September 28, 1993

Docket No. 50-293

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Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Boulette:

SUBJECT: ISSUANCE OF AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M84862)

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated October 30, 1992.

The amendment revises Technical Specifications (TSs) to specify limiting conditions of operation and surveillance requirements for inservice code testing. The change also incorporates the term "Refueling Interval" in the definitions to specify the interval between designated ASME Code Section XI surveillances and revises the definition of surveillance interval to allow the 25% tolerance to be applied to the refueling interval of 24 months. In addition, by letters dated February 11, 1993 and March 29, 1993, changes were made to the Bases sections regarding core spray and LPCI system, and drywell temperature.

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

Sincerely,
Original signed by:
Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFFICE	PDI-3:LA	PDI-3:PM	*OGC	PDI-3:D	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 28, 1993

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Senior Vice President - Nuclear
Boston Edison Company
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Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to
   License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.149
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 October 30, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 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. 149, 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 6 months.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, 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: September 28, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 149

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.

Remove	Insert
4	4
5a	5a
73b	73b
85	85
95	95
96	96
100	100
101	101
103	103
104	104
106	106
107	107
108	108
113	113
114	114
115	115
122	122
126	126
155a	155a
156	156
157	157
	205g
	205h
	205i

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Refueling Frequencies

1. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

2. Refueling Interval - Refueling interval applies only to ASME Code, Section XI IWP and IWV surveillance tests. For the purpose of designating frequency of these code tests, a refueling interval shall mean at least once every 24 months.

Q. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

R. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

S. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - the value of critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Total Peaking Factor - The ratio of the fuel rod surface heat flux to the heat flux of an average rod in an identical geometry fuel assembly operating at the core average bundle power.

1.0 DEFINITIONS (Continued)

U. Surveillance Frequency - Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

The Surveillance Frequency establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Definition "U" is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

V. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable. The operating cycle interval is 18 months and the 25% tolerance given in Definition "U" is applicable. The refueling interval is 24 months and the 25% tolerance specified in definition "U" is applicable.

W. Fire Suppression Water System - A fire suppression water system shall consist of: a water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include hydrant post indicator valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

X. Staggered Test Basis - A staggered test basis shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

Y. Source Check - A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

3.2 BASES (Cont'd)

Drywell Temperature

The drywell temperature limitations of Specification 3.2.H.1 ensure that safety related equipment will not be subjected to excess temperature. Exposure to excessive temperatures may degrade equipment and can cause loss of its operability.

The temperature elements for monitoring drywell temperature specified in Table 3.2.H were chosen on the basis of their reliability, location, and their redundancy (dual - element RTD's). These temperature elements are the primary elements used for the PCILRT.

The "nominal instrument elevations" provided in Tables 3.2.H and 4.2.H assist personnel in locating the instruments for surveillance and maintenance purposes and define the approximate containment region to be monitored. The "nominal instrument elevations" are not intended to provide a precise instrument location.

The temperature limits specified in 3.2.H.1 are based on the BECo report entitled Drywell Temperature Report, dated January 28, 1982. The limits derived from this report take into consideration the long-term effects of ambient temperature on equipment design limits and material degradation of components required for accident mitigation or plant shutdown. The evaluation process addressed the actual assessment of potential damage and the determination of equipment status from the standpoint of both qualification integrity (for safety-related equipment) and reliability to perform its intended function.

If the drywell temperature exceeds the limits specified in 3.2.H.1 an engineering evaluation must be initiated in order to determine whether any safety related component has been adversely affected.

The limiting drywell temperature value of 215°F (Section 3.2.H.2) was selected as to guarantee that ECCS trips occur on/or before present Technical Specification values.

The time interval of 30 minutes between successive drywell temperature instrument readings (Section 3.2.H.1) was selected so as to guarantee that ECCS trips occur on/before present Technical Specification values in the event of a drywell temperature excursion in excess of 215°F.

The instrument check interval of once per shift provides adequate assurance of equipment operability based upon engineering judgement.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% ΔK. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken if such actions are appropriate.

F. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Specifications 3.3.A through D above do not apply when there is no fuel in the reactor vessel.

G. Scram Discharge Volume

1. The scram discharge volume drain & vent valves shall be operable whenever more than one operable control rod is withdrawn.

2. If any of the scram discharge volume drain or vent valves are made or found inoperable an orderly shutdown shall be initiated and the reactor shall be in Cold Shutdown within 24 hours.

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

G. Scram Discharge Volume

1. Scram discharge volume drain and vent valves;

   a. Verified open at least once per month.

   b. Test as specified in 3.13. These valves may be closed intermittently for testing under administrative control.

2. During each refueling interval verify the scram discharge volume drain and vent valves;

   a) Close within 30 seconds after receipt of a reactor scram signal and

   b) Open when the scram is reset.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shutdown the reactor and maintain the shutdown condition without the use of control rods.

Specification:

A. Normal System Availability

1. During periods when fuel is in the reactor and prior to startup from a cold condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Shutdown Condition, all operable control rods are fully inserted and Specification 3.3.A is met.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. When tested as specified in 3.13 verify that each pump delivers at least 39 GPM against a system head of 1275 psig.

2. As required below:

   a. Once each refueling interval while testing as specified in 3.13 verify the system relief valve set point of 1425 psig ± 43 psig.

3.4 STANDBY LIQUID CONTROL SYSTEM

4.4 STANDBY LIQUID CONTROL SYSTEM

b. At least once during each refueling interval, while testing as specified in 3.13, manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump capacity. The replacement charges to be installed will be selected from the same manufactured batch as the tested charge.

c. When testing to satisfy requirement 4.4.A.2.b, both systems, including both explosive valves, shall be tested in the course of two refueling intervals.

B. Operation with Inoperable Components:

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

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.

Testing the pumps and valves in accordance with ASME B&PV Code Section XI (Articles IWP and IWV, except where specific relief is granted) adequately assesses component operational readiness. 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.

3.4 & 4.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd)

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten the shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long term average availability of the system is not reduced is obtained for a one out of two system by an allowable equipment out of service time of one third of the normal surveillance frequency. This method determines an equipment out of service time of ten days. Additional conservatism is introduced by reducing the allowable out of service time to seven days, and by increased testing of the operable redundant component.

C. The quantity of B^{10} stored in the Standby Liquid Control System Storage Tank is sufficient to bring the concentration of B^{10} in the reactor to the point where the reactor will be shutdown and to provide a minimum 25 percent margin beyond the amount needed to shutdown the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. Test intervals for level monitoring have been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution shall be kept at least 10°F above the maximum saturation temperature to guard against boron precipitation. Minimum solution temperature is 48°F. This is 10°F above the saturation temperature for the maximum allowed sodium pentaborate concentration of 9.22 Wt. Percent.

Each parameter (concentration, pump flow rate, and enrichment) is tested at an interval consistent with the potential for that parameter to vary and also to assure proper equipment performance. Enrichment testing is required when material is received and when chemical addition occurs since change cannot occur by any process other than the addition of new chemicals to the Standby Liquid Control solution tank.

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENT

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.

	Item	Frequency
a.	Simulated Automatic Actuation test.	Once/Operating Cycle
b.	Pump Operability	When tested as specified in 3.13 verify that each core spray pump delivers at least 3300 GPM against a system head corresponding to a reactor vessel pressure of 104 psig
c.	Motor Operated Valve Operability	As specified in 3.13
d.	Core Spray Header Δp Instrumentation	

LIMITING CONDITION FOR OPERATION

3.5.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.

SURVEILLANCE REQUIREMENT

4.5.A Core Spray and LPCI Systems (cont'd)

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	When tested as specified in 3.13 verify that each LPCI pump delivers 4800 GPM at a head across the pump of at least 380 ft
c. Motor Operated valve operability	As specified in 3.13

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENT

4.5.B Containment Cooling System

1. Containment Cooling system Testing shall be as follows:

	Item	Frequency
a.	Pump Operability	When tested as specified in 3.13 verify that each RBCCW pump delivers 1700 GPM at 70 ft TDH and each SSW pump delivers 2700 GPM at 55 ft TDH
b.	Valve Operability	As specified in 3.13
c.	Air test on drywell and torus headers and nozzles	Once/5 years

LIMITING CONDITION FOR OPERATION

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.

SURVEILLANCE REQUIREMENT

C. HPCI System

1. HPCI system testing shall be performed as follows:

a. Simulated Automatic Actuation Test	Once/operating cycle
b. Pump Operability	When tested as specified in 3.13 verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 1000 psig
c. Motor Operated Valve Operability	As specified in 3.13
d. Flow Rate at 150 psig	Once/operating cycle verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 150 psig

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

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.

SURVEILLANCE REQUIREMENT

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	When tested as specified in 3.13 verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 1000 psig
c. Motor Operated Valve Operability	As specified in 3.13
d. Flow Rate at 150 psig	Once/operating cycle verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 150 psig

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

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%. The detailed bases is described in NEDC-31852P and summarized in Section 6.5 of the PNPS FSAR.

The analyses discussed in NEDC-31852P calculated a peak clad fuel temperature of less than 2200°F with a core spray pump flow of 3200 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 analyses in NEDC-31852P calculated a peak clad fuel temperature of less than 2200°F with LPCI pump flows of 4550 gpm, 4033 gpm, and 3450 gpm for two, three, and four pump combinations feeding into a single loop. A single pump flow rate of 4800 gpm ensures sufficient flow to meet or exceed the analyses' assumptions.

The analyses of LOCA for PNPS demonstrated the combination of LPCS/LPCI systems are sufficient to provide core cooling even with a single failure of either an active or passive safety-related component. The analyses determined there were four significant single failures that challenge the Emergency Core Coolant Systems' capability to prevent fuel damage during the postulated LOCA. They are:

1) Battery Failure - Loss of a single battery train could leave only one LPCS pump, two LPCI pumps, and ADS to mitigate the LOCA. This is the most limiting single failure for all but the largest postulated recirculation line breaks and for all postulated non-recirculation line breaks.

2) LPCI Injection Valve Failure - Loss of the injection valve selected by LPCI Loop Selection Logic for the pathway for all LPCI pumps' flow leaves two core spray pumps, HPCI, and ADS for LOCA mitigation. This becomes the limiting single failure for the largest postulated recirculation line breaks.

3) Loss of one emergency diesel generator - This leaves one LPCS pump, two LPCI pumps, and ADS for LOCA mitigation.

4) HPCI Failure - This leaves all other ECCS resources available. It is a significant failure primarily for small line breaks.

In all cases above, the remaining ECCS resources are sufficient to prevent PCT from exceeding 2200°F and other criteria provided in Section 50.46 and Appendix K of 10CFR50.

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.

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 x 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. Pump operability will be demonstrated during normal system operation and/or when system conditions allow capacity and performance testing in accordance with 3.13.

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 valves are tested in accordance with ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted) to assure their operability. The frequency and methods of testing are described in the PNPS IST program. The PNPS IST Program is used to assess the operational readiness of pumps and valves that are safety-related or important to safety. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and 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.

LIMITING CONDITIONS FOR OPERATION

3.6.D Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable. The nominal setpoint for the relief/safety valves shall be selected between 1095 and 1115 psig. All relief/safety valves shall be set at this nominal setpoint ± 11 psi. The safety valves shall be set at 1240 psig ± 13 psi.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours. Note: Technical Specifications 3.6.D.2 - 3.6.D.5 apply only when two Stage Target Rock SRVs are installed.

3. If the temperature of any safety relief discharge pipe exceeds 212°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.

SURVEILLANCE REQUIREMENTS

4.6.D Safety and Relief Valves

1. Testing of safety and relief/safety valves shall be in accordance with 3.13.

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. Whenever the safety relief valves are required to be operable, the discharge pipe temperature of each safety relief valve shall be logged daily.

4. Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment (Con't)

Primary Containment Isolation Valves

2.b. In the event any automatic Primary Containment Isolation Valve becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. (This requirement may be satisfied by deactivating the inoperable valve in the isolated condition. Deactivation means to electrically or pneumatically disarm, or otherwise secure the valve.)*

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under ORC approved administrative controls.

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment (Con't)

Primary Containment Isolation Valves

2.b.1 The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. Test primary containment isolation valves:

1. Verify power operated primary containment isolation valve operability as specified in 3.13.

2. Verify main steam isolation valve operability as specified in 3.13.

c. At least twice per week the main steam line power operated isolation valves shall be exercised by partial closure and subsequent reopening.

d. Verify reactor coolant system instrument line flow check valve operability as specified in 3.13.

2.b.2 Whenever a primary containment automatic isolation valve, is inoperable, the position of the isolated valve in each line having an inoperable valve shall be recorded daily.

3.7 Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber - reactor building vacuum breakers shall be operable at all times when primary containment integrity as required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber - reactor building breakers shall be 0.5 psig.

b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-pressure suppression chamber vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.4.b, c and d, below. Drywell-pressure suppression chamber vacuum breakers shall be considered operable if:

1) The valve is demonstrated to open with the applied force of the installed test actuator as indicated by the position switches and remote position indicating lights.

2) The valve shall return by gravity when released after being opened by remote or manual means, to within 3/32" of the fully closed position.

4.7 Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Verify operability of the pressure suppression chamber-reactor building vacuum breakers as specified in 3.13.

b. Check the associated instrumentation including set points for proper operation every three months.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Periodic Operability Tests

(1) Once each month each drywell-pressure suppression chamber vacuum breaker shall be exercised and the operability of the valve and installed position indicators and alarms verified.

(2) A drywell to suppression chamber differential pressure decay rate test shall be conducted at least every 3 months.

3.7 Primary Containment

3) Neither of the two position alarm systems which annunciate on Panel C-7 and Panel 905 when any vacuum breaker opening exceeds 3/32" are in alarm.

b. Any drywell-suppression chamber vacuum breaker may be non-fully closed as determined by the position switches provided that the drywell to suppression chamber differential decay rate is demonstrated to be not greater than 25% of the differential pressure decay rate for the maximum allowable bypass area of 0.2ft^2.

c. Reactor operation may continue provided that no more than 2 of the drywell-pressure suppression chamber vacuum breakers are determined to be inoperable provided that they are secured or known to be in the closed position.

d. If a failure of one of the two installed position alarm systems occurs for one or more vacuum breakers, reactor operation may continue provided that a differential pressure decay rate test is initiated immediately and performed every 15 days thereafter until the failure is corrected. The test shall meet the requirements of Specification 3.7.A.4.b.

5. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4% oxygen by volume with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

4.7 Primary Containment

b. During each refueling interval:

(1) Each vacuum breaker shall be tested to determine that the disc opens freely to the touch and returns to the closed position by gravity with no indication of binding.

(2) Vacuum breaker position switches and installed alarm systems shall be calibrated and functionally tested.

(3) At least 25% of the vacuum breakers shall be visually inspected such that all vacuum breakers shall have been inspected following every fourth refueling interval. If deficiencies are found, all vacuum breakers shall be visually inspected and deficiencies corrected.

(4) A drywell to suppression chamber leak rate test shall demonstrate that the differential pressure decay rate does not exceed the rate which would occur through a 1 inch orifice without the addition of air or nitrogen.

5. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded at least twice weekly.

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

APPLICABILITY:

Applies to ASME Code Class 1, 2 and 3 or equivalent pumps and valves.

OBJECTIVE:

To assure the operational readiness of ASME Code Class 1, 2, and 3 (Safety Related) or equivalent (important to safety) pumps and valves.

SPECIFICATION:

A. INSERVICE CODE TESTING OF PUMPS AND VALVES

1. Based on the Facility Commercial Operation Date, Inservice Code Testing of safety and safety-related pumps and valves shall be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components" Subsections IWP and IWV as required by 10CFR50.55a(g), except where specific relief has been granted by the NRC pursuant to 10CFR50.55a(g)(6)(i).

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

APPLICABILITY:

Applies to the periodic testing requirements of ASME Code Class 1, 2 and 3 or equivalent pumps and valves.

OBJECTIVE:

To assess the operational readiness of safety and safety-related pumps and valves by performance of inservice tests.

SPECIFICATION:

A. INSERVICE CODE TESTING OF PUMPS AND VALVES

1. Inservice Code Testing activities shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50.55a(g), with the exemptions and alternate testing that have been approved by the NRC pursuant to 10CFR50.55a(g)(6)(i). These exemptions and alternate testing are included in the PNPS Inservice Testing Program.

2. Test Frequencies for Code Terminology when performing Inservice Test activities.

Code Terminology	Frequencies
Weekly	7 Days
Monthly	31 Days
Quarterly or 3 Mths	92 Days
Semiannually/6 Mths	184 Days
9 Months	276 Days
Yearly/Annually	366 Days
Biannual/2 Yrs	732 Days

3. The provisions in Definitions (1.0) for REFUELING INTERVAL, SURVEILLANCE FREQUENCY, and SURVEILLANCE INTERVAL are applicable to Code testing and

3.13 INSERVICE CODE TESTING

4.13 INSERVICE CODE TESTING

to the above frequencies for performing Code testing activities.

4. Performance of Code testing shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall supersede the requirements of Technical Specifications.

3.13 and 4.13 Inservice Code Testing

The Limiting Conditions for Operation establishes the requirement that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the periodically updated edition of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10CFR50, Section 50.55a(g). These requirements apply except when relief has been requested pursuant to 10CFR50.55a(g)(6)(i) and granted by the NRC. The NRC may grant relief pursuant to 10CFR50.55a(a)(3)(i), 10CFR50.55a(3)(ii) or 10CFR50.55a(g)(6)(i).

The detailed procedures for testing of pumps and valves are documented in the PNPS Inservice Testing Program.

This specification includes a clarification of the frequencies for performing the testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in Surveillance Frequencies throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice testing activities.

Under the terms of this Specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example:

- Technical Specifications require components to be declared operable prior to entry into an operational mode. The ASME B&PV Code provision which allows pumps and valves to be tested up to one week after return to normal operation is superseded (and not allowed) by the more restrictive requirements of Technical Specifications.

- The allowance for a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable is superseded (and not allowed) by the more restrictive Technical Specification definition of operability which does not allow a grace period.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

The licensee proposed changes to their Technical Specifications (TSs) in a letter dated October 30, 1992, which included the following: 1) adding a new section in TS to specify limiting conditions for operation (LCO) and surveillance requirements for inservice testing (IST) of pumps and valves; 2) adding the definition of "Refueling Interval" to the TS definition section; 3) revising the definition of "Surveillance Interval" to allow the 25% tolerance on the testing frequency allowed by the ASME Code to apply to the Pilgrim refueling interval of 24 months; 4) modifying the surveillance frequency for specific pumps and valves which are tested in the IST program from monthly to quarterly; and 5) making miscellaneous changes to the Pilgrim TS and Bases to match the TS requirements with the IST program. In addition, by letters dated February 11, 1993 and March 29, 1993, changes were made to the Bases sections regarding core spray and LPCI system, and drywell temperature.

2.0 EVALUATION

2.1 Addition of TS Sections 3.13 and 4.13: Inservice Code Testing

The licensee has proposed to add TS Sections 3.13 and 4.13 to define the LCO and surveillance requirements for IST of safety-related pumps and valves. The proposed TS Sections 3.13 and 4.14 incorporate portions of TS Section 4.0.5 of the Standard TS for General Electric Boiling Water Reactors (BWR/5) which are contained in NUREG 0123, Revision 3, fall 1980.

The proposed TS Sections 3.13.A.1 and 4.13.A.1 state that IST of safety-related pumps and valves shall be based on the facility commercial operation date and performed in accordance with ASME Section XI, Subsections IWP and IWV, except where specific relief has been granted by the NRC. The proposed addition defines the LCO for IST such that all safety-related testing of pumps and valves at Pilgrim Nuclear Power Station shall be conducted in accordance with ASME Section XI, Subsections IWP and IWV. Any safety-related components not tested in accordance with TS Section 4.13.A.1 would violate the LCO. Therefore, the proposed TS section additions are acceptable.

The proposed TS Section 4.13.A.2 contains a list of Code test frequencies in terms of days. The proposed TS section is similar to Section 4.0.5.b of the BWR/5 Standard TS. Although Pilgrim is a BWR/3 vintage plant, the testing frequencies apply to IST of safety-related pumps and valves in all commercial nuclear power plants. The testing frequencies defined by the licensee are consistent with the existing Code terminology. Therefore, the proposed TS section addition is acceptable.

The proposed TS Section 4.13.A.3 states that the definitions in TS Section 1.0 for refueling interval, surveillance frequency, and surveillance interval are applicable to the frequencies defined in proposed TS Section 4.13.A.2. The licensee's proposal to modify the definitions in TS Section 1.0 are evaluated in Section 2.2 of this safety evaluation (SE). The proposed TS Section addition is an enhancement to the TS because it provides a specific reference to applicable TS definitions related to IST. Therefore, the proposed TS section addition is acceptable.

The proposed TS Section 4.13.A.4 states that performance of Code testing shall be in addition to other specified surveillance requirements. This is virtually identical to Section 4.0.5.d of the BWR/5 Standard TS. Also, the proposed TS Section 4.13.A.5 is identical to Section 4.0.5.d of the BWR/5 Standard TS and states that no Code testing shall supersede the TS. Although Pilgrim is a BWR/3 vintage plant, these proposed TS Sections apply to IST of safety-related pumps and valves in all commercial nuclear power plants. Therefore, the proposed TS section additions are acceptable.

2.2 Changes to TS Definition Section

2.2.1 Refueling Frequencies

The licensee has proposed to modify TS Section 1.0.P to be titled Refueling Frequencies, redesignate the definition of Refueling Outage to be TS Section 1.0.P.1, and add the definition of Refueling Interval as TS Section 1.0.P.2. The licensee stated that the Refueling Interval only applies to surveillance testing conducted in accordance with ASME Section XI and defines this interval as once every 24 months. This change specifically defines the testing frequency in the TS for components which can be tested only when the plant is shut down for refueling. Accordingly, the proposed TS section changes are acceptable.

2.2.2 Surveillance Interval

The licensee has proposed to modify the definition of Surveillance Interval in TS Section 1.0.V to include that the Refueling Interval is 24 months and that the allowable extension of the Surveillance Interval, discussed in TS Section 1.0.U, also applies to the Refueling Interval. This addition clarifies that the 25% extension of the Code test interval also applies to the Refueling Interval, which is allowed by the Code, and therefore ensures consistency between the Code and the TS. Accordingly, the proposed TS section change is acceptable.

2.3 Pump Testing Frequency

The licensee has proposed to revise the pump testing frequency requirements of the following pumps from monthly to quarterly: standby liquid control (SLC), TS Section 4.4.A; core spray (CS), TS Section 4.5.A.1.b; low pressure coolant injection (LPCI), TS Section 4.5.A.3.b; reactor building component cooling water (RBCCW) and salt service water (SSW), TS Section 4.5.B.1.a; high pressure coolant injection (HPCI), TS Section 4.5.C.1.b; and reactor core isolation cooling (RCIC), TS Section 4.5.D.1.b. These pumps are currently included in the licensee's IST program and are tested in the TS for operability on a monthly frequency. A pump flow test is conducted once every 3 months. The proposed changes would delete the monthly test and specify the pump test be conducted in accordance with Code testing as specified in TS Section 3.13. In addition, the licensee has proposed to modify each applicable TS section such that pump pressure (or head) and flow rate acceptance criteria are now verified in conjunction with the Code testing of Section 3.13.

The proposed TS Section 3.13 states that IST of safety-related pumps shall be conducted in accordance with the ASME Boiler and Pressure Vessel Code, Section XI (The Code). Paragraph IWP-3220 of the Code specifies that if a measured pump parameter enters the alert range, as defined in Table IWP-3100-2, then the testing frequency is doubled until the cause of the deviation is determined and the condition corrected. The pumps referenced in this TS change are currently tested monthly. Under the proposed TS change, if degradation occurs and a pump parameter falls into the alert range, the licensee would commence testing at an increased frequency until the problem is resolved. Therefore, for pumps with a degraded condition, the proposed TS would require surveillance at a frequency similar to the current testing frequency. Entering the Code-specified required action range would require the licensee to declare the pump inoperable. This aspect is unchanged by the TS change. In addition, NUREG 1366, "Improvements to Technical Specification Surveillance Requirements," issued December 1992, recommends that pump testing which is conducted more often than required by the Code be changed to a quarterly test frequency.

The staff believes that the Code quarterly-testing frequency, coupled with increased testing of pumps performing in the alert range, is adequate to detect and monitor pump condition. The NRC endorses the ASME Code, Section XI, and references this Code in 10 CFR 50.55a as the requirements of IST for pumps. Accordingly, the proposed TS changes are acceptable.

2.4 Valve Testing Changes

2.4.1 Motor Operated Valve (MOV) Testing Frequency

The licensee has proposed to revise the testing frequency of MOVs associated with the following systems: CS TS Section 4.5.A.1.c; LPCI, TS Section 4.5.A.3.c; RBCCW and SSW, TS Section 4.5.B.1.b; HPCI, TS Section 4.5.C.1.c; and RCIC, TS Section 4.5.D.1.c. These MOVs are currently being tested in the

TS on a monthly frequency. The licensee's proposed revision removes the monthly requirement and states that the valves are to be tested as specified in Section 3.13 of the TS.

The proposed TS Section 3.13 states that IST of safety-related valves shall be conducted in accordance with the ASME Boiler and Pressure Vessel Code, Section XI (The Code). With some exceptions, Paragraph IWV-3411 of the Code specifies that Category A and B valves shall be tested every 3 months. If a valve stroke time exceeds the previous stroke-time test as specified in IWV-3417a, then the valve testing frequency is increased to once per month until the problem has been corrected. Therefore, for valves with a degraded condition, the proposed TS would require surveillance at a frequency identical to that required by the current TS for MOVs tested monthly. If an MOV stroke time exceeds the licensee-specified limiting value, the valve will be declared inoperable. This aspect is unchanged by the TS change.

The NRC endorses the ASME Code, Section XI, and incorporates by reference this Code in 10 CFR 50.55a as the requirements of IST for valves. Accordingly, the proposed TS changes are acceptable.

2.4.2 Safety and Relief Valves (SRVs)

The licensee has proposed to replace the testing currently specified in TS Section 4.6.D.1 with a reference stating that SRVs shall be tested in accordance with TS Section 3.13. TS Section 3.13 states that IST of safety-related valves shall be conducted in accordance with ASME Section XI. The Pilgrim IST program currently is administered in accordance with the 1986 Edition of ASME Section XI. Paragraph IWV-3510 of this Edition specifies that testing of SRVs shall be conducted in accordance with ANSI/ASME OM-1-1981 (OM-1). Although testing in accordance with OM-1 may result in fewer SRVs tested during each refueling interval, OM-1 has been approved by the NRC for IST of SRVs. Therefore, the proposed TS change is acceptable.

2.4.3 Primary Containment Isolation Valves (CIVs)

The licensee has proposed to delete the requirements in TS Sections 4.7.A.2.b.1.b.1 and .2 and insert requirements that the primary CIVs and main steam isolation valves (MSIVs) will be tested as specified in TS Section 3.13.

All of the valves referenced in this TS change are tested in the licensee's IST program as currently specified in the TS. The TS revision does not change the testing frequency of these valves. Accordingly, the proposed TS section changes are acceptable.

The licensee has proposed to delete the requirements of TS Section 4.7.A.2.b.1.d and insert requirements that the reactor coolant system instrument line flow check valves will be tested as specified in TS Section 3.13. Relief Request RV-22 in the IST program was granted by the NRC to conduct testing of these valves once every refueling interval (reference NRC letter dated April 22, 1991). The testing frequency is changed from once per operating cycle to once per refueling interval. However, the refueling interval is specifically defined and these valves can only be tested when the plant is shut down for refueling. Therefore, the proposed TS changes are acceptable.

2.4.4 Reactor Building Vacuum Breakers

The licensee has proposed to delete the requirements for the pressure suppression chamber reactor building vacuum breakers from TS Section 4.7.A.3.a and add a reference that the pressure suppression chamber reactor building vacuum breakers be tested as specified in TS Section 3.13. In addition, the licensee has proposed to relocate the requirements for the associated instrumentation previously contained in TS Section 4.7.A.3.a to TS Section 4.7.A.3.b. Testing of the vacuum breakers and their associated instrumentation is currently conducted in accordance with the TS every 3 months.

The pressure suppression chamber reactor building vacuum breakers are currently tested in the licensee's IST program at a quarterly frequency. This testing consists of stroking the vacuum breaker open and verifying that the valve recloses. The testing frequency contained within the IST program is identical to the test frequency requirements currently specified in TS Section 4.7.A.3.a. The proposed TS changes do not change the test frequency or method and are, therefore, acceptable.

2.4.5 Scram Discharge Volume (SDV) Vent and Drain Valves

The licensee has proposed to delete the open verification and cycle requirements contained in TS Section 4.3.G.1 and add two new TS Sections, 4.3.G.1.a and 4.3.G.1.b. The proposed TS Section 4.3.G.1.a contains the requirement previously contained in TS Section 4.3.G.1 which states that the SDV vent and drain valves will be verified open at least once per month. The proposed TS Section 4.3.G.1.b states that the valves will be tested as specified in TS Section 3.13. In addition, TS Section 4.3.G.1.b adds the requirement previously contained in TS Section 4.3.G.1 that the valves may be closed intermittently under administrative control.

The SDV vent and drain valves are currently tested in the licensee's IST program at a quarterly frequency. This testing consists of cycling the solenoid valves without measuring individual valve stroke times. The testing contained within the IST program is identical to the testing requirements currently specified in TS Section 4.3.G.1. The proposed TS changes do not change the testing frequency or method and are, therefore, acceptable.

2.5 Standby Liquid Control (SLC) System

2.5.1 Pump Testing

The licensee has proposed to replace the surveillance requirement listed in Section 4.4.A.2.b to test the SLC system monthly by manually initiating the SLC system and recirculating boron solution back to the solution tank. The proposed quarterly test exceeds the current requirement because the test method used verifies the pump flow, system head, and the recirculation flow path, thus, satisfying the current once-per-cycle surveillance on a quarterly basis. The licensee is also proposing to change the once-per-cycle requirement to manually initiate the SLC system and pump demineralized water into the reactor vessel to perform this testing once per refueling interval. This test checks for proper operation of the explosive valves and proper operation of the pumps and valves. The licensee stated that quarterly testing of the pumps and refueling outage testing of the pumps and the explosive valves will be in accordance with the methods and frequencies prescribed in the Code. Therefore, the proposed changes are acceptable.

2.5.2 SLC Pump Relief Valve Testing

The licensee has proposed to replace the surveillance requirement currently contained in TS Section 4.4.A.2.a with a testing requirement that the each SLC pump relief valve set point shall be 1425 psig ± 43 psig during testing in accordance with TS Section 3.13 every refueling outage. This change establishes a tighter tolerance on the relief valve set point than the previous TS. The proposed set point criteria are consistent with current Code requirements. Accordingly, the proposed TS change is acceptable.

2.5.3 Miscellaneous Changes to SLC System TS

The licensee has proposed to change the phrase contained in the proposed TS Section 4.4.A.2.b from "pump operability" to "pump capacity." The licensee states that this change will more accurately describe the purpose of the test. TS Section 4.4.A currently states that the operability of the SLC system shall be verified, in part, by the performance of the test in the proposed TS Section 4.4.A.2.b. Therefore, substitution of the word "capacity" for "operability" will not change the requirements for the SLC pump. Accordingly, the proposed TS change is acceptable.

2.6 Miscellaneous TS Changes

The licensee is proposing to modify several TS Bases sections to be consistent with the proposed changes and additions. The affected bases sections include Section 3.4.A for the SLC system, Section 3.5.B for containment cooling, and Section 4.5 for core and containment cooling systems surveillance frequencies. The licensee proposes adding a Bases section for TS Sections 3.13 and 4.14 for inservice code testing. The proposed changes and additions to the TS Bases explain and support the changes in the TS. Accordingly, the proposed TS changes and additions are acceptable.

The licensee has proposed to delete TS Section 4.6.D.5 which requires SRVs that have been in service to be tested in the as-found condition during both Cycles 6 and 7. Pilgrim is currently in operating Cycle 10. This TS section is obsolete. Therefore, the licensee's proposed change is acceptable.

The licensee has proposed to change the term "refueling outage" to "refueling interval" in TS Sections 4.7.A.4.b, 4.7.A.4.b.(3), 4.3.G.2, and proposed TS Section 4.4.A.2.c, to establish a specific testing frequency of 24 months for the applicable surveillance. The change is for clarification and consistency with the new definitions. Therefore, the proposed changes to the TS sections are acceptable.

The licensee has proposed to renumber several TS subsections within TS Sections 4.4.A, 4.5.A, 4.5.B, 4.5.C, and 4.5.D that were either deleted or relocated as a result of this TS change submittal. These proposed changes are editorial changes which do not affect the TS requirements and are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 61108). Accordingly, the 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 the amendment.

5.0 CONCLUSION

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

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