FEB - 6 1974

Docket No. 50-293

Boston Edison Company
ATTN: Maurice J. Feldman
Vice President, Operations
and Engineering
800 Boylston Street
Boston, Massachusetts 02199

Change No. 3
License No. DPR-35

Gentlemen:

Your letter dated December 14, 1973, requested certain modifications to the Technical Specifications of Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. The modifications involve:

1. Eliminating surveillance requirements on systems or components during periods when they are not required.

2. Allowing the removal of more than two control rod drives at times when there is no fuel in the reactor vessel.

3. Deleting the suppression pool water volume requirements when there is no fuel in the reactor vessel.

4. Increasing the allowable total leak rate specification for testable penetrations and isolation valves to 60% of L_{to} (45) and 60% L_{to} (23), respectively.

We have evaluated the proposed changes and have determined that these proposed changes do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the Pilgrim Nuclear Power Station in the manner proposed. A copy of our related Safety Evaluation is enclosed.

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Two additional changes which also were proposed in your letter required expedited review to permit the unloading of the Pilgrim core and were acted on separately by the Regulatory staff as Amendment No. 2 to Facility Operating License No. DPR-35 dated January 7, 1974.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility Operating License No. DPR-35 are hereby changed by replacing the existing pages 86, 152, 154 and 155 with the enclosed pages and adding a new page 5a.

Sincerely,

/S/

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages 5a, 86, 152, 154 and 155

cc w/enclosures:
See next page

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cc w/enclosures:
Mr. Dale G. Stoodley, Counsel
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Mr. J. Edward Howard, Superintendent
Nuclear Engineering Department
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Mr. Grant Baston, Pilgrim Division Head
Boston Edison Company
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Plymouth Public Library
North Street
Plymouth, Massachusetts 02360

Mr. L. D. Weislogel
Quality Assurance & Reliability Manager
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Mr. Paul Sears
Chairman, Board of Selectmen
Town of Plymouth
Plymouth, Massachusetts 02133

Henry Kolbe, M. D.
Acting Commissioner of Public Health
Massachusetts Department of Public Health
600 Washington Street
Boston, Massachusetts 02111

Mr. Hans L. Hamester
ATTN: Joan Sause
Office of Radiation Programs
Environmental Protection Agency
Room 647A East Tower, Waterside Mall
401 M Street, S. W.
Washington, D. C. 20460

Mr. Wallace Stickney
Environmental Protection Agency
JFK Federal Building
Boston, Massachusetts 02203

Anthony Z. Roisman, Esquire
1712 N Street, N. W.
Washington, D. C. 20036
(w/cy of BECo's ltr dtd 12/14/73)

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

BOSTON EDISON COMPANY

DOCKET NO. 50-293

CHANGE NO. 3 TO THE TECHNICAL SPECIFICATIONS OF THE

PILGRIM NUCLEAR POWER STATION

FACILITY OPERATING LICENSE NO. DPR-35

Introduction

By letter dated December 14, 1973, Boston Edison requested changes to the Technical Specifications appended to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. The proposed changes involve:

1. Eliminating surveillance requirements on systems or components during periods when they are not required.
2. Allowing the removal of more than two control rod drives at times when there is no fuel in the reactor vessel.
3. Deleting the suppression pool water volume requirements when there is no fuel in the reactor vessel.
4. Increasing the allowable total leak rate specification for testable penetrations and isolation valves to 60% of L_{to} (45) and 60% L_{to} (23), respectively.

Discussion

These changes were originally proposed by the licensee during a meeting held between the licensee and representatives of the Regulatory staff on May 24, 1973, in Bethesda and have been the subject of additional discussion between the staff and Boston Edison since that time.

The first change proposed by the licensee would in effect waive the requirements for periodic checking, testing, and calibration of instruments and equipment during those periods in which these instruments, equipment or systems were not required to be operable. This waiver was also requested in BECo's letter dated December 14, 1973. The licensee pointed out that it is not possible to conduct certain of the specified tests when the reactor is shut down and the reactor vessel head has been removed; e.g., the weekly tests for HPCI and RCIC operability. Nor are there any functional requirements on these two systems during the described condition. The licensee agreed to check, test, or calibrate the instruments and equipment prior to declaring them operable or as soon as practicable thereafter.

Changes 2 and 3 above are for a relief from restrictions which have relevance only when fuel is present in the reactor.

Change 4 asks that the technical specifications relating to the limits established for the allowable total leak rate associated with leak tests of testable penetration and isolation valves be changed to reflect conformance with Appendix J of 10 CFR Part 50 which was published subsequent to preparation of the present Pilgrim Technical Specifications.

Evaluation

The staff's evaluation of the four proposed changes is as follows:

(1) In reviewing the request eliminating surveillance requirements on systems or components when they are not required to be operable, we concluded that it was never the intent of the Commission to require that an item of equipment or a system be maintained in an operable condition when the reactor is in a mode or condition in which that equipment or system is not required to serve any safety related function or protection against any accident condition. In fact, this intent is illustrated by several of the notes to the tables in Sections 4.1 and 4.2 of the Technical Specifications. Therefore, we conclude that the addition

of the definition "Surveillance Interval" to the Technical Specifications does not reduce the safety provisions of that document. Further, we find the provisions for restoration of inoperative equipment to be acceptable as stated. This definition is added to the Technical Specifications by addition of page 5a.

With the addition of the definition for "Surveillance Interval" as described above, the Regulatory staff concludes that it is appropriate to add the definition "Surveillance Frequency" to clarify our intent with regard to the frequency and timing of inspections and tests to be conducted by the licensee. This definition of Surveillance Frequency is consistent with that given in other Technical Specifications recently issued; e.g., Peach Bottom II and III, Browns Ferry and Cooper. Therefore, this change is a further step in seeking uniformity and clarity in the Technical Specifications for operating BWRs. We concluded that this change does not modify the safety provisions of the Technical Specifications and is therefore acceptable. This definition is included on the new page 5a.

(2) In reviewing the request to allow removal of more than two control rods when there is no fuel in the reactor vessel, we concluded that neither the control rods nor the control rod drives serve a safety objective when no fuel is in the reactor vessel, and the requirements on control rod operability are not appropriate for the stated condition. Therefore, we have added a sentence to technical specification 3.3.F stating, "Specification 3.3.A through D do not apply when there is no fuel in the reactor vessel". This addition satisfies the above request and recognizes that it is not the intent to require that equipment be maintained in an operable condition when its operation can serve no safety function. This change is implemented by a revision to page 86 of the Technical Specifications.

(3) Our review of this proposed change deleting the suppression pool water volume requirements when there is no fuel in the reactor vessel concluded that there are no safety requirements for maintaining the suppression pool flooded or cooled to 90°F as required by this specification when there is no fuel in the reactor vessel. Under this condition there is no requirement for the primary containment (or containment pressure suppression system) to be maintained. This is illustrated by specification 3.7.A.2 which requires the primary containment integrity to be maintained only when ". . . fuel is in the reactor vessel ....". Similarly, there is no requirement for the suppression pool to serve as a heat sink for condensing discharged steam from the ADS relief valves, the RCIC turbine, or the HPCI turbine as illustrated by specifications 3.5.E.1, 3.5.D.1, and 3.5.C.1 which require the respective subsystems be operable only " . . . whenever there is irradiated fuel in the reactor vessel . . .". Also, there is no requirement for the suppression pool to serve as a source of emergency cooling water during a loss-of-coolant accident unless fuel is in the reactor vessel. This is illustrated by specifications 3.5.A.1, 3.5.B.1, and 3.5.C.1 which incorporate the above quoted phrase. We, therefore, conclude that the requested change does not compromise any safety requirement and have revised specification 3.7.A.1 on page 152 to make it conditional upon the presence of fuel in the reactor vessel.

(4) We conclude that the total leakage rates for all testable penetrations and isolation valves should be raised to 60% to be consistent with the requirements of 10 CFR 50, Appendix J, and to be consistent with the requirements in other recently issued Technical Specifications. In reviewing this matter, we further conclude that this modification does not increase the allowable operational leak rate and that the protection provided by the containment remains unchanged. The revisions to the Technical Specifications are shown as indicated on pages 154 and 155.

Conclusion

We have concluded that the issuance of these Changes does not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. Consequently, these technical specifications should be changed as set forth in Change No. 3 to the Technical Specifications of Facility Operating License No. DPR-35.

Paul W O'Connor
Paul W. O'Connor
Operating Reactors Branch #2
Directorate of Licensing

Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: February 6, 1974

1.0 DEFINITIONS (Cont'd)

U. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

V. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable.

3.3.D Control Rod Accumulators

1. Inoperable accumulator.

2. Directional control valve electrically disarmed while in a non-fully inserted position.

3. Scram insertion time greater than the maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

E. Reactivity Anomalies

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

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

4.3.D Control Rod Accumulators

E. Reactivity Anomalies

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

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water volume - 84,000 ft^3
   b. Maximum water volume - 94,000 ft^3
   c. Maximum suppression pool temperature during normal power operation - 80°F
   d. Maximum suppression pool temperature during RCIC, HPCI or ADS operation - 130°F.
   e. In order to continue reactor power operation after being on RCIC, HPIC or ADS operation, the suppression chamber pool temperature must be reduced to 80°F within 24 hours following the return to reactor power operation.

2. 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).

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day.

2. Integrated Leak Rate Testing

   a. The primary containment integrity shall be demonstrated by performing an Integrated Primary Containment Leak Test (IPCLT) in accordance with either Method A or Method B, as follows:

      Method A

      Perform leak rate test prior to initial unit operation at the test pressure of 45 psig, $P_t(45)$, to obtain measured leak rate $L_m(45)$, or

      Method B

      Perform leak rate test prior to initial unit operation at the test pressures of 45 psig, $P_t(45)$, and 23 psig, $P_t(23)$, to obtain the measured leak rates, $L_m(45)$ and $L_m(23)$, respectively.

4.7.A Primary Containment (Cont'd)

c. Corrective Action for IPCLT

Methods A and B

If leak repairs are necessary to meet the allowable operational leak rate, the integrated leak rate test need not be repeated provided local leakage measurements are conducted and the leak rate differences prior to and after repairs, when corrected to the test pressure and deducted from the integrated leak rate measurements, yield a leakage rate value not in excess of the allowable operational leak rate.

d. Frequency for IPCLT

Methods A and B

After the initial pre-operational leakage rate test, two integrated leak rate tests shall be performed at approximately equal intervals between the major shutdowns for inservice inspection conducted at ten-year intervals. In addition, an integrated leakage rate test shall be performed at the end of the ten-year interval, which may coincide with the inservice inspection shutdown period.

e. Local Leak Rate Tests (LLRT)

Methods A and B

(1) Primary containment testable penetrations and isolation valves shall be tested at a pressure of ≧45 psig, except for the main steamline isolation valves which shall be tested at a pressure of ≧23 psig, each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once each operating cycle.

(2) Personnel air lock door seals shall be tested at a pressure of ≧10 psig each operating cycle.

f. Acceptance Criteria and Corrective Action for LLRT

Method A

If the total leakage rates listed below are exceeded, repairs and re-tests shall be performed to correct the conditions.

(1) Double-gasketed seals 10% $L_{to}(45)$

(2) All testable penetrations and isolation valves 60% $L_{to}(45)$

(3) Any one penetration or isolation valve except main steam line isolation valves 5% $L_{to}(45)$

4.7.A Primary Containment (Cont'd)

(4) Any one main steam line isolation valve 11.5 scf/hr @23 psig.

METHOD B

If the total leakage rates listed below as adjusted to a test pressure of 23 psig are exceeded, repairs and retests shall be performed to correct the condition.

(1) Double-gasketed seals - $10\%L_{to}(23)$

(2) (a) Testable penetrations and isolation valves $60\% \ L_{to}(23)$

(b) Any one penetration or isolation valve except steamline isolation valves $5\%L_{to}(23)$

(c) Any one main steamline isolation valve 11.5 scf/hr @23 psig

Leak rates measured at 45 psig shall be adjusted to a test pressure of 23 psig according to:

$$LLRT(23)_{adj} = LLRT_{meas} \times \frac{L_m(23)}{L_m(45)}$$

g. Continuous Leak Rate Monitor

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

h. Drywell Surfaces

The interior surfaces of the drywell and torus above water line shall be visually inspected each operating cycle for evidence of deterioration.

Change No. 3 dtd
February 6, 1974