January 24, 1997

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M96117 AND M96118)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No.118 to Facility Operating License No. NPF-39 and Amendment No. 81 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 June 28, 1996, as supplemented by letters dated November 4 and 5, and December 9, 1996.

These amendments revise the TSs to incorporate performance-based testing, in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors," Option B. This option allows utilities to extend the frequencies of the Type A Containment Leak Rate Test (ILRT), and Type B and C Local Leak Rate Tests (LLRTs) based on the performance of the containment and components.

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

Docket Nos. 50-352/353

Enclosures: 1. Amendment No.118 to License No. NPF-39
2. Amendment No. 81 to License No. NPF-85
3. Safety Evaluation

cc w/encls: See next page

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OFFICE	PDI-2/PM	PDI-2/LA	OGC*	PDI-2/D	BC:SCSB*
NAME	FRinaldi:rb	MO'Brien		JStolz	CBerlinger
DATE	1/23/97	1/2/97	01/22/97	1/23/97	01/15/97

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OFFICIAL RECORD COPY DOCUMENT NAME: LI96117.AMD

January 24, 1997

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M96117 AND M96118)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No.118 to Facility Operating License No. NPF-39 and Amendment No. 81 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 June 28, 1996, as supplemented by letters dated November 4 and 5, and December 9, 1996.

These amendments revise the TSs to incorporate performance-based testing, in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors," Option B. This option allows utilities to extend the frequencies of the Type A Containment Leak Rate Test (ILRT), and Type B and C Local Leak Rate Tests (LLRTs) based on the performance of the containment and components.

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Docket Nos. 50-352/353

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3. Safety Evaluation

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OFFICE	PDI-2/PM	PDI-2/LA	OGC*	PDI-2/D	BC:SCSB*
NAME	FRinaldi:rb	MO'Brien		JStolz	CBerlinger
DATE	1/23/97	1/21/97	01/22/97	1/23/97	01/15/97

OFFICIAL RECORD COPY DOCUMENT NAME: LI96117.AMD

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

January 24, 1997

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M96117 AND M96118)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 118to Facility Operating License No. NPF-39 and Amendment No. 81 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 June 28, 1996, as supplemented by letters dated November 4 and 5, and December 9, 1996.

These amendments revise the TSs to incorporate performance-based testing, in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors," Option B. This option allows utilities to extend the frequencies of the Type A Containment Leak Rate Test (ILRT), and Type B and C Local Leak Rate Tests (LLRTs) based on the performance of the containment and components.

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

Sincerely,

Frank Rinaldi

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

Docket Nos. 50-352/353

Enclosures:
1. Amendment No.118 to License No. NPF-39
2. Amendment No. 81 to License No. NPF-85
3. Safety Evaluation

cc w/encls: See next page

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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. 118
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 June 28, 1996, as supplemented by letters dated November 4 and 5, and December 9, 1996, 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 License Condition 2.D. on page 7 of Facility Operating License No. NPF-39*, and the license is also 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. 118 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1. Page 7 of License No. NPF-39
2. Changes to the Technical Specifications

Date of Issuance: January 24, 1997

*Page 7 is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the License and 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.

1. Revise License as follows:

Remove Page	Insert Page
7	7

2. Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-8	3/4 6-8
3/4 6-14	3/4 6-14
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
-	6-14c

(14) Refueling Floor Volume Connection to Standby Gas Treatment System (Section 6.2.3, SSER-2 and SSER-3)

Prior to any movement of irradiated fuel within the refueling floor volume the licensee shall complete and test all modifications required to connect the refueling floor volume to standby gas treatment system. During the interim period, the licensee shall not remove the reactor pressure vessel head prior to the NRC staff review and approval.

(15) Emergency Planning

Procedures Subject to 44 CFR Part 350

In the event the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50. These include (a) exemption from General Design Criteria (GDC) 61 of Appendix A, operation of that portion of the standby gas treatment system (SGTS) that serves the refueling area until the first refueling (Section 6.2.3 of SSER-2 and SSER-3), (b) exemption from GDC-56 of Appendix A, the requirement for additional automatic containment isolation valves for the hydrogen recombiner lines and the requirement for automatic isolation of existing isolation valves in the Drywell Chilled Water (DCW) and the Reactor Enclosure Cooling Water (RECW) systems until prior to startup following the first refueling outage (Section 6.2.4.2 of the SER, SSER-1 and SSER-3), (c) exemption from GDC-19 of Appendix A, as related to the requirement for redundant remote shutdown capability (Section 7.4.2.3 of SSER-3 and SSER-5), (d) exemption from the requirement of Appendix J, the testing of containment air locks at times when the containment integrity is not required (Section 6.2.6.1 of the SER and SSER-3), (e) exemption from the requirements of Appendix J, the leak rate testing of the Main Steam Isolation Valves (MSIVs) at the peak calculated containment pressure, Pa, and exemption from the requirements of Appendix J that the measured MSIV leak rates be included in the summation for the local leak rate test (Section 6.2.6 of SSER-3), (f) exemption from the requirement of Appendix J, the local leak rate testing of the Traversing Incore Probe Shear Valves (Section 6.2.6 of the SER and SSER-3), (g) a one-time exemption from the requirement of Appendix J to perform local leak

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying the primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

* See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A Test) in accordance with the Primary Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Primary Containment Leakage Rate Testing Program for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests.

c. *Less than or equal to 100 scf per hour through any one main steam isolation valve not to exceed 200 scf per hour for all four main steam lines, when tested at P_t, 22.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 48.4 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate (Type A Test) exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program, or

c. The measured leakage rate exceeding 100 scf per hour through any one main steam isolation valve, or exceeding 200 scf per hour for all four main steam lines, or

d. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) (Type A Test) to be in accordance with the Primary Containment Leakage Rate Testing Program, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR O ATION (Continued)

ACTION: (Continued)

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to be in accordance with the Primary Containment Leakage Rate Testing Program, and

c. The leakage rate to $\leq$11.5 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to $\leq$200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing the reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

* Exemption to Appendix "J" to 10 CFR Part 50.

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PRIMARY CONTAINMENT AIR LOCK

LIMITING CONDITION FOR OPERATION

3.6.1.3 The primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Primary Containment Leakage Rate Testing Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment air lock shall be demonstrated OPERABLE:

a. By verifying the seal leakage rate is in accordance with the the Primary Containment Leakage Rate Testing Program.

b. By conducting an overall air lock leakage test in accordance with the Primary Containment Leakage Rate Testing Program.

c. At least once per 6 months by verifying that only one door in the air lock can be opened at a time.***

*** Except that the airlock doors need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the airlock doors' interlock is tested within 8 hours after the primary containment has been deinerted and provided the shield door to the airlock is maintained locked closed.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined by a visual inspection of those surfaces. This inspection shall be performed in accordance with the Primary Containment Leakage Rate Testing Program.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the liner and concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

c. By verifying 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 24 months,

with the temperature alarm setpoint for:

1. High water temperature:

a) First setpoint ≤ 95°F

b) Second setpoint ≤ 105°F

c) Third setpoint ≤ 110°F

d) Fourth setpoint ≤ 120°F

d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 92 days, and

3. CHANNEL CALIBRATION at least once per 24* months,

with the water level alarm setpoint for high water level ≤ 24'1-1/2"

e. Drywell-to-suppression chamber bypass leak tests shall be conducted to coincide with the Type A test 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.

f. 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-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted.

* The CHANNEL CALIBRATION for level transmitters LT-55-1N062B, -1N062F shall be performed at least once per 18 months.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value calculated in the safety analyses for the design basis LOCA maximum peak containment pressure of ≤ 44 psig, P_a. As an added conservatism, the measured overall integrated leakage rate (Type A Test) is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the Primary Containment Leakage Rate Testing Program.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCK

The limitations on closure and leak rate for the primary containment air lock are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the Primary Containment Leakage Rate Testing Program. Only one closed door in the air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE ALTERNATE DRAIN PATHWAY

Calculated doses resulting from the maximum leakage allowances for the main steamline isolation valves in the postulated LOCA situations will not exceed the criteria of 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been continuously maintained. The requirement for the MSIV Leakage Alternate Drain Pathway serves to reduce the offsite dose.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum calculated pressure in the event of a LOCA. A visual inspection in accordance with the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the calculated containment peak pressure does not exceed the design pressure of 55 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5.0 psid. The limit of - 1.0 to + 2.0 psig for initial containment pressure will limit the total pressure to ≤ 44 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, deinerting and pressure control. The 180 hours per 365 day limit on purge valve operation is imposed to protect the integrity of the SGTS filters. Analysis indicates that should a LOCA occur while this pathway is being utilized, the associated pressure surge through the (18 or 24") purge lines will adversely affect the integrity of SGTS. This limit is not imposed, however, on the subject valves when pressure control is being performed through the 2-inch bypass line, since a pressure surge through this line does not threaten the OPERABILITY of SGTS.

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163 "Performance-Based Containment Leakage Test program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

   2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

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. 81
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 June 28, 1996, as supplemented by letters dated November 4 and 5, and December 9, 1996, 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 License Condition 2.D. on page 4 of Facility Operating License No. NPF-85*, and the license is also 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. 81 , are hereby incorporated in the license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implement within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1. Page 4 of License No. NPF-85
2. Changes to the Technical Specifications

Date of Issuance: January 24, 1997

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*Page 4 is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 81

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the License and 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.

1. Revise License as follows:

Remove Page	Insert Page
4	4

2. Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-8	3/4 6-8
3/4 6-14	3/4 6-14
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
-	6-14c

(4) Physical Security and Safeguards

The licensee shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Limerick Generating Station, Units 1 & 2, Physical Security Plan," with revisions submitted through October 31, 1988; "Limerick Generating Station, Units 1 & 2, Plant Security Personnel Training and Qualification Plan," with revisions submitted through October 1, 1985; and "Limerick Generating Station, Units 1 & 2, Safeguards Contingency Plan," with revisions submitted through November 15, 1986.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) exemption from the requirement of Appendix J, the testing of containment air locks at times when the containment integrity is not required (Section 6.2.6.1 of the SER and SSER-3), (b) exemption from the requirements of Appendix J, the leak rate testing of the Main Steam Isolation Valves (MSIVs) at the peak calculated containment pressure, Pa, and exemption from the requirements of Appendix J that the measured MSIV leak rates be included in the summation for the local leak rate test (Section 6.2.6.1 of SSER-3), (c) exemption from the requirement of Appendix J,

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying the primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

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* See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A Test) in accordance with the Primary Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Primary Containment Leakage Rate Testing Program for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests.

c. *Less than or equal to 100 scf per hour through any one main steam isolation valve not to exceed 200 scf per hour for all four main steam lines, when tested at P_t, 22.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 48.4 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate (Type A Test) exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program, or

c. The measured leakage rate exceeding 100 scf per hour through any one main steam isolation valve, or exceeding 200 scf per hour for all four main steam lines, or

d. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) (Type A Test) to be in accordance with the Primary Containment Leakage Rate Testing Program, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to be in accordance with the Primary Containment Leakage Rate Testing Program, and

c. The leakage rate to ≤11.5 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to ≤200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

*Exemption to Appendix "J" to 10 CFR Part 50.

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PRIMARY CONTAINMENT AIR LOCK

LIMITING CONDITION FOR OPERATION

3.6.1.3 The primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Primary Containment Leakage Rate Testing Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment air lock shall be demonstrated OPERABLE:

a. By verifying the seal leakage rate is in accordance with the Primary Containment Leakage Rate Testing Program.

b. By conducting an overall air lock leakage test in accordance with the Primary Containment Leakage Rate Testing Program.

c. At least once per 6 months by verifying that only one door in the air lock can be opened at a time.***

***Except that the airlock doors need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the airlock doors' interlock is tested within 8 hours after the primary containment has been deinerted and provided the shield door to the airlock is maintained locked closed.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined by a visual inspection of those surfaces. This inspection shall be performed in accordance with the Primary Containment Leakage Rate Testing Program.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the liner and concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

c. By verifying 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 24 months,

with the temperature alarm setpoint for:

1. High water temperature:

   a) First setpoint ≤ 95°F

   b) Second setpoint ≤ 105°F

   c) Third setpoint ≤ 110°F

   d) Fourth setpoint ≤ 120°F

d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 92 days, and

3. CHANNEL CALIBRATION at least once per 24* months,

with the water level alarm setpoint for high water level ≤ 24'1-1/2"

e. Drywell-to-suppression chamber bypass leak tests shall be conducted to coincide with the Type A test at an initial differential pressure of 4 psi and verifying that the $A/\sqrt{k}$ calculation 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.

f. 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.e is not conducted.

* The CHANNEL CALIBRATION for level transmitters LT-55-2N062B, -2N062F shall be performed at least once per 18 months.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value calculated in the safety analyses for the design basis LOCA maximum peak containment accident pressure of 44 psig, P_a. As an added conservatism, the measured overall integrated leakage rate (Type A Test) is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the Primary Containment Leakage Rate Testing Program.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCK

The limitations on closure and leak rate for the primary containment air lock are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the Primary Containment Leakage Rate Testing Program. Only one closed door in the air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE ALTERNATE DRAIN PATHWAY

Calculated doses resulting from the maximum leakage allowances for the main steamline isolation valves in the postulated LOCA situations will not exceed the criteria of 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been continuously maintained. The requirement for the MSIV Leakage Alternate Drain Pathway serves to reduce the offsite dose.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum calculated pressure in the event of a LOCA. A visual inspection in accordance with the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the calculated containment peak pressure does not exceed the design pressure of 55 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5.0 psid. The limit of - 1.0 to + 2.0 psig for initial containment pressure will limit the total pressure to ≤ 44 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, deinerting and pressure control. The 180 hours per 365 day limit on purge valve operation is imposed to protect the integrity of the SGTS filters. Analysis indicates that should a LOCA occur while this pathway is being utilized, the associated pressure surge through the (18 or 24") purge lines will adversely affect the integrity of the SGTS. This limit is not imposed, however, on the subject valves when pressure control is being performed through the 2-inch bypass line, since a pressure surge through this line does not threaten the OPERABILITY of SGTS.

PROCEDURES AND PROGRAMS (Continued)

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163 "Performance-Based Containment Leakage Test program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.0 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall airlock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a.

   2) Seal leakage rate is less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the tests described in the Primary Containment Leakage Rate Testing Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 118 AND 81 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 June 28, 1996, as supplemented November 4 and 5, and December 9, 1996, the Philadelphia Electric Company (the licensee) submitted a request for changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications (TSs). The requested changes would revise the TSs to incorporate performance based testing, in accordance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors," Option B. This option allows utilities to extend the frequencies of the Type A Containment Leak Rate Test (ILRT), and Type B and C Local Leak Rate Tests (LLRTs) based on the performance of the containment and components. The November 4 and 5, and December 9, 1996, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or the Federal Register notice.

By letter dated November 30, 1993, the licensee had previously proposed an extension of the test interval for the drywell bypass leakage rate test from 18 months to coincide with the Appendix J Option A Type A test interval, which the Limerick TSs specify as 40 ± 10 months. The staff approved the licensee's request in a letter dated February 17, 1994. As part of the licensee's June 28, 1996 submittal, the licensee proposed maintaining the requirement to perform drywell bypass leakage rate tests at the same interval as the Appendix J Type A test. However, Option B would permit, under specific conditions, increasing the Type A test interval from 40 ± 10 months to 10 years. This issue is also addressed in this safety evaluation report. The proposal to extend the test interval for drywell bypass leakage rate testing was supplemented by submittals dated November 4, and 5, 1996, and December 9, 1996.

2.0 BACKGROUND

2.1 Appendix J Option B

The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TSs. The program references

Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," which specifies a method acceptable to the NRC for complying with Option B dated September 1995.

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the *Federal Register* (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the *Federal Register* on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TSs. The licensee has referenced RG 1.163 in the Limerick Unit 1 and Unit 2 TSs.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

2.2 Drywell Bypass Leakage Rate Testing

During a postulated loss-of-coolant accident (LOCA) inside containment, the drywell is pressurized with steam and air. The resulting large pressure difference between the drywell and the wetwell forces the steam through the suppression pool where it is condensed, resulting in a lower containment pressure. If the steam were to bypass the suppression pool and pressurize the wetwell, containment design pressure may be exceeded. Consequently, a test is performed to ensure that the leakage between the drywell and the wetwell is less than a specified amount. The leakage is specified as $A/\sqrt{K}$, where A is the flow area of the leakage path and K is the geometric and frictional loss coefficient. For Limerick Units 1 and 2 the design value of $A/\sqrt{K}$ is 0.05 ft^2. The technical specifications limit is 10% of this value for conservatism.

The licensee's November 30, 1993 submittal provided a technical basis for extending the test interval from 18 months to the same test interval as the Appendix J Type A test. This interval is 40 ± 10 months. As part of the licensee's proposal, a vacuum breaker leakage rate test was required to be performed during those refueling outages when the drywell bypass leakage test was not performed. As discussed above, Appendix J Option B would allow the Type A test interval to be extended to 10 years. Therefore, the staff reviewed the licensee's November 30, 1993 submittal again and requested additional information to determine the acceptability of the 10-year drywell bypass leakage rate test interval. Our evaluation is given below.

3.0 EVALUATION

3.1 Evaluation of Licensee's Proposal to Adopt Appendix J Option B for Containment Leakage rate testing

The licensee's June 28, 1996, letter to the NRC proposes to establish a "Containment Leakage Rate Testing Program" and proposes to add this program to the TSs. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires a change to existing Limerick Unit 1 and Unit 2 TS Sections 4.6.1.1, 3.6.1.2, 4.6.1.2, 3.6.1.3, 4.6.1.3, 4.6.1.5.1, 4.6.2.1 and the addition of the "Containment Leakage Rate Testing Program" to Section 6.8.4.g. Corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

The licensee's June 28, 1996 submittal discusses three exemptions from Appendix J. TS LCO 3.6.1.2.b identifies an exemption which permits MSIV leakage to be considered separately from other Type C leakage and for testing the MSIVs at a lower pressure than that specified in Appendix J. Pursuant to paragraph V.B.1 of Option B, this exemption will be retained with editorial changes to reflect its continued applicability. TS surveillance requirement 4.6.3.5 identifies a Traversing Incore Probe (TIP) system explosive isolation valve surveillance test which is performed in place of an Appendix J Type C test. Pursuant to pararaph V.B.1 of Option B, this exemption will also be retained. TS surveillance requirement 4.6.1.3.a.2 identifies an exemption to the Appendix J requirements for air lock leakage rate testing. Appendix J Option A states: "Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a." The exemption allows a 10 psig leakage rate test of containment air lock seals prior to establishing primary containment integrity when the air lock has been used and no maintenance has been performed on the air lock that could affect its sealing capability. This exemption is no longer required since Option B does not specify detailed requirements for air lock testing. The details for this test are now included in NEI 94-01 Revision 0 which is endorsed by RG 1.163. However, the licensee requests that the exemption be retained in the Limerick Generating Station Unit 1 and Unit 2 licenses in order to document that the test is performed at a pressure lower than P_a (10 psig). The licensee has included the air lock test pressure in TS Section 6.8.4.g, the Primary Containment Leakage Rate Testing Program, as specified by Section 10.2.2.1 of NEI 94-01, Revision 0.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TSs of the November 2, 1995, letter and are, therefore, acceptable to the staff.

3.2 Evaluation of Licensee's proposal to Extend Drywell Bypass Test Interval from 40 ± 10 months to 10 years.

The most probable leakage paths between the drywell and the wetwell of a BWR Mark II containment are through the four sets of vacuum breakers. The licensee's November 30, 1993 submittal proposed adding a surveillance requirement to leak test this leakage path during those refueling outages when the drywell bypass leakage rate test was not performed. The staff approved this proposal in a safety evaluation report which was transmitted to the licensee by letter dated February 17, 1994. The licensee has not performed any drywell bypass leakage tests since that time. The licensee has performed vacuum breaker leakage rate tests. In a letter dated December 9, 1996, the licensee stated that the measured leakage rates have been below the acceptance criteria specified in TS 4.6.2.1.f.

The other leakage paths are diaphragm floor penetrations such as the downcomer and main steam safety/relief valve (SRV) discharge line penetrations, cracks in the diaphragm floor and liner plate and cracks in the downcomers and SRV discharge lines that pass through the suppression chamber air space. Isolation valves in lines which are cross-connected between the drywell and the wetwell are another possible leakage path. The staff's February 17, 1994 safety evaluation report considered all these flow paths and found that an extension in the test interval would not result in a significant increase in the leakage through any of these leakage paths. The staff has again reexamined these leakage paths and concludes that a further extension of the test interval to 10 years would not result in significant leakage through any of these paths. In addition to the vacuum breaker tests which are performed at all refueling outages during which a drywell bypass leakage rate test is not performed, the licensee indicated in the November 4, 1996 letter in response to a staff question that the liner plate over the diaphragm slab is visually inspected at every refueling outage.

In addition, the staff requested that the licensee assess any possible increases in risk associated with the increased interval. The licensee responded in the November 4, 1996 letter to the staff by outlining the actions that could be taken according to the emergency operating procedures if an increase in containment drywell pressure were to occur. The operator would be directed to use wetwell and drywell sprays or to depressurize the reactor vessel if these sprays were to be unavailable. In addition, the licensee pointed out that risk studies for Limerick have demonstrated that the primary contributors to containment overpressure failure are due to failure of the vacuum breakers and failure of decay heat removal equipment. The frequency of loss of vapor suppression is only a minor contributor to the containment overpressure failure frequency.

The staff therefore finds the licensee's proposal to increase the drywell bypass test interval to 10 years to be acceptable.

The licensee proposed some minor editorial changes to the TSs in a telephone conference dated January 9, 1997, that did not change the context of the TSs and, therefore, are acceptable to the staff.

4.0 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.

5.0 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 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 55038). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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