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. 131
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 February 25, 1997, as supplemented by letters dated September 8 and November 18, 1997 and January 8 and July 2, 1998, 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, pages 1 through 5 and 7 and 8 of Facility Operating License No. DPR-39 are hereby amended by changing Philadelphia Electric Company to PECO Energy Company.

3. Further, 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. 131 , are hereby incorporated into this license. PECO Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 1 through 5 and 7 and 8 of Facility Operating License NPF-39
2. Changes to the Technical Specifications
3. Cover page and page 4-4 of the Environmental Protection Plan
4. Appendix C, Additional Conditions, Page 1

Date of Issuance: October 23, 1998

*Pages 1 through 5, and 7 and 8, and are attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Facility Operating License (FOL), the Appendix A Technical Specifications, the Appendix B Environmental Protection Plan (EPP), and Appendix C Additional Conditions, with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

	Remove	Insert
FOL	1	1
	2	2
	3	3
	4	4
	5	5
	6	-
	7	7
	8	8
Appendix A	x	x
	xix	xix
	3/4 3-8	3/4 3-8
	3/4 3-57	3/4 3-57
	3/4 3-103	3/4 3-103
	3/4 5-1	3/4 5-1
	3/4 5-2	3/4 5-2
	3/4 6-15	3/4 6-15
	3/4 6-16	3/4 6-16
	3/4 6-42	3/4 6-42
	3/4 6-43	3/4 6-43
	3/4 7-1	3/4 7-1
	3/4 7-3	3/4 7-3
	3/4 8-15	3/4 8-15
	6-8	6-8
	6-9	6-9
Appendix B	EPP Cover	EPP Cover
	EPP 4-4	EPP 4-4
Appendix C	Additional Conditions Page 1	Appendix C Additional Conditions Page 1

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

PECO ENERGY COMPANY
DOCKET NO 50-352
LIMERICK GENERATING STATION, UNIT 1
FACILITY OPERATING LICENSE

License No. NPF-39

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

   A. The application for license filed by PECO Energy Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Limerick Generating Station, Unit 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-106 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. The licensee is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Facility Operating License No. NPF-39, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings, the Partial Initial Decisions issued by the Atomic Safety and Licensing Board dated March 8, 1983, August 29, 1984, May 2, 1985 and July 22, 1985, and the Decision of the Appeal Board dated September 26, 1984, regarding this facility, and approval by the Nuclear Regulatory Commission in its Memorandum and Order dated August 8, 1985, the license for Fuel Loading and Low Power Testing, License No. NPF-27, issued on October 26, 1984, is superseded by Facility Operating License NPF-39 hereby issued to the PECO Energy Company (the licensee), to read as follows:

A. This license applies to the Limerick Generating Station, Unit 1, a boiling water nuclear reactor and associated equipment, owned by PECO Energy Company. The facility is located on the licensee's site in Montgomery and Chester Counties, Pennsylvania on the banks of the Schuylkill River approximately 1.7 miles southeast of the city limits of Pottstown, Pennsylvania and 21 miles northwest of the city limits of Philadelphia, Pennsylvania, and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report-Operating License Stage, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses PECO Energy Company:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Montgomery and Chester Counties, Pennsylvania, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PECO Energy Company is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. , are hereby incorporated in the license. PECO Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection (Section 9.5, SSER-2,-4)*

PECO Energy Company shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report dated August 1983 through Supplement 9, dated August 1989, and Safety Evaluation dated November 20, 1995, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

The Information contained on FOL pages 5 and 6
were intentionally omitted.

(16) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No.128 , are hereby incorporated into this license. PECO Energy Company shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50. 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 of SSER-3), (c) 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).

These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore these exemptions are hereby granted pursuant to 10 CFR 50.12 and 50.47(c). With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and 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 and 2, Physical Security Plan," with revisions submitted through December 7, 1987; "Limerick Generating Station, Units 1 and 2, Plant Security Personnel Training and Qualification Plan," with revisions submitted through October 1, 1985; and "Limerick Generating Station, Units 1 and 2, Safeguards Contingency Plan," with revisions submitted through November 15, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

G. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

INSTRUMENTATION (Continued)

BASES

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	Q	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) DELETED
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations' as performed per Specification 3.10.3.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition, provided at least one other operable channel in the same trip system is monitoring that parameter.

Amendment No. ~~70~~, 131

OFFGAS GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The offgas monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.2.5 and 3.11.2.6 respectively, are not exceeded.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

a. With an offgas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.12-1.

b. With less than the minimum number of offgas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Annual Radioactive Effluent Release Report.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each offgas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.12-1.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

1. Two OPERABLE CSS pumps, and

2. An OPERABLE flow patch capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of four subsystems with each subsystem comprised of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure coolant injection (HPCI) system consisting of:

1. One OPERABLE HPCI pump, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with at least five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* ** #, and 3* ** ##.

---

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

**The ADS is not required to be OPERABLE when the reactor steam dome pressure is less that or equal to 100 psig.

#See Special Test Exception 3.10.6.

##Two LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR Shutdown cooling permissive setpoint.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least two LPCI subsystems are OPERABLE, restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

1. With one LPCI subsystem inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHR cross-tie valve (HV-51-182 A or B) open, or power not removed from one closed RHR cross-tie valve operator, close the open valve and/or remove power from the closed valves operator within 72 hours, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With no RHR cross-tie valves (HV-51-182 A, B) closed, or power not removed from both closed RHR cross-tie valve operators, or with one RHR cross-tie valve open and power not removed from the other RHR cross-tie valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With two LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore at least three LPCI subsystems to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With three LPCI subsystems inoperable, provided that both CSS subsystems are OPERABLE, restore at least two LPCI subsystems 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.

6. With all four LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

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

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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

TABLE 3.6.3-1
PRIMARY CONTAINMENT ISOLATION VALVES
NOTATION

NOTES (Continued)

15. Check valve used instead of flow orifice.

16. Penetration is sealed by a flange with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings. Both the TIP Purge Supply (Penetration 35B) and the TIP Drive Tubes (Penetrations 35 C thru G) are welded to their respective flanges. Leakage through these seals is included in the Type C leakage rate total for this penetration. The ball valves (XV-141A thru E) are Type C tested. It is not practicable to leak test the shear valves (XV-140A thru E) because squib firing is required for closure. Shear valves (XV-140A thru E) are normally open.

17. Instrument line isolation provisions consist of an excess flow check valve. Because the instrument line is connected to a closed cooling water system inside containment, no flow orifice is provided. The excess flow check valves are subject to operability testing, but no Type C test is performed nor required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

18. In addition to double "O" ring seals, this penetration is tested by pressurizing volume between doors per Specification 4.6.1.3.

19. The RHR system safety pressure relief valves which are flanged to facilitate removal will be equipped with double O-ring seal assemblies on the flange closest to primary containment. These seals will be leak rate tested by pressurizing between the O-rings, and the results added into the Type C total for this penetration.

20. See Specification 3.3.2, Table 3.3.2-1, for a description of the PCRVICS isolation signal(s) that initiate closure of each automatic isolation valve. In addition, the following non-PCRVICS isolation signals also initiate closure of selected valves:

LFHP With HPCI pumps running, opens on low flow in associated pipe, closes when flow is above setpoint

LFRC With RCIC pump running, opens on low flow in associated pipe, closes when flow is above setpoint

LFCH With CSS pump running, opens on low flow in associated pipe, closes when flow is above setpoint

LFCC Steam supply valve fully closed or RCIC turbine stop valve fully closed

All power operated isolation valves may be opened or closed remote manually.

NOTES (Continued)

.. 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.

34. Auto isolation signals have been removed from HV-087-124 A/B and 125 A/B. Valves to be closed with associated circuit breakers locked open during OPCONs 1, 2, and 3.

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent residual heat removal service water (RHRSW) system subsystems, with each subsystem comprised of:

a. Two OPERABLE RHRSW pumps, and

b. An OPERABLE flow path capable of taking suction from the RHR service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water through one Unit 1 RHR heat exchanger,

shall be OPERABLE:

a. In OPERABLE CONDITIONS 1, 2, and 3, two subsystems.

b. In OPERABLE CONDITIONS 4 and 5, the subsystem(s) associated with systems and components required OPERABLE by Specification 3.4.9.2, 3.9.11.1, and 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHRSW pump in each subsystem inoperable, restore at least one of the inoperable RHRSW pumps to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one RHRSW subsystem otherwise inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE RHRSW pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With both RHRSW subsystems otherwise inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

---

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

PLANT SYSTEMS

EMERGENCY SERVICE WATER SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 1 and common safety-related equipment,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two loops.

b. In OPERATIONAL CONDITIONS 4, 5, and *, one loop.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and *.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*When handling irradiated fuel in the secondary containment.

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators no aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized:

a. A.C. power distribution:

1. Unit 1 Division 1, Consisting of:

a)	4160-VAC Bus:	D11 (10A115)
b)	480-VAC Load Center:	D114 (10B201)
c)	480-VAC Motor Control Centers:	D114-R-C1 (10B219)
		D114-R-C (10B213)
		D114-R-G (10B211)
		D114-R-G1 (10B215)
		D114-D-G (10B515)
d)	120-VAC Distribution Panels:	10Y101
		10Y206

2. Unit 1 Division 2, Consisting of:

a)	4160-VAC Bus:	D12 (10A116)
b)	480-VAC Load Center:	D124 (10B202)
c)	480-VAC Motor Control Centers:	D124-R-C1 (10B220)
		D124-R-C (10B214)
		D124-R-G (10B212)
		D124-R-G1 (10B216)
		D124-D-G (10B516)
d)	120-VAC Distribution Panels:	10Y102
		10Y207

3. Unit 1 Division 3, Consisting of:

a)	4160-VAC Bus:	D13 (10A117)
b)	480-VAC Load Center:	D134 (10B203)
c)	480-VAC Motor Control Centers:	D134-R-H1 (10B221)
		D134-R-H (10B217)
		D134-R-E (10B223)
		D134-C-B (00B131)
		D134-D-G (10B517)
d)	120-VAC Distribution Panels:	10Y103
		10Y163

4. Unit 1 Division 4, Consisting of:

a)	4160-VAC Bus:	D14 (10A118)
b)	480-VAC Load Center:	D144 (10B204)

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of (1) Administrative Procedures and changes thereto, (2) new programs or procedures required by specification 6.8 and requiring a 10 CFR 50.59 safety evaluation, and (3) proposed changes to programs or procedures required by Specification 6.8 and requiring a 10 CFR 50.59 safety evaluation;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. DELETED.

f. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President, Limerick Generating Station, Plant Manager, and to the Nuclear Review Board;

g. Review of all REPORTABLE EVENTS;

h. Review of unit operations to detect potential hazards to nuclear safety;

i. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Vice President, Limerick Generating Station, plant Manager or the Chairman of the Nuclear Review Board;

j. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Nuclear Review Board; and

k. Review of the Emergency Plan and implementing procedures and submittal of the recommended changes to the Nuclear Review Board.

l. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice President, Limerick Generating Station, Plant Manager, and to the Nuclear Review Board.

m. Review of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

n. Review of the Fire Protection Program and implementing procedures and the submittal of recommended changes to the Nuclear Review Board.

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6b. through f. constitutes an unreviewed safety question.

RESPONSIBILITIES (Continued)

c. Provide written notification within 24 hours to the Vice President, Limerick Generating Station and the Nuclear Review Board of disagreement between the PORC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President, Limerick Generating Station, Plant Manager, and the Nuclear Review Board.

6.5.2 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.2.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NRB shall report to and advise the Senior Vice President and Chief Nuclear Officer on those areas of responsibility pertaining to NRB Review and Audits.

COMPOSITION

6.5.2.2 The Chairman, members, and alternates of the NRB shall be appointed in writing by the Chief Nuclear Officer, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of 5 years technical experience, of which a minimum of 3 years shall be in one or more areas given in Specification 6.5.2.1. The NRB shall be composed of no less than eight and no more than 12 members.

The members and alternates of the NRB will be competent in the area of Quality Assurance practice and cognizant of the Quality Assurance requirements of 10 CFR Part 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance organization available to them.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-39

LIMERICK GENERATING STATION

UNITS 1 AND 2

PECO ENERGY COMPANY

DOCKET NOS. 50-352, 50-353

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

sensitive land uses in the site vicinity (e.g., residences, schools, churches, cemeteries, hospitals, parks); and (3) previously conducted noise surveys in the site vicinity.

The selection, calibration and use of equipment, conduct of the surveys, and the analysis and reporting of data shall conform to the provisions of the applicable American National Standards Institute Standards.

The results of the surveys conducted under this program shall be summarized, interpreted and reported in accordance with Section 5.4.1 of this EPP.

The final report of this program shall present a brief assessment by the licensee of the environmental impact of plant and supplemental cooling water system operation on the various offsite acoustic environments, and shall describe the mitigative measures, if any, that have been, or are to be taken to reduce the impact of plant or supplemental cooling water system noise levels on the offsite environments. This report shall also contain a list of noise-related complaints or inquiries received by PECO Energy Company concerning the Limerick Generating Station or its supplemental cooling water system subsequent to issuance of the operating license along with a description of the action taken by PECO Energy Company to resolve these complaints or inquiries.

This program shall terminate upon completion of the collection of the specified sound level data for each phase and submission of an acceptable final report.

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-39

PECO Energy Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
128	This amendment authorizes the licensees to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated October 6, 1997, as supplemented by letter dated February 2, 1998, and evaluated in the safety evaluation dated May 14, 1998.	30 days from May 14, 1998
128	The suppression pool floor and the low pressure ECCS (RHR and Core Spray) suction strainers shall be visually inspected for sludge accumulation and foreign material. The visual inspection allows use of a remote camera in lieu of divers. The interval of these inspections shall be every other refueling outage. The inspection interval may be increased based on findings of two consecutive inspections. Should the licensee choose to increase the inspection interval, data supporting this increase shall be submitted to the NRC for review.	30 days from May 14, 1998

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. 92
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 February 25, 1997, as supplemented by letters dated September 8 and November 18, 1997 and January 8 and July 2, 1998, 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, pages 1, 2, 3 and 5 of Facility Operating License No. NPF-85 are hereby amended by changing Philadelphia Electric Company to PECO Energy Company.

3. Further, 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. 92 , are hereby incorporated in the license. PECO Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robt A. Capra

Robert A. Capra, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1. Pages 1, 2, 3, and 5 of Facility Operating License No. NPF-85
2. Changes to the Technical Specifications
3. Cover page and page 4-3 of the Environmental Protection Plan

Date of Issuance: October 23, 1998

*Pages 1, 2, 3, and 5 of the license are attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Facility Operating License (FOL), the Appendix A Technical Specifications and the Appendix B Environmental Protection Plan (EPP) with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

	Remove	Insert
FOL	1	1
	2	2
	3	3
	5	5
Appendix A	x	x
	xix	xix
	3/4 3-8	3/4 3-8
	3/4 3-103	3/4 3-103
	3/4 5-1	3/4 5-1
	3/4 5-2	3/4 5-2
	3/4 6-14	3/4 6-14
	3/4 6-15	3/4 6-15
	3/4 6-16	3/4 6-16
	3/4 6-42	3/4 6-42
	3/4 6-43	3/4 6-43
	3/4 6-45	3/4 6-45
	3/4 7-1	3/4 7-1
	3/4 7-3	3/4 7-3
	3/4 8-4	3/4 8-4
	6-8	6-8
	6-9	6-9
Appendix B	EPP Cover	EPP Cover
	EPP 4-3	EPP 4-3

PECO ENERGY COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

FACILITY OPERATING LICENSE

License No. NPF-85

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

   A. The application for license filed by PECO Energy Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Limerick Generating Station, Unit 2 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-107 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. The licensee is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Facility Operating License No. NPF-85, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings and the Decision of the Atomic Safety and Licensing Board, LBP-85-25, dated July 22, 1985, the Commission's Order dated July 7, 1989, and the Commission's Memorandum and Order dated August 25, 1989, regarding this facility, Facility Operating License NPF-85 is hereby issued to the PECO Energy Company (the licensee), to read as follows:

A. This license applies to the Limerick Generating Station, Unit 2, a boiling water nuclear reactor and associated equipment, owned by PECO Energy Company. The facility is located on the licensee's site in Montgomery and Chester Counties, Pennsylvania on the banks of the Schuylkill River approximately 1.7 miles southeast of the city limits of Pottstown, Pennsylvania and 21 miles northwest of the city limits of Philadelphia, Pennsylvania, and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report-Operating License Stage, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses PECO Energy Company:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Montgomery and Chester Counties, Pennsylvania, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PECO Energy Company is authorized to operate the facility at reactor core power levels of 3458 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No.   , are hereby incorporated into this license. PECO Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection (Section 9.5, SSER-2,-4)*

PECO Energy Company shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report dated August 1983 through Supplement 9, dated August 1989, and Safety Evaluation dated November 20, 1995, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

the local leak rate testing of the Traversing Incore Probe Shear Valves (Section 6.2.6.1 of the SER and SSER-3), and (d) an exemption from the schedule requirements of 10 CFR 50.33(k)(1) related to availability of funds for decommissioning the facility (Section 22.1, SSER 8). The special circumstances regarding exemptions (a), (b) and (c) are identified in Sections 6.2.6.1 of the SER and SSER 3. An exemption from the criticality monitoring requirements of 10 CFR 70.24 was previously granted with NRC materials license No. SNM-1977 issued November 22, 1988. The licensee is hereby exempted from the requirements of 10 CFR 70.24 insofar as this requirement applies to the handling and storage of fuel assemblies held under this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items a, b, c, and d above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

F. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Amendment No. ~~44~~, ~~64~~, 92

BASES

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	Q	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) DELETED

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.

(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	Q	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	Q	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	W	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) DELETED
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

OFFGAS GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The offgas monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.2.5 and 3.11.2.6 respectively, are not exceeded.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

a. With an offgas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.12-1.

b. With less than the minimum number of offgas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Annual Radioactive Effluent Release Report.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each offgas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.12-1.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE CSS pumps, and

   2. An OPERABLE flow patch capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of four subsystems with each subsystem comprised of:

   1. One OPERABLE LPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure coolant injection (HPCI) system consisting of:

   1. One OPERABLE HPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with at least five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* ** #, and 3* ** ##.

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

**The ADS is not required to be OPERABLE when the reactor steam dome pressure is less that or equal to 100 psig.

#See Special Test Exception 3.10.6.

##Two LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR Shutdown cooling permissive setpoint.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least two LPCI subsystems are OPERABLE, restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

1. With one LPCI subsystem inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHR cross-tie valve (HV-51-282 A or B) open, or power not removed from one closed RHR cross-tie valve operator, close the open valve and/or remove power from the closed valves operator within 72 hours, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With no RHR cross-tie valves (HV-51-282 A, B) closed, or power not removed from both closed RHR cross-tie valve operators, or with one RHR cross-tie valve open and power not removed from the other RHR cross-tie valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With two LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore at least three LPCI subsystems to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With three LPCI subsystems inoperable, provided that both CSS subsystems are OPERABLE, restore at least two LPCI subsystems 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.

6. With all four LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

---

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

SURVEILLANCE REQUIREMENTS (Continued)

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 $A/\sqrt{k}$ 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 $A/\sqrt{k}$ 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.

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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

NOTES (Continued)

15. Check valve used instead of flow orifice.

16. Penetration is sealed by a flange with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings. Both the TIP Purge Supply (Penetration 35B) and the TIP Drive Tubes (Penetrations 35 C thru G) are welded to their respective flanges. Leakage through these seals is included in the Type C leakage rate total for this penetration. The ball valves (XV-241A thru E) are Type C tested. It is not practicable to leak test the shear valves (XV-240A thru E) because squib firing is required for closure. Shear valves (XV-240A thru E) are normally open.

17. Instrument line isolation provisions consist of an excess flow check valve. Because the instrument line is connected to a closed cooling water system inside containment, no flow orifice is provided. The excess flow check valves are subject to operability testing, but no Type C test is performed nor required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).

18. In addition to double "O" ring seals, this penetration is tested by pressurizing volume between doors per Specification 4.6.1.3.

19. The RHR system safety pressure relief valves are flanged to facilitate removal and are equipped with double O-ring seal assemblies on the flange closest to primary containment. These seals will be leak rate tested by pressurizing between the O-rings, and the results added into the Type C total for this penetration.

20. See Specification 3.3.2, Table 3.3.2-1, for a description of the PCRVICS isolation signal(s) that initiate closure of each automatic isolation valve. In addition, the following non-PCRVICS isolation signals also initiate closure of selected valves:

LFHP With HPCI pumps running, opens on low flow in associated pipe, closes when flow is above setpoint

LFRC With RCIC pump running, opens on low flow in associated pipe, closes when flow is above setpoint

LFCH With CSS pump running, opens on low flow in associated pipe, closes when flow is above setpoint

LFCC Steam supply valve fully closed or RCIC turbine stop valve fully closed

All power operated isolation valves may be opened or closed remote manually.

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.

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

35. Auto isolation signals have been removed from HV-087-224 A/B and 225 A/B. Valves to be closed with associated circuit breakers locked open during OPCONs 1, 2, and 3.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

3. At least once per 24 months by:

a) Verifying each valve's opening setpoint, from the closed position, to be 0.5 psid ± 5%, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

c) Verifying that each outboard valve's position indicator is capable of detecting disk displacement ≥0.050", and each inboard valve's position indicator is capable of detecting disk displacement ≥0.120".

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent residual heat removal service water (RHRSW) system subsystems, with each subsystem comprised of:

a. Two OPERABLE RHRSW pumps, and

b. An OPERABLE flow path capable of taking suction from the RHR service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water through one Unit 2 RHR heat exchanger,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two subsystems.

b. In OPERATIONAL CONDITIONS 4 and 5, the subsystem(s) associated with systems and components required OPERABLE by Specification 3.4.9.2, 3.9.11.1, and 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHRSW pump in each subsystem inoperable, restore at least one of the inoperable RHRSW pumps to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one RHRSW subsystem otherwise inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE RHRSW pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With both RHRSW subsystems otherwise inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

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

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 2 and common safety-related equipment,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two loops.

b. In OPERATIONAL CONDITIONS 4, 5, and *, one loop.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and *.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*When handling irradiated fuel in the secondary containment.

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators not aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

SURVEILLANCE REQUIREMENTS (Continued)

b. By removing accumulated water:

1) From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and

2) From the storage tank at least once per 31 days.

c. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A, except that the filters specified in ASTM D2276-78, Sections 5.1.6 and 5.1.7, may have a nominal pore size of up to three (3) microns.

e. At the following frequency by:

1) Every 18 months subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2) Every 24 months verify each diesel generator's capability to reject a load of greater than or equal to that of its single largest post-accident load while maintaining voltage at 4285 ± 420 volts and frequency at 60 ± 1.2 hz and after steady state conditions are reached, voltage is maintained at 4280 ± 120 volts.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of (1) Administrative Procedures and changes thereto, (2) new programs or procedures required by Specification 6.8 and requiring a 10 CFR 50.59 safety evaluation, and (3) proposed changes to programs or procedures required by Specification 6.8 and requiring a 10 CFR 50.59 safety evaluation;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. DELETED.

f. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President, Limerick Generating Station, Plant Manager, and to the Nuclear Review Board;

g. Review of all REPORTABLE EVENTS;

h. Review of unit operations to detect potential hazards to nuclear safety;

i. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Vice President, Limerick Generating Station, plant Manager or the Chairman of the Nuclear Review Board;

j. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Nuclear Review Board; and

k. Review of the Emergency Plan and implementing procedures and submittal of the recommended changes to the Nuclear Review Board.

l. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice President, Limerick Generating Station, Plant Manager, and to the Nuclear Review Board.

m. Review of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

n. Review of the Fire Protection Program and implementing procedures and the submittal of recommended changes to the Nuclear Review Board.

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6b. through f. constitutes an unreviewed safety question.

RESPONSIBILITIES (Continued)

c. Provide written notification within 24 hours to the Vice President, Limerick Generating Station and the Nuclear Review Board of disagreement between the PORC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President, Limerick Generating Station, Plant Manager, and the Nuclear Review Board.

6.5.2 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.2.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NRB shall report to and advise the Senior Vice President and Chief Nuclear Officer on those areas of responsibility pertaining to NRB Review and Audits.

COMPOSITION

6.5.2.2 The Chairman, members, and alternates of the NRB shall be appointed in writing by the Chief Nuclear Officer, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of 5 years technical experience, of which a minimum of 3 years shall be in one or more areas given in Specification 6.5.2.1. The NRB shall be composed of no less than eight and no more than 12 members.

The members and alternates of the NRB will be competent in the area of Quality Assurance practice and cognizant of the Quality Assurance requirements of 10 CFR Part 50, Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance organization available to them.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-85

LIMERICK GENERATING STATION

UNITS 1 AND 2

PECO ENERGY COMPANY

DOCKET NOS. 50-352 and 50-353

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

August 25, 1989

The selection, calibration and use of equipment, conduct of the surveys, and the analysis and reporting of data shall conform to the provisions of the applicable American National Standards Institute Standards.

The results of the surveys conducted under this program shall be summarized, interpreted and reported in accordance with Section 5.4.1 of this EPP.

The final report of this program shall present a brief assessment by the licensee of the environmental impact and supplemental cooling water system operation on the various offsite acoustic environments, and shall describe the mitigative measures, if any, that have been, or are to be taken to reduce the impact of plant or supplemental cooling water system noise levels on the offsite environments. This report shall also contain a list of noise-related complaints or inquiries received by PECO Energy Company concerning the Limerick Generating Station or its supplemental cooling water system subsequent to issuance of the operating license along with a description of the action taken by PECO Energy Company to resolve these complaints or inquiries.

This program shall terminate upon completion of the collection of the specified sound level data for each phase and submission of an acceptable final report.

4.2.4.2 Point Pleasant Pumphouse

An ASLB ruling (LBP-83-11; March 8, 1983) requires that the licensee conduct a one-time field study after the transformers are placed in operation at Point Pleasant. The noise from operation of the transformers shall be reduced to a level so that the transformer core tones will be inaudible (i.e., not above the masking level, as defined below) at the site boundary.

The licensee shall determine, based on onsite measurements, the delta L(ex) (i.e., the noise level in excess of the masking level) for each tone. The masking level is defined as "N" dB above the ambient spectrum level, where "N" is defined as follows: