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

October 4, 1996

SUBJECT: ISSUANCE OF AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M95326)

Dear Mr. Boulette:

The Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated May 1, 1996.

The amendment revises the Technical Specifications (TSs) to reflect the implementation of 10 CFR Part 50, Appendix J, Option B at the Pilgrim Nuclear Power Station. The amendment changes the TSs to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide (RG) 1.1.63, "Performance-Based Containment Leakage-Rate Testing Program." RG 1.163 was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," which you have adopted, provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described in the safety evaluation.

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

Sincerely,

/S/

Alan Wang, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 167 to License No. DPR-35
2. Safety Evaluation

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

October 4, 1996

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Boston Edison Company
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Project Directorate I-1
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Docket No. 50-293

Enclosures: 1. Amendment No. 167 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

DATED: October 4, 1996

AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-35-PILGRIM NUCLEAR POWER STATION

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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. 167
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 May 1, 1996, 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.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to the start of Refueling Outage 11.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 167

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
3/4.7-4	3/4.7-4
3/4.7-5	3/4.7-5
B3/4.7-3	B3/4.7-3
B3/4.7-4	B3/4.7-4
B3/4.7-5	B3/4.7-5
B3/4.7-6	B3/4.7-6
B3/4.7-7	B3/4.7-7

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

Primary Containment Integrity

2. a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 Mw(t).

Primary containment integrity means that the drywell and pressure suppression chamber are intact and that all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All blind flanges and manways are closed.

4. All automatic primary containment isolation valves and all instrument line flow check valves are operable except as specified in 3.7.A.2.b.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

Primary Containment Integrity

2. a. The primary containment integrity shall be demonstrated by performing Primary Containment Leak Tests in accordance with 10CFR50 Appendix J, Option B and Regulatory Guide 1.163 dated September 1995*, with exemptions as approved by the NRC and exceptions as follows:

1. The main steam line isolation valves shall be tested at a pressure ≥23 psig, and normalized to a value equivalent to P_a.

2. Personnel air lock door seals shall be tested at a pressure ≥10 psig. Results shall be normalized to a value equivalent to P_a.

3. Leakage rate acceptance criteria are:

1. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with the Containment Leakage Rate Testing Program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

2. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$

3. Door seals leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 10 psig.

* Definition 1.U is not applicable to Leak Rate Tests.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

5. All containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolated position.

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 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

4. Combined main steam lines: 46 scfh @ 23 psig.

where P_a = 45 psig
L_a = 1.0% by weight of the contained air @ 45 psig for 24 hrs.

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.

A. Primary Containment (Cont)

Primary Containment Testing

The primary containment pre-operational test pressures were based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak drywell pressure is about 45 psig which would rapidly reduce to 27 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5%/day at a pressure of 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.25%/day at 45 psig. Calculations made by the AEC staff with this leak rate and a standby gas treatment system filter efficiency of 95% for halogens and assuming the fission product release fractions stated in TID 14844, show that the maximum total whole body passing cloud dose is about 13 REM and the maximum total thyroid dose is about 110 REM at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population zone distance of 4.3 miles are about 3 REM total whole body and 70 REM total thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site dose and 10CFR100 guidelines.

The maximum allowable test leak rate (L_a) is 1.0%/day at a pressure of 45 psig. This value for the test condition was derived from the maximum allowable accident leak rate of 1.25%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 as determined from the guide on containment testing.

Establishing the test limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness

capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate or the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leakage rate testing is based on the guidelines in Regulatory Guide 1.163 dated September 1995, NEI 94-01 Revision 0 dated July 26, 1995, and ANSI/ANS 56.8-1994. Specific acceptance criteria for as-found and as-left leakage rates, as well as methods of defining the leakage rates, are contained in the primary containment leakage rate testing program.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is in accordance with 10CFR50 App. J, Option B and Regulatory Guide 1.163 dated September 1995.

Type A, Type B and Type C tests will be performed using the technical methods and techniques specified in ANSI/ANS 56.8 - 1994, or other alternative testing methods approved by the NRC.

A note is included in Surveillance 4.7.A.2.a stating that definition 1.U is not applicable. The 25% allowable extension of surveillance intervals is already included in the primary containment leakage rate testing program, therefore an additional 25% is not allowed.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

Group 1 - process lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, main steam space high temperature, or reactor vessel high water level.

Group 2 - isolation valves are closed by reactor vessel low water level or high drywell pressure. The group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 2 isolation signal by a transient or spurious signal.

Group 3 - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

Group 4 and 5 - process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided.

Group 7 - The HPCI vacuum breaker line is designed to remain operable when the HPCI system is required. The signals which initiate isolation of the HPCI vacuum breaker line are indicative of a break inside containment and reactor pressure below that at which HPCI can operate.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance, prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

A. Primary Containment (Cont)

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment. A program for periodic testing and examination of the excess flow check valves is in place.

Primary Containment Painting

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, assures the paint is intact. Experience at Pilgrim Station and other BWR's with this type of paint indicates that the inspection interval is adequate.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One valve may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 10 drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression system tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 5 psig vacuum; therefore, with two vacuum relief valves secured in the closed position and eight operable valves, containment integrity is not impaired.

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is 0.2 ft^2, which is equivalent to all vacuum breakers open 3/32". (See letters from Boston Edison to the Directorate of Licensing, dated May 15, 1973 and October 22, 1974)

A. Primary Containment (Cont)

Reactor operation is not permitted if differential pressure decay rate is demonstrated to exceed 25% of allowable, thus providing a margin of safety for the primary containment in the event of a small break in the primary system.

Each drywell suppression chamber vacuum breaker is equipped with three switches. One switch provides full open indication only. Another switch provides closed indication and an alarm should any vacuum breaker come off its closed seat by greater than 3/32". The third switch provides a separate and redundant alarm should any vacuum breaker come off its closed seat by greater than 3/32". The two alarms above are those referred to in Section 3.7.A.4.a.3 and 3.7.A.4.d.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance. Mark I Containment Long Term Program testing showed that maintaining a drywell to

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated May 1, 1996, Boston Edison Company (the licensee) requested changes to the Technical Specifications (TSs) for Pilgrim Nuclear Power Station. The revised TS references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leakage Test Program," which specifies a method acceptable to the NRC for complying with Option B dated September 1995. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B at the Pilgrim Plant by adopting the definitions and acceptance criteria proposed by the staff for implementation of Option B.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Containment Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995 (60 FR 49495), and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

RG 1.163 was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TSs. The licensee has referenced RG 1.163 in the Pilgrim TSs.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's May 1, 1996, letter to the NRC proposes to revise the TSs to reflect the implementation of 10 CFR Part 50, Appendix J, Option B at the Pilgrim Station. The amendment changes the TSs to implement 10 CFR Part 50, Appendix J, Option B, by referring to RG 1.1.63, "Performance-Based Containment Leakage-Rate Testing Program." RG 1.163 references NEI 94-01 and ANSI/ANS-56.8-1994 as described above. The revised TSs will reference RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires a change to existing TS 4.7, "Primary Containment Integrity." Corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B, and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The licensee has proposed changes to surveillance requirement TS 4.7.A.2 to reference 10 CFR Part 50, Appendix J, Option B, and RG 1.163, "Performance-Based Containment Leakage-Rate Testing Program." The licensee has also redefined P_a, L_a, and the leakage rate acceptance criteria to be consistent with the model TS provided to NEI by letter dated November 2, 1995. The staff did note that the licensee did not adopt all the acceptance criteria of the NEI document for the primary containment. By conference call on June 27, 1996, the staff discussed this with the licensee and the licensee agreed that the NEI wording was applicable to Pilgrim. The licensee supplemented its application by modifying the proposed TS to add the following to the primary containment criteria as proposed in the NEI document: "During the first unit startup following testing in accordance with the Containment Leakage Rate Testing Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests and $< 0.75\ L_a$ for the Type A tests." This modification is within the scope of the original notice of the application. In addition, consistent with the model the licensee has noted that definition 1.U (test interval extension) is not applicable to the Leak Rate Test as Option B already includes a 25% extension of the test interval. Previous exemptions approved by the staff and exceptions noted in the TSs were maintained.

The staff has reviewed these proposed changes and concluded that, despite the different format of the licensee's current TSs, all the important elements of the guidance regarding Type A, B, and C testing provided in the NRC letter to NEI are included in the TS proposed by the licensee and the proposed changes are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

4.0 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.

5.0 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 (61 FR 28606). 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.

6.0 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: A. Wang

Date: October 4, 1996