

March 4, 1991

Mr. George W. Davis
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Davis:

SUBJECT: ISSUANCE OF AMENDMENT NO. 135 TO FACILITY OPERATING LICENSE NO.
DPR-35 - PILGRIM NUCLEAR POWER STATION (TAC NO. 76482)

The Commission has issued the enclosed Amendment No. 135 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated March 15, 1990.

This amendment changes the surveillance requirements for redundant core and containment cooling systems and the allowed out-of-service period for the containment cooling system and low pressure coolant injection pumps.

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

Sincerely,

15/

Ronald Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:

- 1. Amendment No. 135 to License No. DPR-35
- 2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD-3
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2/12/91

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2/12/91

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3/14/91

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

March 4, 1991

Docket No. 50-293

Mr. George W. Davis
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

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

Sincerely,

A handwritten signature in black ink, appearing to read "Ronald Eaton".

Ronald Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 135 to
License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

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Pilgrim Nuclear Power Station

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AMENDMENT NO. 135 TO DPR-35 PILGRIM NUCLEAR POWER STATION DATED March 4, 1991

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Docket File 50-293 ←

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 135
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:
 - A. The application for amendment filed by the Boston Edison Company (the licensee) dated March 15, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 135, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Susan F. Shankman, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 4, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 135

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

<u>Remove</u>	<u>Insert</u>
i	i
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105	105
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BASES:

3.4 & 4.4 STANDBY LIQUID CONTROL SYSTEM

- A. The requirements for SLC capability to shutdown the reactor are identified via the station Nuclear Safety Operational Analysis (Appendix G to the FSAR, Special Event 45). If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the standby liquid control system is required. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control system is designed to inject a quantity of boron that produces a minimum concentration equivalent to 675 ppm of natural boron in the reactor core. The 675 ppm equivalent concentration in the reactor core is required to bring the reactor from full power to at least a three percent Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning etc. The system will inject this boron solution in less than 125 minutes. The maximum time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The Standby Liquid Control system is also required to meet 10CFR50.62 (Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants). The Standby Liquid Control system must have the equivalent control capacity (injection rate) of 86 gpm at 13 percent by wt. natural sodium pentaborate for a 251" diameter reactor pressure vessel in order to satisfy 10CFR50.62 requirements. This equivalency requirement is fulfilled by a combination of concentration, B^{10} enrichment and flow rate of sodium pentaborate solution. A minimum 8.42% concentration and 54.5% enrichment of B^{10} isotope at a 39 GPM pump flow rate satisfies the ATWS Rule (10CFR50.62) equivalency requirement.

Because the concentration/volume curve has been revised to reflect the increased B^{10} isotopic enrichment, an additional requirement has been added to evaluate the solution's capability to meet the original design shutdown criteria whenever the B^{10} enrichment requirement is not met.

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. The only practical time to fully test the liquid control system is during a refueling outage. Various components of the system are individually tested periodically, thus making more frequent testing of the entire system unnecessary.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and suppression pool cooling systems.

Objective

To assure the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Systems

1. Both core spray systems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 below.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the Surveillance Requirements of the core and suppression pool cooling systems which are required when the corresponding Limiting Condition for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Systems

1. Core Spray System Testing.

<u>Item</u>	<u>Frequency</u>
a. Simulated Automatic Actuation test.	Once/Operating Cycle
b. Pump Operability	Once/month
c. Motor Operated Valve Operability	Once/month
d. Pump flow rate Each pump shall deliver at least 3300 gpm against a system head corresponding to a reactor vessel pressure of 10 ⁴ psig.	Once/3 months
e. Core Spray Header Δ p Instrumentation	

LIMITING CONDITION FOR OPERATIONSURVEILLANCE REQUIREMENT3.5.A Core Spray and LPCI Systems
(cont'd)4.5.A Core Spray and LPCI Systems
(cont'd)

2. From and after the date that one of the core spray systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray system and active components of the LPCI system and the diesel generators are operable.
3. The LPCI system shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4 and 3.5.F.5.
4. From and after the date that the LPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days the containment cooling system (including 2 LPCI pumps) and active components of both core spray systems, and the diesel generators required for operation of such components if no external source of power were available shall be operable.
5. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

- | | |
|-----------|---------------|
| Check | Once/day |
| Calibrate | Once/3 months |
| Test Step | Once/3 months |
2. This section intentionally left blank
 3. LPCI system Testing shall be as follows:
 - a. Simulated Automatic Actuation Test
Once/Operating Cycle
 - b. Pump Operability
Once/month
 - c. Motor Operated valve operability
Once/Month
 - d. Pump Flow
Once/3 months
- Each LPCI pump shall pump 4800 gpm at a head across the pump of at least 380 ft.

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LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5.B Containment Cooling System

1. Except as specified in 3.5.B.2 and 3.5.F.3 below, both containment cooling system loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Condition.
2. From and after the date that one containment cooling system loop is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 72 hours unless such system loop is sooner made operable, provided that the other containment cooling system loop, including its associated diesel generator, is operable.
3. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

4.5.B Containment Cooling System

1. Containment Cooling system Testing shall be as follows:

<u>Item</u>	<u>Frequency</u>
a. Pump & Valve Operability	Once/3 months
b. Pump Capacity Test Each RBCCW pump shall deliver 1700 gpm at 70 ft. TDH. Each SSWS pump shall deliver 2700 gpm at 55 ft. TDH.	After pump maintenance and every 3 months
c. Air test on drywell and torus headers and nozzles	Once/5 years

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

C. HPCI System

1. The HPCI system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F; except as specified in 3.5.C.2 below.
2. From and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, providing that during such seven days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable.
3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 150 psig within 24 hours.

C. HPCI System

1. HPCI system testing shall be performed as follows:
 - a. Simulated Automatic Actuation Test Once/operating cycle
 - b. Pump Operability Once/month
 - c. Motor Operated Valve Operability Once/month
 - d. Flow Rate at 1000 psig Once/3 months
 - e. Flow Rate at 150 psig Once/operating cycle

The HPCI pump shall deliver at least 4250 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5.D Reactor Core Isolation Cooling (RCIC) System

1. The RCIC system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F; except as specified in 3.5.D.2 below.
2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.
3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 150 psig within 24 hours.

4.5.D Reactor Core Isolation Cooling (RCIC) System

1. RCIC system testing shall be performed as follows:
 - a. Simulated Automatic Actuation Test Once/operating cycle
 - b. Pump Operability Once/month
 - c. Motor Operated Valve Operability Once/month
 - d. Flow Rate at 1000 psig Once/3 months
 - e. Flow Rate at 150 psig Once/operating cycle

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5.E Automatic Depressurization System (ADS)

1. The Automatic Depressurization System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.
2. From and after the date that one valve in the Automatic Depressurization System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI system is operable.
3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 104 psig within 24 hours.

4.5.E Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:
 - a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. The ADS manual inhibit switch will be included in this test.
 - b. With the reactor at pressure, each relief valve shall be manually opened until a corresponding change in reactor pressure or main turbine bypass valve positions indicate that steam is flowing from the valve.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

4.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling systems and the remaining diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.
2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.
3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown condition, both core spray systems, the LPCI and containment cooling systems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.
4. During a refueling outage, for a period of 30 days, refueling operation may continue provided that one core spray system or the LPCI system is operable or Specification 3.5.F.5 is met.
5. When irradiated fuel is in the reactor vessel and the reactor is in the Refueling Condition with the torus drained, a single control rod drive mechanism may be removed, if both of the following conditions are satisfied:

1. When it is determined that one diesel generator is inoperable, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter until the inoperable diesel is repaired.

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

- a) No work on the reactor vessel, in addition to CRD removal, will be performed which has the potential for exceeding the maximum leak rate from a single control blade seal if it became unseated.
- b) i) the core spray systems are operable and aligned with a suction path from the condensate storage tanks. ii) the condensate storage tanks shall contain at least 200,000 gallons of usable water and the refueling cavity and dryer/separator pool shall be flooded to a least elevation 114'-0"

3.5.G

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3.5.H Maintenance of Filled Discharge Pipe

Whenever core spray systems, LPCI system, HPCI or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5.H Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray systems, LPCI system, HPCI and RCIC are filled:

1. Every month prior to the testing of the LPCI system and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow observed.
2. Following any period where the LPCI system or core spray systems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

4.5.H Maintenance of Filled Discharge Pipe (Cont'd)

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. The pressure switches which monitor the discharge lines to ensure that they are full shall be functionally tested every month and calibrated every three months.

BASES:

3.5.A Core Spray and LPCI System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss of coolant analysis performed by General Electric in accordance with Section 50.46 and Appendix K of 10CFR50, the Pilgrim I Emergency Core Cooling Systems are adequate to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit calculated fuel clad temperature to less than 2200°F, to limit calculated local metal water reaction to less than or equal to 17%, and to limit calculated core wide metal water reaction to less than or equal to 1%.

General Electric Company Proprietary Report EAS-65-0989, "Safety Evaluation for Interim Operation of Pilgrim Nuclear Power Station with Reduced Core Spray System Flow Rate" (September 1989) calculates a peak fuel clad temperature of less than 2200°F with a Core Spray pump flow of 3240 gallons per minute (gpm). A flow rate of 3300 gpm ensures adequate flow for events involving degraded voltage.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Pilgrim, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis takes credit for core spray flow into the core at vessel pressure below 205 psig. However, the analysis is conservative in that no credit is taken for spray cooling heat transfer in the hottest fuel bundle until the pressure at rated flow for the core spray (104 psig vessel pressure) is reached.

The LPCI system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI system and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling systems.

The combination of the core spray systems and the LPCI system assures that adequate core cooling is achieved assuming any coincident single failure of an active safety-related component. Core Standby Cooling System (CSCS) performance evaluations consider only the most severe single failure for each break size range. These single failures include the LPCI injection valve, one diesel generator, the HPCI system or one ADS valve. With these single failures, the combinations of analyzed low pressure CSCS capacity include two core spray pumps, one core spray pump and two LPCI pumps, or two core spray and four LPCI pumps. Each core spray system consists of one pump and associated piping and valves with all active components required to be operable. The LPCI system consists of four LPCI pumps and associated piping and valves with all active components required to be operable.

BASES:

3.5.A Core Spray and LPCI System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss of coolant analysis performed by General Electric in accordance with Section 50.46 and Appendix K of 10CFR50, the Pilgrim I Emergency Core Cooling Systems are adequate to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit calculated fuel clad temperature to less than 2200°F, to limit calculated local metal water reaction to less than or equal to 17%, and to limit calculated core wide metal water reaction to less than or equal to 1%.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Pilgrim, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis takes credit for core spray flow into the core at vessel pressure below 205 psig. However, the analysis is conservative in that no credit is taken for spray cooling heat transfer in the hottest fuel bundle until the pressure at rated flow for the core spray (104 psig vessel pressure) is reached.

The LPCI system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI system and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling systems.

The combination of the core spray systems and the LPCI system assures that adequate core cooling is achieved assuming any coincident single failure of an active safety-related component. Core Standby Cooling System (CSCS) performance evaluations consider only the most severe single failure for each break size range. These single failures include the LPCI injection valve, one diesel generator, the HPCI system or one ADS valve. With these single failures, the combinations of analyzed low pressure CSCS capacity include two core spray pumps, one core spray pump and two LPCI pumps, or two core spray and four LPCI pumps. Each core spray system consists of one pump and associated piping and valves with all active components required to be operable. The LPCI system consists of four LPCI pumps and associated piping and valves with all active components required to be operable.

BASES:

3.5.A Core Spray and LPCI Systems (Cont'd)

Should one core spray system become inoperable, the remaining core spray and the LPCI system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems; i.e., the core spray and LPCI, a seven-day repair period was obtained.

If the LPCI system is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. Based on judgments of the reliability of the remaining core spray systems, a 7-day repair period was set.

The LPCI system is not considered inoperable when the RHR System is operating in the shutdown cooling mode.

BASES:

3.5.B Containment Cooling System

The containment cooling system for Pilgrim I consists of two independent loops each of which to be an operable loop requires one LPCI pump, two RBCCW pumps, and two SSW pumps to be operable. There are installed spares for margin above the design conditions. Each system has the capability to perform its function; i.e., removing 64×10^6 Btu/hr (Ref. Amendment 18), even with some system degradation. If one loop is out-of-service, reactor operation is permitted for 72 hours.

With components or systems out-of-service, overall core and containment cooling reliability is maintained by the operability of the remaining cooling equipment.

Since some of the SSW and RBCCW pumps are required for normal operation, capacity testing of individual pumps by direct flow measurement is impractical. The pump capacity test is a comparison of measured pump performance parameters to shop performance tests combined with a comparison to the performance of the previously tested pump. These pumps are rotated during operation and performance testing will be integrated with this or performed during refueling when pumps can be flow tested individually. Tests during normal operation will be performed by measuring the shutoff head. Then the pump under test will be placed in service and one of the previously operating pumps secured. Total flow indication for the system will be compared for the two cases. Where this is not feasible due to changing system conditions, the pump discharge pressure will be measured and its power requirement will be used to establish flow at that pressure.

BASES:

3.5.C HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (Appendix G) and a detailed functional analysis of the HPCI System (Section 6).

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 4250 gpm at reactor pressures between 1100 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reached equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The analysis in the FSAR, Appendix G, shows that the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC is required as an alternate source of makeup to the HPCI only in the case of loss of all offsite A-C power. Considering the HPCI and the ADS plus RCIC as redundant paths, and considering judgments of the reliability of the ADS and RCIC systems, a 7-day allowable repair time is specified.

The requirement that HPCI be operable when reactor coolant temperature is greater than 365°F is included in Specification 3.5.C.1 to clarify that HPCI need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

BASES:

3.5.D RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The nuclear safety analysis, FSAR Appendix G, shows that RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgments on the reliability of the HPCI system, an allowable repair time of seven days is specified.

The requirement that RCIC be operable when reactor coolant temperature is greater than 365°F is included in Specification 3.5.D.1 to clarify that RCIC need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

BASES:

3.5.E Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Operational Analysis (Appendix G) and a detailed functional analysis of the ADS (Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray systems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of 800,000 lb/hr at a reactor pressure of 1125 psig.

The allowable out of service time for one ADS valve is determined as seven days because of the redundancy and because of HPCIS operability; therefore, redundant protection for the core with a small break in the nuclear system is still available.

The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

BASES:

3.5.H Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI system, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. An analysis has been done which shows that if a water hammer were to occur at the time at which the system were required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

BASES:

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

BASES:

3.6.F and 4.6.F

Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the Pilgrim Nuclear Power Station FSAR. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. At or below 80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

The flow mismatch restriction also derives from the "Core Flow Coastdown" concern. This concern postulates that if the recirculation loop with the higher flow is broken, the "effective core flow" is determined by the loop with the lower flow. Compared to a matched flow condition, this would start pump coastdown from a lower flow/speed with the reactor power effectively above the rated rod line. Therefore, boiling transition may occur earlier during a postulated LOCA event, which could result in higher calculated peak cladding temperatures (PCTs). Therefore, the purpose of the "Core Flow Coastdown" flow mismatch restriction is to maintain Pilgrim within its analyzed conditions.

Specification 3.6.F allows 30 minutes to correct a mismatch in recirculation pump speeds in order to take manual control of the recirculation pump MG set scoop tube positioner in the event that its control system should fail.

4.9

The diesel fuel oil quality must be checked to ensure proper operation of the diesel generators. Water content should be minimized because water in the fuel could contribute to excessive damage to the diesel engine.

The Electrical Protection Assemblies (EPAs) on the RPS inservice power supplies (either two motor generator sets or one motor generator and the alternate supply), consist of protective relays that trip their incorporated circuit breakers on overvoltage, undervoltage or underfrequency conditions. There are 2 EPAs in series per power source. It is necessary to periodically test the relays to ensure the sensor is operating correctly and to ensure the trip unit is operable. Based on experience at conventional and nuclear power plants, a six month frequency for the channel functional test is established. This frequency is consistent with the Standard Technical Specifications.

The EPAs of the power sources to the RPS shall be determined to be operable by performance of a channel calibration of the relays once per operating cycle. During calibration, a transfer to the alternate power source is required; however, prior to switching to alternate feed, de-energization of the applicable MG set power source must be accomplished. This results in a half-scram on the channel being calibrated until the alternate power source is connected and the half scram is cleared. Based on operating experience, drift of the EPA protective relays is not significant. Therefore, to avoid possible spurious scrams, a calibration frequency of once per cycle is established.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 135 TO FACILITY OPERATING LICENSE NO. DPR-35
BOSTON EDISON COMPANY
PILGRIM NUCLEAR POWER STATION
DOCKET NO. 50-293

INTRODUCTION

By letter dated March 15, 1990, the Boston Edison Company (the licensee) requested an amendment to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. The proposed amendment would change the Technical Specifications (TS) to 1) remove the need for surveillance testing on the operational train of the Core and Containment Cooling Systems when the redundant train is inoperable; 2) to reduce the allowed out of service period for the Containment Cooling System and the Diesel Generators from 7 days to 72 hours; 3) to eliminate the 30 day out of service time for the inoperability of the one Low Pressure Cooling Injection (LPCI) pump; and 4) to implement editorial changes by replacing the word "LPCI subsystem" with "LPCI system" throughout this section of the TS and removing an expired footnote from section 3.5.B.

The licensee changed the Pilgrim Nuclear Power Station's Bases for TS Section 3.4 & 4.4.A Standby Liquid Control System (SLCS) and 3.6.F and 4.6.F Jet Pump Flow Mismatch in a letter dated September 27, 1990.

EVALUATION

For the Containment and Core Cooling Systems, the current TS require the immediate and daily demonstration of the operability of the redundant train when the alternate train is placed or found in an inoperable condition. The purpose of this testing is to demonstrate that both trains have not been incapacitated due to a common mode failure. The immediate and daily testing requirements result in excessive surveillance testing and unnecessary equipment wear. This increase of equipment wear offsets any gain in assurance of equipment availability due to the increased likelihood of the equipment being left in the testing mode (inoperable) through operator error.

Daily surveillance testing, in addition to the regularly scheduled surveillance test, is excessive since the regularly scheduled surveillance test adequately demonstrates operability without significantly diminishing the equipment reliability due to unnecessary wear on components and increased potential of operator errors. Since regularly scheduled surveillance tests are planned to provide assurance that equipment will be available during the interim periods between regularly scheduled surveillances, daily testing is counter-productive.

These proposed TS changes bring this section of the TS into conformance with the Standard Technical Specifications (STS) by removing the need to place redundant systems in daily testing arrangements during periods without system redundancy, by reducing potential system failures due to improper system alignment after excessive testing and by reducing wear on components caused by excessive surveillance testing. Since this portion of the TS change results in improved overall system availability, the NRC staff concludes that the removal of the daily testing requirement during time periods without system redundancy is acceptable.

Reducing the allowed Out of Service (OOS) period from seven days to 72 hours for the Containment Cooling System (CCS) and the Diesel Generators (DGs) and eliminating the 30 day OOS period for a single LPCI pump increases the availability of safety equipment during power operation. Furthermore, these changes to the TS result in making the Pilgrim TS more consistent with the STS.

The reference to I. M. Jacob's "APED-5736: Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" (General Electric Company, April, 1969) is also deleted from the Bases. References to APED-5736 is being deleted from the Bases since it no longer forms the basis for the OOS times and the redundant testing requirements that are being changed by this proposed TS amendment.

Changing the term LPCI subsystem to LPCI system in the text and basis is an editorial change which does not impact plant safety. Likewise, incorporating this editorial change for the Containment Cooling System, Core Spray System, HPCI, and RCIC also will not influence the safety of the plant.

An administrative change is made to Section 4.5.A.1.d to include a surveillance frequency of "once/3 months" for the core spray pump flow rate test. This proposed administrative change restores the frequency which was inadvertently deleted by Amendment #42 to Pilgrim's TS.

Removing the footnote which granted conditional relief to the requirements of section 3.5.B is acceptable because the footnote has expired and no longer has any impact on TS 3.5.B.

During the review of the TS amendment, the staff observed a deletion from the TS that was not justified in the licensee's submittal. This deletion involved the removal of the requirement for immediate and daily testing of the operational diesel generator when the redundant diesel generator is inoperable. This part of the proposed TS change is not acceptable, and the licensee should maintain the requirement for the immediate and daily testing of the operational diesel generator in the TS.

The Bases change for the SLCS has been revised to correct the minimum boron concentration calculated to occur in the reactor vessel upon SLCS injection. The correct boron concentration in the reactor at the cold shutdown consideration is equivalent to 675 ppm of natural boron. This concentration results

in a shutdown margin of 4.01% Wk for the current cycle, which exceeds the minimum required shutdown margin of 3% Wk. This correction only affects Technical Specification Bases Pages 100 and does not reduce the margin of safety defined by the 3% Wk minimum required shutdown margin.

The Technical Specification Bases for jet pump flow mismatch have also been revised on Page 148 to add further justification for the restriction based on core flow coastdown concerns. These bases changes do not alter Technical Specification 3.6.F because additional justification is provided for the existing specification.

The staff has reviewed the changes to the Bases for Sections 3.4 & 4.4.A and 3.6.F and 4.6.F and offers no objection.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets 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 this amendment.

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 18408) on May 2, 1990 and consulted with the Commonwealth of Massachusetts.

Mr. Joseph Kriesberg, Director, Massachusetts Citizens for Safe Energy, Boston, Massachusetts responded to this notice in the form of three questions on the Core and Containment Cooling System surveillance. NRC responded to Mr. Kriesberg's inquiry by letter, dated August 9, 1990, in question and answer format. The Commonwealth of Massachusetts was consulted and did not have any comments.

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

Principal Contributor: Peter Hearn

Dated: March 4, 1991