October 2, 1990

Docket Nos.: 50-352
and 50-353

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Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Deks
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS (TAC NOS. 75104/75105)

RE: LIMERICK GENERATING STATION, UNITS 1 AND 2

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. NPF-39 and Amendment No. 9 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 October 11, 1989 as supplemented by your letter of April 9, 1990.

These amendments revise the TSs to specify the number of suppression chamber to drywell vacuum breaker pairs which are required to be operable as three rather than four pairs.

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

Sincerely,

Original signed by
Richard J. Clark

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

Enclosures:
1. Amendment No. 46 to
License No. NPF-39
Amendment No. 9 to
License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

[GAH]

PDI-2/LA	PDI-2/PM	SPLB	OGC	PDI-2/D
MO'Brien	RClark:cf	CMcCracken		WButler
/90	9/26/90	9/28/90	09/21/90	10/1/90

R.Anand 9/27/90

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

October 2, 1990

Docket Nos.: 50-352
and 50-353

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
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SUBJECT: SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS (TAC NOS. 75104/75105)

RE: LIMERICK GENERATING STATION, UNITS 1 AND 2

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These amendments revise the TSs to specify the number of suppression chamber to drywell vacuum breaker pairs which are required to be operable as three rather than four pairs.

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Richard J. Clark, Project Manager
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1. Amendment No. 46 to
License No. NPF-39
Amendment No. 9 to
License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

Limerick Generating Station
Units 1 & 2

cc:

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

Mr. Rod Krich 52A-5
Philadelphia Electric Company
955 Chesterbrook Boulevard
Wayne, Pennsylvania 19087-5691

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

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

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

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

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

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

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

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

Marty J. McCormick, Jr.
Plant Manager
Limerick Generating Station
P.O. Box A
Sanatoga, Pennsylvania 19464

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 October 11, 1989, as supplemented by letter dated April 9, 1990 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. 46 , 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

/S/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 2, 1990

PDI-2/LA MO'Brien / /90	PDI-2/PM RClark:tf 05/16/90	OGC 05/21/90	PDI-2/D WButler 10/1/90

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 2, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 46

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 6-43	3/4 6-43*
3/4 6-44	3/4 6-44
B 3/4 6-3	B 3/4 6-3*
B 3/4 6-4	B 3/4 6-4

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES (Continued)

21. Automatic isolation signal causes TIP to retract; ball valve closes when probe is fully retracted.

22. Isolation barrier remains water filled or a water seal remains in the line post-LOCA. Isolation valve may be tested with water. Isolation valve leakage is not included in 0.60 La total Type B & C tests.

23. Valve does not receive an isolation signal. Valves will be open during Type A test. Type C test not required.

24. Both isolation signals required for valve closure.

25. Deleted

26. Valve stroke times listed are maximum times verified by testing per Specification 4.0.5 acceptance criteria. The closure times for isolation valves in lines in which high-energy line breaks could occur are identified with a single asterisk. The closure times for isolation valves in lines which provide an open path from the containment to the environs are identified with a double asterisk.

27. The reactor vessel head seal leak detection line (penetration 29A) excess flow check valve is not subject to OPERABILITY testing. This valve will not be exposed to primary system pressure except under the unlikely conditions of a seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source; therefore, this valve need not be OPERABILITY tested.

28. (DELETED)

29. Valve may be open during normal operation; capable of manual isolation from control room. Position will be controlled procedurally.

30. Valve normally open, closes on scram signal.

31. Valve 41-1016 is an outboard isolation barrier for penetrations X-9A, B and X-44. Leakage through valve 41-1016 is included in the total for penetration X-44 only.

32. Feedwater long-path recirculation valves are sealed closed whenever the reactor is critical and reactor pressure is greater than 600 psig. The valves are expected to be opened only in the following instances:

    a. Flushing of the condensate and feedwater systems during plant startup.

    b. Reactor pressure vessel hydrostatic testing, which is conducted following each refueling outage prior to commencing plant startup.

    Therefore, valve stroke timing in accordance with Specification 4.0.5 is not required.

33. Valve also constitutes a Unit 2 Reactor Enclosure Secondary Containment Automatic Isolation Valve and a Refueling Area Secondary Containment Automatic Isolation Valve as shown in Table 3.6.5.2.1-1 and Table 3.6.5.2.2-1 respectively.

JUN 14 1989
Effective when OL is issued to Unit 2

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Three pairs of suppression chamber - drywell vacuum breakers shall be OPERABLE and all suppression chamber - drywell vacuum breakers shall be closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more vacuum breakers in one of the three required pairs of suppression chamber - drywell vacuum breaker pairs inoperable for opening but known to be closed, restore at least one inoperable pair of vacuum breakers 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 one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable:

   1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.7 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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.

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. 9
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 October 11, 1989, as supplemented by letter dated April 9, 1990 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. 9 , 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

/S/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 2, 1990

PDI-2/LA W Butler / /90	PDI-2/PM RClark:tr 05/16/90	OGC 05/21/90	PDI-2/D WButler 10/1/90

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 2, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 9

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 6-43	3/4 6-43*
3/4 6-44	3/4 6-44
B 3/4 6-3	B 3/4 6-3*
B 3/4 6-4	B 3/4 6-4

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES (Continued)

21. Automatic isolation signal causes TIP to retract; ball valve closes when probe is fully retracted.

22. Isolation barrier remains water filled or a water seal remains in the line post-LOCA. Isolation valve may be tested with water. Isolation valve leakage is not included in 0.60 La total Type B & C tests.

23. Valve does not receive an isolation signal. Valves will be open during Type A test. Type C test not required.

24. Both isolation signals required for valve closure.

25. Deleted

26. Valve stroke times listed are maximum times verified by testing per Specification 4.0.5 acceptance criteria. The closure times for isolation valves in lines in which high-energy line breaks could occur are identified with a single asterisk. The closure times for isolation valves in lines which provide an open path from the containment to the environs are identified with a double asterisk.

27. The reactor vessel head seal leak detection line (penetration 29A) excess flow check valve is not subject to OPERABILITY testing. This valve will not be exposed to primary system pressure except under the unlikely conditions of a seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source; therefore, this valve need not be OPERABILITY tested.

28. (DELETED)

29. Valve may be open during normal operation; capable of manual isolation from control room. Position will be controlled procedurally.

30. Valve normally open, closes on scram signal.

31. Valve 41-2016 is an outboard isolation barrier for penetrations X-9A, B and X-44. Leakage through valve 41-2016 is included in the total for penetration X-44 only.

32. Feedwater long-path recirculation valves are sealed closed whenever the reactor is critical and reactor pressure is greater than 600 psig. The valves are expected to be opened only in the following instances:

    a. Flushing of the condensate and feedwater systems during plant startup.

    b. Reactor pressure vessel hydrostatic testing, which is conducted following each refueling outage prior to commencing plant startup.

    Therefore, valve stroke timing in accordance with Specification 4.0.5 is not required.

33. Valve also constitutes a Unit 1 Reactor Enclosure Secondary Containment Automatic Isolation Valve and a Refueling Area Secondary Containment Automatic Isolation Valve as shown in Table 3.6.5.2.1-1 and Table 3.6.5.2.2-1, respectively.

34. Isolation signal causes recombiner to trip; valve closes when recombiner is not operating.

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Three pairs of suppression chamber - drywell vacuum breakers shall be OPERABLE and all suppression chamber - drywell vacuum breakers shall be closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more vacuum breakers in one of the three required pairs of suppression chamber - drywell vacuum breaker pairs inoperable for opening but known to be closed, restore at least one inoperable pair of vacuum breakers 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 one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable:

   1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.7 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 46 AND 9 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 October 11, 1989, and supplemented by letter dated April 9, 1990, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2. These proposed amendments would change the Technical Specifications (TSs) to specify the number of suppression chamber to drywell vacuum breaker pairs which are required to be operable as three rather than four pairs.

2.0 DISCUSSION

There are four pairs of vacuum breaker valves provided to equalize the pressure between the suppression chamber and the drywell after reactor blowdown and drywell spray actuation, while preventing bypass of the suppression pool during periods of blowdown. Previous analysis indicated that three of the four pairs were required to provide adequate vacuum relief capability to protect the structural integrity of the containment for all postulated events. The fourth pair provided redundancy in the event that a single active failure prevented one valve in any of the three required valve pairs from opening. A reanalysis performed by the licensee has determined that two pairs, rather than three pairs, of vacuum breaker valves are adequate to protect the structural integrity of the containment. Therefore, the licensee proposes to revise TS Limiting Condition for Operation (LCO) 3.6.4.1 to require a minimum of three pairs of operable vacuum breakers.

3.0 EVALUATION

As discussed in Section 6.2 of the NRC's Safety Evaluation Report (NUREG-0991) for the Limerick Generating Station (LGS), the containment systems for Limerick Units 1 and 2 include the Mark II pressure suppression containment structure (primary containment), the secondary containment structure and supporting systems, the containment heat removal system, the containment isolation system, and the combustible gas

control system. The primary and secondary containment structures and associated containment systems function to prevent or control the release of radioactive material that might be released into the containment atmosphere following a postulated loss-of-coolant accident (LOCA) or fuel handling accident.

The primary containment is in the form of a truncated cone over a cylindrical section, with the drywell the upper conical section and the suppression chamber the lower cylindrical section. These two sections comprise a structurally integrated, reinforced concrete pressure vessel, lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. The drywell and suppression chamber are divided by a horizontal diaphragm slab of reinforced concrete structurally connected to the containment wall.

As noted previously, the vacuum relief valves are provided to equalize the pressure between the drywell and suppression chamber following blowdown. As discussed in Section 6.2.1.4 of NUREG-0991, the vacuum breakers are primarily sized to prevent excessive drywell floor reverse pressure (i.e., suppression chamber pressure greater than drywell pressure) and to prevent excessive negative pressure in the drywell such as might result from the inadvertent actuation of a drywell spray train during a postulated accident.

The LGS primary containment design values that we are primarily concerned with in evaluating the capacity of the vacuum breakers are the following:

a) Design differential pressure across the diaphragm slab in the upward direction = 20 psid.

b) Design (negative) pressure of the primary containment with respect to the secondary containment = -5 psig.

To ensure that these design values will not be exceeded, vacuum breakers have been provided between the drywell and the suppression chamber (or wetwell). Four flow paths with two vacuum breaker valves in series on each flow path are provided. The valves are set so that a differential pressure of greater than 1 psid between the suppression chamber and the drywell will result in flow from the wetwell to the drywell to equalize the pressure to within 1 psid.

Events which have the potential to result in these design allowables being exceeded are discussed in the LGS Final Safety Analysis Report (FSAR) "Containment Systems," Sections 6.2.1.1.3, 6.2.1.1.4, and 6.2.1.1.5. The vacuum breaker valves may also serve to relieve a pressure differential between the wetwell and the drywell during containment purge operations and hydrogen recombiner operation. As stated in the FSAR, inadvertent actuation of the drywell spray system following a Loss of Coolant Accident (LOCA) was determined to pose the most severe challenge to the diaphragm slab upward differential pressure and primary containment negative pressure design values.

The initial analysis performed to verify the adequacy of the vacuum breaker sizing was based on highly conservative assumptions. One such assumption was that the upward differential pressure across the diaphragm slab should not exceed 3 psid. Additionally, since valve test data was not available at that time, conservative flow assumptions were used for the vacuum breakers. Based on these assumptions, three flow paths (i.e., three vacuum breaker valve pairs) were determined to be required to maintain the differential pressure below the assumed design value of 3 psid. The fourth flow path provided a redundant flow path in the event that one of the other three flow paths was inoperable as a result of a single active failure which prevented a flow path from performing its intended function.

The initial analysis was followed by a computer analysis incorporating flow test data from the valve vendor for the actual valves in the as-built configuration, rather than assumed flow data. The purpose of this computer analysis, however, was not to determine the number of flow paths required, but to confirm that three operable flow paths would be adequate to prevent the drywell from exceeding the -5 psig design value in the event of the postulated inadvertent post reactor blowdown drywell spray actuation. Three flow paths were found to be adequate for this purpose. The maximum differential pressure across the diaphragm slab in this case was determined to be 4.26 psid, well below the 20 psid design value.

The 3 psid diaphragm slab differential pressure used in the initial calculation is not a required design basis value, but was arbitrarily chosen as a value to use while performing the determination of the number of required vacuum breaker valve pairs. Since the 20 psid design differential pressure value must not be exceeded and the 3 psid value was arbitrary, the fact that the actual differential pressure exceeds 3 psid in the more accurate computer calculations is of no consequence.

Recently, the computer analysis was performed again by the licensee utilizing two flow paths instead of three. The analysis showed two flow paths to be sufficient to avoid exceeding the -5 psig design value. A review of the previous analysis (i.e., using three flow paths) showed that the condensation rate in the drywell is the parameter controlling the resulting peak negative pressure reached. The flow rate through the vacuum breaker valves is not the limiting parameter since the valves are not required to fully open during the event to provide the necessary vacuum relief. Essentially the same peak negative pressure is reached in the drywell for any number of flow paths greater than two. The flow rate through the vacuum breakers only becomes controlling when less than two flow paths are available. Hence, with the valves full open, two flow paths are sufficient to provide adequate vacuum relief.

FSAR Section 6.2.1.1.4 notes that if both trains of drywell spray were to be actuated concurrently, in violation of existing plant procedure, the drywell design negative pressure of -5 psig could be exceeded, if the suppression pool temperature is below 105°F. With only two vacuum breaker flow paths operable instead of three, the suppression pool temperature below which the -5 psig design pressure could be exceeded, if both spray trains were actuated concurrently, will be somewhat higher. Since, as discussed in the FSAR "Response to NRC Questions," question 480.4, drywell spray actuation is under strict administrative controls, and concurrent actuation of both spray trains is in violation of plant procedures, this increase in suppression pool temperature below which concurrent spray train actuation could result in exceeding the -5 psig design pressure is still of no consequence, and does not constitute any actual reduction in a margin of safety.

The licensee performed an evaluation of the proposed changes to determine if an unreviewed safety question exists. The evaluation concluded that the proposed change does constitute an unreviewed safety question. This results from the fact that the reduction of required flow paths does decrease the margin of safety as defined in TS Section 3/4 6.4. TS LCO 3.6.4.1 presently requires the operability of four vacuum breaker flow paths. If three of the vacuum breaker pairs operate, the primary containment design values will not be exceeded. Calculation has shown that even if only two vacuum breaker pairs operate, the primary containment design values still will not be exceeded. However, there will be a small increase (from -4.821 psig to -4.845 psig) in the magnitude of the drywell peak negative pressure in the event of the postulated drywell depressurization, even though this value will still be within the -5 psig design primary containment pressure limit. There will also be a small increase (from 4.26 psid to 5.77 psid) in the maximum upward differential pressure developed across the diaphragm slab. This value is still within the 20 psid design differential pressure. Although the resulting drywell negative pressure and diaphragm slab differential pressure are acceptable, they still constitute a small reduction in the margin of safety since they are slightly closer to the design values than for the three vacuum breaker flow path case.

As noted above, the postulated inadvertent activation of a drywell spray by an operator during a small break LOCA was the design basis accident transient resulting in the most rapid condensation of steam in the drywell and thus the maximum differential pressure between the drywell and wetwell. In the submittal of October 11, 1989, the licensee provided the results of this transient analysis but not the detailed analysis and data used in the calculations (e.g., spray water temperature, valve opening times, flow characteristics of valves, etc.). The transient analysis was discussed in a telecom with the licensee on December 15, 1989. The licensee was requested to provide the transient analysis. The analysis was provided by the licensee's letter of April 9, 1990. The letter provided analysis to support the results in the October 11, 1990

application and additional justification for the proposed changes to the TSs. The additional information strengthened but in no way changed the staff's proposed No Significant Hazards Consideration Determination.

We have reviewed the licensee's reanalyzes of the postulated events leading to potentially rapid drywell depressurization with respect to the wetwell and find them conservative. The licensee has demonstrated that two operable flow paths are adequate to prevent exceeding containment design values during the postulated events. Requiring three vacuum breaker flow paths to be operable meets the staff's single active failure criteria. Therefore, the proposed change in the TSs to reduce the number of suppression chamber to drywell vacuum breaker pairs which are required to be operable from four to three pairs is acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that these 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 47607) on November 15, 1989 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and the security nor to the health and safety of the public.

Principal Contributors: R. Anand, R. Clark

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