August 22, 1994

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. 156 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M89628)

The Commission has issued the enclosed Amendment No. 156 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated June 9, 1994, as supplemented August 10, 1994.

This amendment increases the allowed out-of-service time from 7 days to 14 days for the automatic depressurization system, the high pressure coolant injection system, and the reactor core isolation cooling system. A change is also made to Section 4.5.H, "Maintenance of Filled Discharge Pipe" to reflect Amendment 149 issued September 28, 1993.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance and Final Determination of No Significant Hazards Consideration, and Opportunity for Hearing, will be included in the Commission's biweekly Federal Register 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. 156 to License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

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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. 156
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 June 9, 1994, and supplemented August 10, 1994, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

(for) Walter R. Butler
Charles Miller, Acting Assistant Director
for Region I Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 22, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 156

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
107	107
108	108
109	109
111	111
116	116
117	117
118	118

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 14 days unless such system is sooner made operable, providing that during such 14 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 14 days provided that during such 14 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.

LIMITING CONDITION FOR OPERATION

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 14 days unless such valve is sooner made operable, provided that during such 14 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.

SURVEILLANCE REQUIREMENT

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.

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 exceededing 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

(Intentionally left blank)

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 assure that the discharge piping of the core spray systems, LPCI system, HPCI and RCIC are filled:

1. Every month the LPCI system and core spray system discharge piping 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.

BASES:

3.5.C HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the HPCI System (FSAR 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 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 offset 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 14-day allowable repair time is specified.

The requirement that HPCI be operable when reactor coolant temperature is greater that 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 coolant to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The Station Nuclear Safety Operational Analysis, FSAR Appendix G, shows that RCIC also serves as redundant makeup system on total loss of all offset 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 14 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 Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the ADS (FSAR 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 14 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated June 9, 1994, and supplemented August 10, 1994, the Boston Edison Company (the licensee or BECo) submitted a request for changes to the Pilgrim Nuclear Power Station Technical Specifications (TSs). The requested changes would increase the allowed out-of-service (OOS) time from 7 days to 14 days for the automatic depressurization system (ADS), the high-pressure coolant injection (HPCI) system, and the reactor core isolation cooling (RCIC) system. A change is also proposed to Section 4.5.H "Maintenance of Filled Discharge Pipe." The monthly surveillance is retained, but the words connecting this requirement to the surveillance testing of low-pressure coolant injection (LPCI) and core spray (CS) are deleted.

2.0 EVALUATION

On receipt of an initiation signal, emergency core cooling system (ECCS) pumps automatically start; simultaneously, the system aligns and the pumps inject water, taken either from the condensate storage tank (CST) or suppression pool, into the reactor coolant system (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small, the HPCI system will maintain coolant inventory as well as vessel level while the RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with low pressure coolant injection (LPCI) and CS. In this event, the ADS timed sequence would be allowed to time out and open the selected safety/relief valves (S/RVs) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS cool the core.

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool is circulated through a heat exchanger cooled by the residual heat removal (RHR) service water system.

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Depending on the location and size of the break, portions of the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break. Although no credit is taken in the safety analysis for the RCIC system, it performs a similar function as HPCI, but has reduced makeup capability. Nevertheless, it will maintain inventory and cool the core while the RCS is still pressurized following a reactor pressure vessel (RPV) isolation. All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The HPCI system consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI system. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The ADS consists of 7 of the 11 S/RVs. It is designed to provide depressurization of the RCS during a small-break loss-of-coolant accident, (LOCA) if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of the S/RVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves. If HPCI is unavailable, the reactor vessel will not depressurize rapidly without assistance from the ADS. The ADS, coupled with with one of the several low pressure ECCS pumps, serves as a backup to HPCI for "small break" LOCA mitigation.

The HPCI system is designed to provide core cooling for a wide range of reactor pressures (162 psid to 1135 psid, vessel to pump suction). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full-flow test line is provided to route water from and to the CST to allow testing of the HPCI system during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS system discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI system is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

ADS provides some supplemental assistance to the low-pressure ECCS in the intermediate break size range. For postulated large pipe breaks where the water level changes rapidly, the vessel pressure also drops rapidly and the low pressure ECCS are the primary mitigating systems. The ADS is not designed for, nor required for, mitigation of large-break LOCA.

Pilgrim's ADS uses four (4) pressure relief valves mounted on the steam line ring header outside the vessel and inside the drywell. Steam is rejected to the torus when vessel level signals indicate that high-pressure coolant resources are incapable of maintaining adequate coolant coverage of the core. As a momentary low vessel level indication is not conclusive of high pressure coolant capacity inadequacy, timers delay the blowdown for 2-13 minutes before ADS actuation depending on whether drywell pressure is also high. If vessel level indications remain persistently below the low-low level setpoint during that period, blowdown will commence through the ADS valves, allowing low pressure coolant injection/spray resources to refill the vessel to satisfactory levels.

All four ADS valves are credited in the LOCA analysis. (NEDC-31852P "Pilgrim Nuclear Power Station SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis" September 1990).

The analyses of the limiting LOCA assuming one ADS valve is OOS shows a longer core uncovery period than the base case in NEDC-31852P; however peak centerline temperature (PCT) reaches 1500 °F for the limiting break size, well below the allowed the 2200 °F limit.

A steam line break outside containment (SLBOC) is classified as an accident and therefore must also consider the failure of a single component. The evaluation determined HPCI failure to be most limiting when one ADS is OOS. At 275 psig vessel pressure, the permissive level for LPCI and low pressure core spray (LPCS) is reached and the injection valves begin opening. The rate of depressurization changes, collapsing core voids, and level rapidly recovers, thereby halting PCT rise. The initial connection of the higher

pressure ECCS to the lower pressure vessel results in almost immediate restoration of cooling to the fuel clad hottest location. In each evaluated scenario, the burst of initial LPCI/LPCS flow is enough to reduce PCT before it exceeds its initial operating value.

A reactor water cleanup line break (RWCUB) was evaluated for one ADS valve OOS. It becomes the most limiting nonrecirculation line break when one ADS valve is OOS. Being a break outside containment, 13 minutes elapses before blowdown occurs, yielding a PCT of 699 °F.

The HPCI system also provides some supplemental assistance to the low-pressure ECCS in the intermediate break size range. For postulated large pipe breaks where the water level changes rapidly, the vessel pressure also drops rapidly and the low-pressure ECCS are the primary mitigating systems. Since HPCI is a steam turbine powered system utilizing reactor vessel steam pressure as the power source, it is not relied upon for mitigation of large-break local.

In addition to mitigating the consequences of small pipe breaks, the HPCI system also provides a source of inventory makeup to maintain adequate core cooling during isolation type events such as the loss of feedwater. The function of HPCI is redundant to the RCIC for these events.

The justification to extend HPCI OOS to 14 days is:

- RCIC, although not credited in Pilgrim's accident analysis, is adequate to mitigate the lesser vessel coolant inventory transients if HPCI is unavailable. Such an event might involve a low rate of inventory loss or low core decay heat condition.

- For more significant transients or accidents involving vessel inventory loss, the unavailability of HPCI will affect the consequences of a given event. However, the consequences do not exceed those for events already analyzed in the final safety analysis report (FSAR). The nuclear fuel operating limit criteria (LHGR, MAPLHGR, MCPR, etc.) currently established for protection of fuel safety limits would protect against fuel damage if the HPCI was OOS during any postulated transient or accident.

The SAFER/GESTR-LOCA analysis for Pilgrim (NEDC 31852P), assumes HPCI is available after a 90-second delay after the initiating ECCS signal. In the small break range (0.1 ft^2), the PCT for nominal conditions did not exceed 900 °F.

Without HPCI being available, the peak clad temperature for GE 8X8EB/NB fuel reaches 970 °F for nominal conditions and 1111 °F when Appendix K (10CFRPart 50) assumptions are applied. Since the limiting PCT, both nominally and per Appendix K, is the large break for which HPCI contributes virtually no benefit, the conclusion is that HPCI's temporary period of inoperability would result in no consequence to fuel integrity during a LOCA. The primary consequence of HPCI unavailability is the increased probability that vessel

blowdown may be necessary to permit LPCI/LPCS flows to terminate fuel heat-up if RCIC flow is either inadequate or unavailable.

The analysis also included 0.05 and 0.15 ft^2 breaks. The sensitivity of HPCI unavailability to PCT associated with a LOCA was demonstrated to maximize at the 0.1 ft^2 level. During a loss of all feedwater event, HPCI is the primary emergency coolant resource to avoid core uncovery. RCIC is a backup.

The RCIC is a steam driven coolant injection system designed to deliver 400 gpm of coolant from the condensate storage tank or suppression pool at vessel pressures above 150 psig. The RCIC system provides a backup to the HPCI system but has a capacity less than 10% of HPCI. The primary justification for extending the OOS for RCIC is that its small coolant capacity makes it a minor contributor to accident mitigation. When available, HPCI flow capacity can maintain core coolant coverage or prevent fuel damage. When HPCI is not available, core uncovery could occur for a loss-of-feedwater event. However, this is only for a short period until automatic vessel blowdown on low-low reactor water level allows low pressure coolant injection and spray to restore the water level. Fuel heatup resulting from this is negligible.

TS Section 4.5.H.1 requires the monthly venting of the LPCI and CS discharge piping high point to test these systems. Amendment 149 changed the surveillance frequency for the LPCI and CS systems from monthly to quarterly in conformance with the inservice testing (IST) program. However, Pilgrim intends to continue the monthly venting of the LPCI and CS discharge piping. The monthly venting is consistent with STS.

The licensee has requested changes to increase the allowed OOS time from 7 days to 14 days for the ADS, the HPCI system, and the RCIC system. The Bases are also changed to reflect the increased OOS time. Increasing the allowed OOS time provides additional time to make repairs. The increased time can be used to procure parts when such are not readily available onsite. The increased OOS time can also contribute to plant availability by potentially averting a shutdown compelled by the expiration of the 7-day clock. Application of a 14-day OOS time for ADS, HPCI, and RCIC is consistent with boiling-water reactor revised Standard Technical Specifications. The additional seven days requested by the licensee is acceptable because it is a short time period and the likelihood of a random single failure in conjunction with a LOCA occurring during the LCO period of 14 days is very small.

The NRC staff finds the above TS changes and analysis acceptable. A change is also proposed to Section 4.5.H "Maintenance of Filled Discharge Pipe." The monthly surveillance is retained, but the words connecting this requirement to the surveillance testing of LPCI and CS are deleted. The changes to TS Section 4.5.H.1 are administrative in nature and are found acceptable by the staff.

3.0 EMERGENCY CIRCUMSTANCES

The RCIC system serves as a standby source of cooling water to provide limited decay heat removal whenever the main feedwater is isolated from the reactor vessel. Although RCIC does provide some supplemental assistance to HPCI during a postulated LOCA, this is not a design basis requirement for the system. The RCIC flow is considerably smaller than the HPCI flow and is therefore not an important contributor to LOCA mitigation. The Pilgrim LOCA analyses were performed without taking credit for RCIC.

The RCIC system is a steam driven coolant injection system designed to deliver 400 gpm of coolant from the condensate storage tank or suppression pool at vessel pressures above 150 psig. The RCIC system provides a backup to the HPIC system but has a capacity less than 10% of HPCI. Examples of such scenarios where RCIC may be used are (1) loss-of-feedwater and (2) steam line break outside containment. To a lesser extent, RCIC would contribute to coolant inventory makeup for more major coolant inventory loss scenarios, e.g., recirculation line rupture LOCA at pressure, whether HPCI was available or not.

On Wednesday, August 3, 1994, during the operability surveillance RCIC tripped on high-steam flow at startup. Subsequent investigation revealed that the cause for the trip was that the governor control valve failed to respond to the control system demand because of valve binding. Preliminary investigation into why the control valve was binding identified that the alignment pins were not properly aligned. The valve was rebuilt and properly aligned and post-work testing (PWT) followed.

At approximately 0400 on Friday, August 5, 1994, RCIC was run for post-work testing. During the surveillance, it was noted the oil level on the coupling end bearing decreased while the level rose in the governor end bearing. This occurred approximately 15 minutes into the run. Shortly after, fluctuations in speed were identified, and a small amount of oil was noted on the RCIC skid. The turbine was manually tripped at that time and an investigation into the change in oil level began.

Based on discussions with the vendor (Dresser Rand) and troubleshooting observations, the cause for the change in bearing oil levels was due to air becoming entrained in the oil. The air forms a bubble in the drain line from the governor end bearing, retarding oil drainage. This failure to drain properly increases the bearing oil level. Since oil is not draining from this bearing, the only source of oil to the sump and the oil pump is from the coupling end, lowering the oil level. Dresser Rand stated that the most likely cause is the pumping action of the trip disk which aerates the oil; however, other sources such as pump suction piping leaks could have been the cause. On August 10, 1994, the licensee asked that the amendment submitted on June 9, 1994, be processed on an emergency basis. These proposed amendment changes increase the allowed OOS time from 7 days to 14 days for the ADS the HPCI and the RCIC.

The primary justification for extending the OOS for RCIC is that its small coolant capacity makes it a minor contributor to accident mitigation. When HPCI is not available, core uncovery could occur for a loss-of-feedwater event. However, this is only for a short period until automatic vessel blowdown on low-low reactor water level allows low pressure coolant injection and spray to restore the water level. Fuel heatup resulting from this is negligible.

Safety criteria used to determine the acceptability of extending continued operation with the RCIC system OOS is consistent with Pilgrim's licensing basis. For example, events with the expected frequency of occurrence greater than once-per-reactor lifetime are required to meet the transient MCPR thermal limit: more than 99.9% of the fuel rods are expected to avoid boiling transition. Very low probability events, such as a LOCA, are required to satisfy the criteria of 10CFR50.46; the primary criterion being that the PCT be maintained less than 2200 °F.

The NRC staff granted orally on August 10, 1994, PNPS's request for enforcement discretion associated with increasing the OOS time from 7 to 14 days for RCIC to be effective until the staff's review of the proposed license amendment is completed. This enforcement discretion was confirmed by the NRC letter to PNPS, dated August 12, 1994.

The staff determined that the request for amendment warranted an emergency basis in order to prevent an unnecessary plant shutdown.

The staff concluded that the exercise of enforcement discretion in this instance involved minimum safety impact and was satisfied that it was warranted from a public health and safety perspective.

The NRC staff does not believe that PNPS has abused the emergency provisions of 10 CFR 50.91(a)(5) in this instance. The licensee originally applied for the amendment in a nonemergency basis. Subsequent to that submittal, the licensee discovered air bubbles in the lubricating oil, a condition, not previously observed. The licensee needed the additional OOS time to address the phenomena. Upon discovering the problem, the licensee promptly sought emergency relief in order to prevent an unnecessary shutdown. In accordance with 10 CFR 50.91(a)(5), the Commission has determined that emergency circumstances exist warranting prompt action. The situation could not have been avoided, and the licensee and the Commission must act quickly and time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment. The Commission has also determined that the amendment, as discussed in Section 4.0, does not involve a significant hazards consideration.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has made a final determination that the amendment involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92(c), this means that the operation of the facility in accordance with

the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards considerations, which is presented below:

1. The Operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Safety criteria used to determine the acceptability of extending continued operation with one ADS valve, the HPCI or RCIC system out-of-service (OOS) is consistent with Pilgrim's licensing basis. For example, events with the expected frequency of occurrence greater than once-per-reactor lifetime are required to meet the transient MCPR [minimum critical power ratio] thermal limit: more than 99.9% of the fuel rods are expected to avoid boiling transition. Very low probability events, such as a LOCA [loss-of-coolant accident], are required to satisfy the criteria of 10CFR50.46: the primary criterion being that the Peak Cladding Temperatures (PCT) be maintained less than 2200°F.

   For intermediate frequency events, e.g. safe shutdown in the event of a fire, 10CFR50 Appendix R involves a "no fuel damage" criterion. To evaluate these types of events, the GE [General Electric] SAFER/GESTR-LOCA licensing methodology was used to calculate the system responses and PCTs.

   Analyses performed by Pilgrim's NSSS [nuclear steam supply system] vendor, General Electric, for various limiting-case scenarios involving ADS, HPCI, or RCIC out-of-service situations demonstrated 10CFR50.46 limits (i.e. a PCT less than 2200°F) were met. (The most severe PCT was 1500°F). The core damage frequency analysis for Pilgrim is unchanged by operating Pilgrim in accordance with this proposed amendment. The 14 day OOS for HPCI, RCIC and ADS also conforms to the OOS time for these systems found in BWR [boiling-water reactor] Standard Technical Specifications. Hence, increasing the allowed OOS time from 7 to 14 days does not result in a challenge to fuel cladding integrity or BWR Standard Technical Specifications, and operating Pilgrim in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The removal of the association between LPCI [low-pressure coolant injection] and Core Spray system testing and surveilling their filled discharge pipes is an administrative change because the specified surveillance frequency is unchanged. This proposed change reflects Amendment #149, issued by the NRC September 28, 1993, and is proposed to ensure consistency between Pilgrim's Technical Specification sections. This administrative change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   As discussed above, a variety of limiting-case scenarios were analyzed to demonstrate the effects of increasing the OOS time for one ADS valve, the HPCI system, or the RCIC system. The conclusion of the analyses is that this proposed change does not violate Pilgrim's licensing basis or 10CFR50.46 requirements.

   Some scenarios result in elevated PCTs, but they are still significantly below the 10CFR50.46 limit of 2200°F. Therefore, since the licensing-basis and code required PCT continues to be met and because the proposed change comports the requirements of BWR Standard Technical Specifications, operating Pilgrim in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   As discussed in above question 1, the proposed change to section 4.5.H.1 is administrative and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   Certain scenarios analyzed for system unavailability result in evaluated PCTs. However, these elevated PCTs are significantly below the 10CFR50.46 limit of 2200°F. Therefore, there is no reduction in the safety margin for PCT resulting from the change from 7 to 14 days. The proposed change also corresponds to the requirements of BWR Standard Technical Specifications concerning OOS for HPCI, RCIC and ADS. Therefore, operating Pilgrim Station in accordance with this proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, concludes that the standards of 50.92(c) are satisfied. Therefore, the NRC staff has determined that the request involves no significant hazards consideration.

5.0 STATE CONSULTATION

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

6.0 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 made a final no significant hazards consideration finding with respect to this amendment. 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.

7.0 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.

Principal Contributor: Karen Cotton

Date: August 22, 1994