations are not likely to deteriorate between normal suppression pool bypass tests. The staff acknowledges that requiring the performance of a complete normal suppression pool bypass test on a schedule consistent with the ILRT assures that potential degradation due to corrosion of the drywell liner or downcomer piping in the suppression pool airspace would be detected in a timely manner. The staff also acknowledges that (a) limiting total vacuum breaker leakage to 24% of the total leakage limit (which itself is 10% of the design capability), and (b) limiting individual vacuum breaker set leakage to 12% (twice the assumed leakage from a single set) of the specified limit, provide large, conservative margins.

2.7 Proposed Vacuum Breaker Leakage Test

The licensee intends to demonstrate compliance with the proposed TS 4.6.2.1.e by measuring and summing the $A/\sqrt{K}$ for each of the five vacuum breaker sets. The combined $A/\sqrt{K}$ would be limited to 0.173 in^2 [(24%)(10%)(0.0500 ft^2)(144 in^2/ft^2) = 0.173 in^2]. In addition, the individual $A/\sqrt{k}$ for each vacuum breaker set would be limited to 0.0865 in^2 [(0.173 in^2 ÷ 4 sets of vacuum breakers)(a factor of 2 times the acceptable total) = 0.0865 in^2] equivalent to a leakage area twice the assumed leakage from a single breaker set.

The leakage test will be conducted on each set of vacuum breakers (i.e., four vacuum breaker sets per unit) during each refueling outage when the drywell-to-suppression chamber bypass leak test would not be required to be performed. If maintenance is performed on the vacuum breaker assemblies, this additional test will be performed post-maintenance to verify that the leakage is acceptable. This test will be conducted at a drywell-to-suppression chamber differential pressure of 4.0 psi (i.e., the same as differential pressure required for the drywell-to-suppression chamber bypass leak test) by either pressurizing the drywell side of the vacuum breakers or inducing a vacuum on the suppression chamber side of the vacuum breakers. The total vacuum breaker leakage areas for all four sets of vacuum breakers will be less than or equal to 24% of the TS limit (i.e., 0.24 x 0.720 in^2 = 0.173 in^2). This proposed acceptable vacuum breaker leakage area provides a 76% margin of the TS limit to account for the leakage paths other than the vacuum breakers. Previous bypass leakage testing measured a maximum bypass leakage area of 5.56% of the TS limit. The 76% margin is sufficiently large to accomodate the other expected leakage sources. In addition, each set of vacuum breakers will be limited to a leakage area twice the assumed leakage from a single vacuum breaker set, assuming the leakage area is evenly distributed among the four sets of vacuum breakers (i.e., four sets equate to 24% of the TS limit where each set is 6% and twice this total is 12% of the TS limit). This allows a leakage of less than or equal to 0.0865 in^2 for an individual set of vacuum breakers. This criterion is stipulated to identify individual sets of vacuum breakers with a higher leakage area.

2.8 Summary

The staff has reviewed the information provided by the licensee in support of an application for amendment and has concluded that individual vacuum breaker leakage tests are an acceptable alternative to an integrated suppression pool bypass test during outages for which a Type A containment integrated leak rate test is not conducted. This conclusion is based on the licensee analyses of potential suppression pool bypass leakage paths. The analysis demonstrated that vacuum breakers are the predominant potential source of leakage and that the leakage for the other sources is conservatively accommodated by the margins included in the proposed TS. The proposed TS changes are therefore acceptable.

3.0 STATE CONSULTATION

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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 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 (59 FR 626). 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.

5.0 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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