BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

10CFR50.90

BECo 94-068

E. T. Boulette, PhD
Senior Vice President — Nuclear

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Docket 50-293

PROPOSED TECHNICAL SPECIFICATION CHANGE TO
INCREASE ADS, HPCI AND RCIC ALLOWED OUT OF-SERVICE TIMES TO 14 DAYS

Boston Edison Company (BECO) hereby proposes the attached modification to Appendix A of Operating License No. DPR-35 in accordance with 10CFR50.90. This proposed change 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 proposed change is also made to Section 4.5.H "Maintenance of Filled Discharge Pipe" to reflect Amendment #149 issued September 28, 1993.

Technical Specification Bases for these systems are modified to reflect the increased allowed out of service interval.

The requested changes are described in Attachment A. The revised Technical Specification pages are provided in Attachment B. Attachment C provides the existing pages marked-up to show the proposed changes.

E T Boulette
E.T. Boulette, PhD
Senior Vice President - Nuclear

Commonwealth of Massachusetts)
County of Plymouth)

Then personally appeared before me, E. T. Boulette, who being duly sworn, did state that he is Senior Vice President - Nuclear of Boston Edison Company and that he is duly authorized to execute and file the submittal contained herein in the name and on behalf of Boston Edison Company and that the statements in said submittal are true to the best of his knowledge and belief.

My commission expires: October 5, 1995 Peter M Kahler
DATE NOTARY PUBLIC

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Attachments: (A) Description of Proposed Change
(B) Amended Technical Specification Pages
(C) Marked-up Pages from current Technical Specifications

1 signed original and 37 copies

cc: Mr. R. Eaton, Project Manager
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
Mail Stop: 14D1
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

U. S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Senior NRC Resident Inspector
Pilgrim Nuclear Power Station

Mr. Robert M. Hallisey, Director
Radiation Control Program
Massachusetts Department of Public Health
305 South Street
Jamaica Plain, MA 02130

PMK/nas/ADSHPCI

Proposed Change

Changes are proposed to 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 (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 LPCI and Core Spray are deleted.

Bases are changed to reflect the increased OOS time.

Reason for Change

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 (BWR) Standard Technical Specifications.

Section 4.5.H.1 requires the monthly venting of the LPCI and Core Spray discharge piping high point prior to testing these systems. Amendment #149 changed the surveillance frequency for the LPCI and Core Spray 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 Core Spray discharge piping. Therefore, we are changing 4.5.H.1 to unlink the venting requirements from the testing of the LPCI and Core Spray systems. The monthly venting is consistent with Standard Technical Specifications.

Justification for Changes

ADS Evaluation

Currently, PNPS may continue to operate for seven (7) days after one Automatic Depressurization System (ADS) valve is made or found inoperable. The seven day LCO is based on system redundancy and HPCI operability, which provides redundant core protection for small break events. To support operation with one ADS valve out-of-service for fourteen days the ADS function of the Safety Relief Valves (SRV) was reviewed, including the accidents and abnormal operational transients associated with operation with one ADS valve out-of-service. In addition, the ADS system design basis and the safety criteria were also reviewed.

The primary design goal of the ADS system is to provide rapid depressurization in the reactor vessel to mitigate the consequences of postulated "small break" LOCAs. The "small break" range is approximately $0.15ft^2$ and smaller for postulated breaks in the recirculation line, which is the most limiting break location for core cooling. If HPCI is unavailable, the reactor vessel will not depressurize rapidly without assistance from the ADS. The ADS, coupled with one of the several low pressure Emergency Core Cooling System (ECCS) pumps, serves as a backup to HPCI for "small break" LOCA mitigation.

ADS 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. 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 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 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 ECC systems to the lower pressure vessel results in almost immediate restoration of cooling to the fuel clad's 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 non-recirculation 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.

HPCI Evaluation

Currently, PNPS may continue to operate for seven (7) days after the High Pressure Coolant Injection (HPCI) system is made or found inoperable. The seven day LCO is based on RCIC, ADS and low pressure ECCS operability. To support operation with the HPCI system out-of-service for fourteen days, the accidents and abnormal operation transients associated with operation with the HPCI system out-of-service were reviewed. In addition, the HPCI system design and the safety criteria were also reviewed.

The primary design goal of the HPCI system is to provide core cooling to mitigate the consequences for postulated "small break" LOCAs. The "small break" range is approximately 0.0 to 0.15ft^2 for postulated breaks in the recirculation line, which is the most limiting break location for core cooling. During a postulated LOCA for this range, HPCI is sufficient to prevent core uncovery. If HPCI is unavailable, the reactor vessel will not depressurize rapidly without assistance from the ADS. The ADS, coupled with one of the several low pressure ECCS pumps, serves as a backup to HPCI for "small break" LOCA mitigation.

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 LOCAs.

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 HPCI system is a steam driven coolant injection system designed to deliver 4250 gpm of coolant from the condensate storage tank or suppression pool at vessel pressures above 150 psig. This is the primary source of high pressure emergency coolant resources as the companion Reactor Coolant Isolation Cooling (RCIC) system supplies less than one-tenth of the HPCI capacity. However, because RCIC is not credited in the PNPS safety analyses in the event of HPCI OOS, low pressure coolant injections systems must be relied upon to assure adequate coverage. Examples of such scenarios are loss of feedwater and steam line break outside containment. However, since RCIC flow is small, its failure is not significant for more major coolant inventory loss scenarios, e.g., recirculation line rupture (LOCA) at pressure, whether HPCI is available or not. If HPCI is out-of-service, the worst additional failure during an accident involving vessel inventory loss would be a component that reduces or delays low pressure coolant injection capacity.

Hence, 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 are not worse than events already analyzed in the 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^oF.

Without HPCI being available, the peak clad temperature for GE 8X8EB/NB fuel reaches 970^oF for nominal conditions and 1111^oF when Appendix K (10CFR50) 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.

When feedwater flow is lost, a low water level scram terminates prompt neutron power, leaving about 7% decay power which rapidly decays to lower levels. Voids collapse in the core and steam production ceases, leaving vessel inventory temporarily unchanged. The subcooling of the vessel water must be reduced to zero before significant steaming and vessel inventory loss can begin. However, upon reaching low-low reactor water level, RCIC would begin adding 400 gpm of coolant. To convert this to steam takes about 1280 BTU/LBM or 75 megawatts for the total flow. This is about 3.75% of rated power, a level to which decay heat would fall to in less than 4 minutes after a reactor scram on low level. Given the large initial downcomer and lower plenum inventory of subcooled water to be heated to saturated enthalpy, only localized boiling in the core can be expected, resulting in relatively minor steam production, partially compensated for by incoming RCIC flow. After about four minutes of power decay and RCIC flow, the net vessel heat gain becomes negative until normal water level is restored and RCIC is throttled back. Thus, a significant margin of core coverage is maintained.

The two additional analyses of loss-of-feedwater include failures that require vessel depressurization and low pressure coolant makeup resources to avoid fuel heat up. Heat up of cladding is negligible because initial core coolant uncovery occurs several minutes after the scram resulting in a brief uncovery period of lower power, and steam cooling is available to the exposed cladding from boiling lower in the core.

A steam line break outside containment (SLBOC) is another event resulting in reactor isolation at high pressure. Unlike loss-of-feedwater, vessel pressure initially drops to about 800 psig eliminating most of the subcooling of the vessel coolant inventory. Credit for RCIC flow is not taken for this accident. Hence boiloff of coolant inventory is expected when HPCI is unavailable with subsequent discharge out the SRV's as boiling raises pressure in the isolated vessel. However, a large amount of energy must be expended to convert each pound of water to steam and the heat source continues to decay, dropping by a factor of 5 in 30 minutes. Hence, 28-1/2 minutes pass before the top of the core uncovers. The worst single failure in this case is the failure of one ADS valve. Again, the late uncovery means slow and limited fuel heat up due to power decay. In fact, the fuel clad temperature never exceeds its initial value after dropping down the saturated temperature curve with pressure as ADS blows down the vessel.

RCIC EVALUATION

Currently, PNPS may continue to operate for seven (7) days after the Reactor Core Isolation Cooling (RCIC) system is made or found inoperable. The seven day LCO is based on HPCI operability. To support operation with the RCIC system out-of-service for fourteen days, the accidents and abnormal operational transients associated with operation with the RCIC system out-of-service were reviewed. In addition, the RCIC system design basis and the safety criteria were also reviewed.

The RCIC system serves as a standby source of cooling water to provide a 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 pig. The RCIC system provides a backup to the HPCI 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 make-up for more major coolant inventory loss scenarios, e.g., recirculation line rupture (LOCA) at pressure, whether HPCI was available or not.

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 heat-up resulting from this is negligible.

Safety Evaluation and Determination of No significant Hazards Considerations

The Code of Federal Regulations (10CFR50.91) requires licensees requesting an amendment to provide an analysis, using the standards in 10CFR50.92, that determines whether a significant hazards consideration exists. The following analysis is provided in accordance with 10CFR50.91 and 10CFR50.92 for the proposed amendment.

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 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 Peak Cladding Temperatures (PCT) be maintained less than 2200^{o}F.

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

Analyses performed by Pilgrim's NSSS vendor, General Electric, (summarized above under "Justification for Changes") 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^{o}F) were met. (The most severe PCT was 1500^{o}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 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 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 $2200^{o}F$. 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 elevated PCTs. However, these elevated PCTs are significantly below the 10CFR50.46 limit of $2200^{o}F$. 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.

As discussed above, the administrative change to section 4.5.H.1 does not involve a significant reduction in a margin of safety.

This proposed change has been reviewed and recommended for approval by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Schedule of Change

This change will become effective 30 days following BECo's receipt of the Commission's approval.