JUL 23 1974

Docket No. 50-293

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 3 to Facility License No. DPR-35. This amendment includes Change No. 5 to the Technical Specifications, Appendix A, and is in response to your requests dated October 16, 1973, February 15, 1974, and March 4, 1974.

This amendment changes the bases which support certain technical specifications relating to overpressure transients to reflect the results of a reanalysis of these transients and also increase the required standby liquid control system boron concentration.

The Safety Evaluation and the Federal Register Notice relating to this action are also enclosed.

Local PDR as given to R. Graham & mailing RWG 2:30

Sincerely,

Original signed by:
Karl R. Goller

Karl R. Goller
Assistant Director
for Operating Reactors
Directorate of Licensing

New cover pages and pages 98, 99, 100 + 101, sent to BECO by telecopier 7/25/74 1:30 p.m. RO

Sent by telecopier 7/23/74 at 5:30 RWG

Actual pages of Tech Spec not sent

Enclosures:
1. Amendment No. 3
2. Safety Evaluation
3. Federal Register Notice

617-746-7905

Cover to pgs + pages sent 7/24/74 at 1:35 p.m. RO

RJSchemel
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DATE	7/19/74	7/22/74	7/22/74	7/22/74	7/23/74	

Form AEC-318 (Rev. 9-53) AECM 0240

U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-35

1. The Atomic Energy Commission (the Commission) has found that:

   A. The applications for amendment by Boston Edison Company (the licensee) dated October 16, 1973, February 15, 1974, and March 4, 1974, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-35 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 5".

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

FOR THE ATOMIC ENERGY COMMISSION

Karl R. Goller

Karl R. Goller
Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 5 to Appendix A
Technical Specifications

Date of Issuance: July 23, 1974

ATTACHMENT TO LICENSE AMENDMENT NO. 3

CHANGE NO. 5 TO TECHNICAL SPECIFICATIONS (APPENDIX A)

FACILITY OPERATING LICENSE NO. DPR-35

The attached pages supersede pages bearing the same number, except as otherwise indicated. The revised pages have marginal lines indicating where the changes appear.

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Page 126

BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

2.1 The abnormal operational transients applicable to operation of the Pilgrim Station have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 1998 MWt at 100 percent recirculation flow. The analyses were based upon plant operation in accordance with the operating map given in Fig. 3.7-1 of the FSAR. In addition, 1998 MWt is the licensed maximum power level of Pilgrim, and this maximum steady-state power will never be knowingly exceeded.

Transient analyses were not performed for a power level that specifically included instrument errors. To permit appropriate conclusions from analyses which do not include instrument errors, conservatism was incorporated in the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamics performance.

The void reactivity coefficient utilized in the analysis is conservatively estimated to be about 25% larger than the most negative value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 80% of the 5
control rods. The scram delay time and rate of rod insertion used in the analysis are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The insertion of the first dollar of reactivity strongly turns the transient and the stated 10% insertion time conservatively accomplishes this desired initial effect. The time for 50% and 90% insertion are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The design peaking factors at the full power conditions for the Pilgrim Nuclear Power Station result in an MCHFR value of 2.63. For analyses of the thermal consequences of the transients, higher peaking factors are used, so that an MCHFR of 1.9 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

BASES:

2.2 The valve sizing analysis considered four, 10% capacity relief/safety valves and two 8% capacity safety valves. These are sized and set pressures are established in accordance with the following three requirements of Section III of the ASME Code: 5

1. The lowest safety valve must be set to open at or below vessel design pressure and the highest safety valve be set at or below 105% of design pressure.

2. The valves must limit the reactor pressure to no more than 110% of design pressure.

3. Protection systems directly related to the valve sizing transient must not be credited with action (i.e., an indirect scram must be assumed).

A main steam line isolation with flux scram has been selected to be used as the safety valve sizing transient since this transient results in the highest peak vessel pressure of any transient when analyzed with an indirect scram. The original FSAR analysis concluded that the peak pressure transient with indirect scram would be caused by a loss of condenser vacuum (turbine trip with failure of the bypass valves to open). However, later observations have shown that the long lengths of steam lines to the turbine buffer the faster stop valve closure isolation and thereby reduce the peak pressure caused by this transient to a value below that produced by a main steam line isolation with flux scram.

Item 3 above indicates that no credit be taken for the primary scram signal generated by closure of the main steam isolation valves. 5
Figure 10* shows that two other scram initiation signals would be generated, one at 2.7 seconds due to high neutron flux and one at approximately 3 seconds due to high reactor pressure. Thus item 3 will be satisfied by assuming a scram due to high neutron flux.

Relieving capacity of 40% (4 relief/safety valves) results in a peak pressure of 1240 psig in the vessel dome and 1269 psig at the vessel bottom during the transient conditions used in the valve sizing analysis.

*Attachment to Boston Edison's March 4, 1974 letter which provides BECo's transient reanalysis.

2.2 BASES

The relief/safety valve settings satisfy the Code requirements that the lowest safety valve set point be at or below the vessel design pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief/safety valve actuation is required are given in Section 14 of the Safety Analysis Report. 5

In the expected case, wherein the scram signal generated by closure of the main steam isolation valves does in fact initiate a scram, the set point and capacity of the relief/safety valves is sufficient to remove enough energy from the reactor to prevent the safety valves from lifting (Figure 14.0-7 of FSAR).

LIMITING CONDITION FOR OPERATION

3.3.B Control Rods

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the Reactor Engineer, either:

   a. Both RBM channels shall be operable: or

   b. Control rod withdrawal shall be blocked: or

   c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

	% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
	10	.55
	30	1.275
5	50	2.00
	90	5.00

SURVEILLANCE REQUIREMENT

4.3.B Control Rods

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

1. After each refueling outage all fully withdrawn insequence rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to achieving the RUN mode of operation. The scram times shall be measured without reliance on the control rod drive pumps. Within 24 hours after exceeding 10% design power, all untested operable control rods shall be scram tested as described above.

LIMITING CONDITION FOR OPERATION

3.3.C **Scram Insertion Time**

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

5

% Inserted From Fully Withdrawn	Avg. Scram Insertion Time Sec.
10	.58
30	1.35
50	2.12
90	5.30

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

SURVEILLANCE REQUIREMENT

4.3.C **Scram Insertion Time**

2. At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. After attaining full power level, 20 of the operable control rods, selected to be uniformly distributed throughout the core, shall be scram-time tested at full reactor pressure at the time intervals listed below following any outage exceeding 72 hours in duration: 1 week, 2 weeks, 4 weeks, 8 weeks, 16 weeks and continuing at 16 week intervals:

   a. If the arithmetic mean for 90% insertion time of the tested control rod drive increases by more than 0.25 seconds, or 90% insertion time exceeds 3.5 seconds then an additional sample of 20 control rods, selected to be uniformly distributed throughout the core, shall be scram tested.

BASES:

In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately
5 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C."

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $k_{eff} < 1.0$ - other repeating rod sequences with more rods withdrawn resulted in $k_{eff} > 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

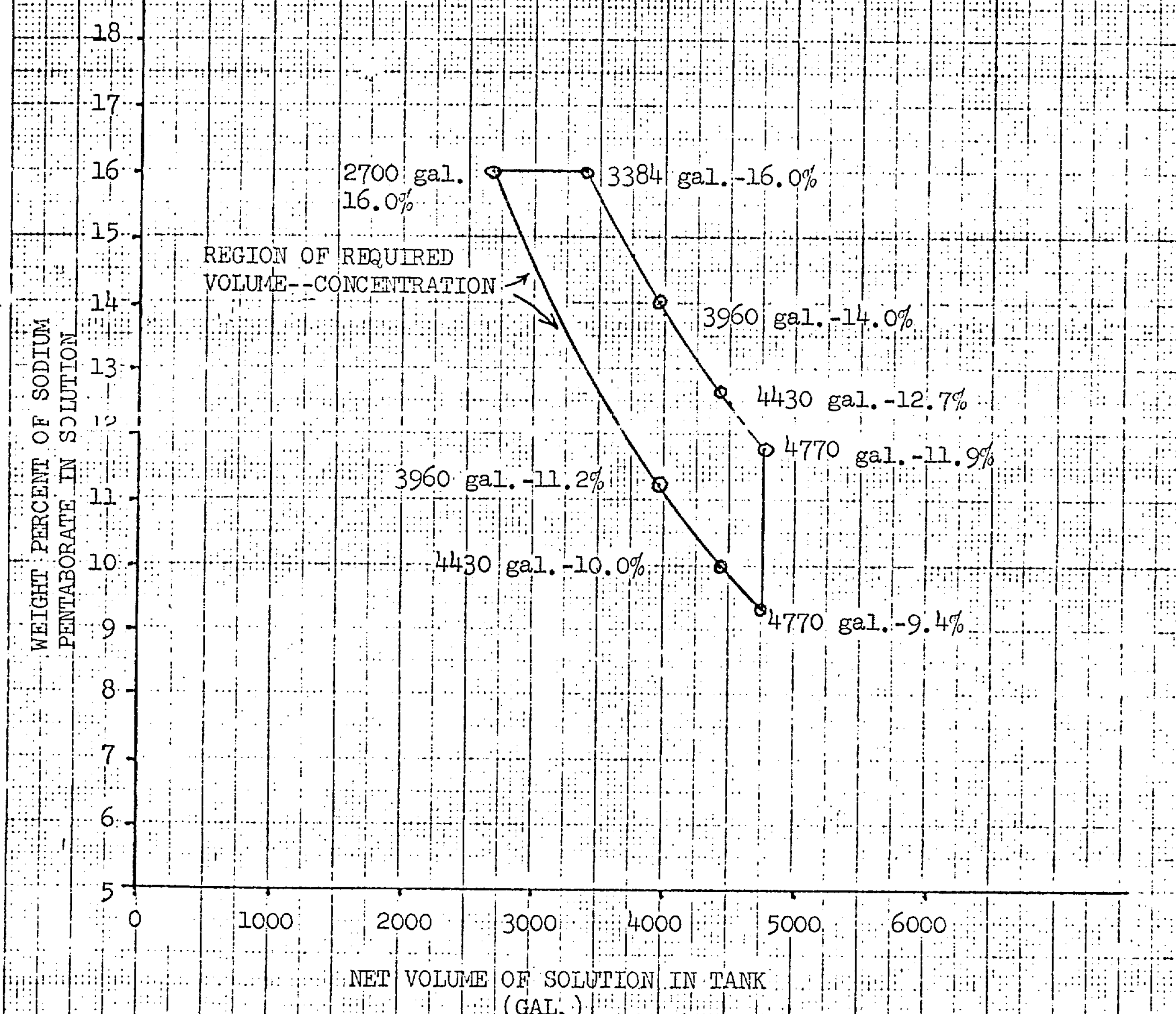

FIGURE 3.4.1
SODIUM PENTABORATE SOLUTION
VOLUME AND CONCENTRATION
REQUIREMENTS

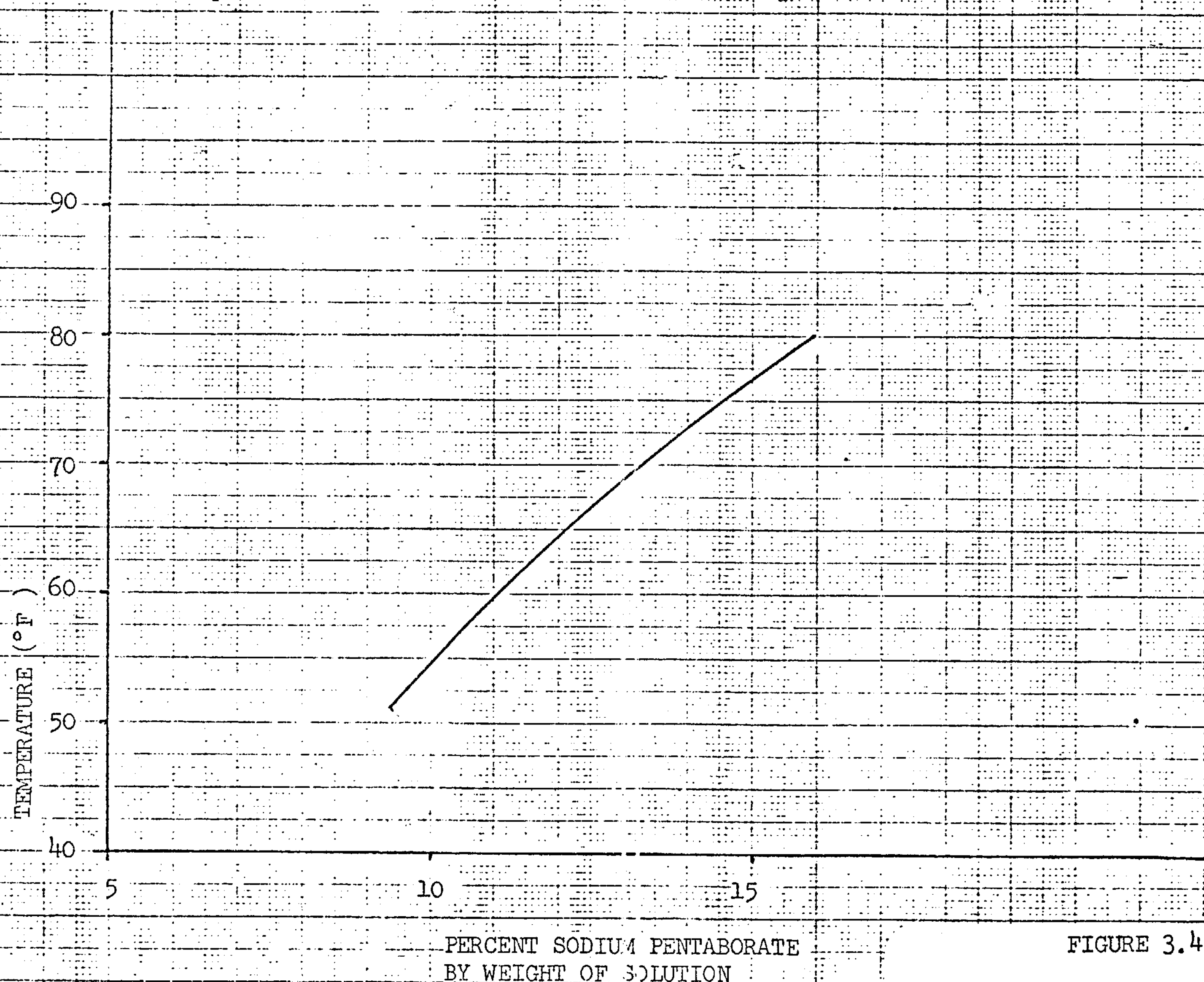

FIGURE 3.4.2

TEMPERATURE REQUIREMENTS OF THE SODIUM PENTABORATE SOLUTION

BASES:

3.4 STANDBY LIQUID CONTROL SYSTEM

A. The conditions under which the Standby Liquid Control System must provide shutdown capability are identified via the Station Nuclear Safety Operational Analysis (Appendix G). The requirements of this specification are taken from the Operational Nuclear Safety Requirements of subsection 3.8.6 of the Final Safety Analysis Report. If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 700 ppm of
boron in the reactor core in less than 125 minutes. 5
The 700 ppm concentration in the reactor core is required to bring the reactor from full power to a five percent Δ k subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the

3.4 STANDBY LIQUID CONTROL SYSTEM (Cont'd)

remaining system will perform its intended function and that the long term average availability of the system is not reduced is obtained for a one out of two system by an allowable equipment out of service time of one third of the normal surveillance frequency. This method determines an equipment out of service time of ten days. Additional conservatism is introduced by reducing the allowable out of service time to seven days, and by increased testing of the operable redundant component.

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

The solution is kept at least 10°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

The volume concentration requirement of the solution are such that should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature-concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 2700 gal. of solution having a 16.0 percent sodium pentaborate concentration is required to meet this shutdown requirement. For the minimum required pumping rate of 39 gpm, the maximum net storage volume of the boron solution is established as 4770 gal. 5

3.6.C Coolant Chemistry (Cont'd)

power operation is permissible only during the succeeding seven days.

3. If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable.

5

3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant

4.6

D. Safety and Relief Valves

1. At least one safety valve and two relief/safety valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

The set point of the safety valves shall be as specified in Specification 2.2.

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

3. The integrity of the relief/safety valve bellows shall be continuously monitored and the operability of

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 3 TO LICENSE NO. DPR-35

(CHANGE NO. 5 TO APPENDIX A OF TECHNICAL SPECIFICATIONS)

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

INTRODUCTION

By applications dated October 16, 1973, February 15, 1974, and March 4, 1974, Boston Edison requested changes to the Technical Specifications appended to Facility Operating License No. DPR-35. The proposed changes involve:

1. Revising the bases which support certain Technical Specifications relating to overpressure transients to reflect assumptions used when reanalyzing these transients to take into account a change in the scram reactivity curve.

2. Increasing the required scram reactivity insertion rate.

3. Increasing the sodium pentaborate solution concentration in the standby liquid control system.

4. a. Deleting Specification 3.6.D.2.b. which permits the safety valve function of two relief safety valves to be inoperable for 7 days.

   b. Changing the allowable out-of-service interval for the safety valve function of one relief safety valve in Specification 3.6.D.2.a. from 30 days to 7 days.

Boston Edison has determined that a change in the shape of the scram reactivity curve for Pilgrim Station will occur with increasing fuel exposure. This change will reduce the reactivity inserted during the beginning of control rod stroke. A reanalysis of those transients which could be affected by the change in scram reactivity curve has been carried out utilizing a scram reactivity curve which is applicable through the end of cycle 2. The reanalysis also incorporates other refined assumptions including an increased control rod insertion rate and the use of actual performance parameters of installed equipment rather than parameters selected prior to operation.

Additionally, the effectiveness of the liquid poison injected by the standby liquid control system has been lessened by core changes associated with cycle 2 operation. Boston Edison will increase the sodium pentaborate solution concentration to maintain the 5%Δk/k shutdown margin used as the design basis for this system.

EVALUATION

Item No. 1

Boston Edison's reanalysis of those transients which could be affected by changes in the scram reactivity curve shows that the limiting transient for relief valve sizing is the turbine trip without bypass. Using the scram reactivity curve representing end of cycle 2 performance, the calculated peak steamline pressure resulting from a turbine trip without bypass is 1180 psig. The pressure margin to the safety valve setting is 60 psi. This margin is greater than the design criteria minimum margin of 25 psi and is acceptable.

The safety valve sizing transient is the closure of the main steam isolation valves with a failure of the MSIV position switch scram. This transient, using end of cycle 2 parameters, shows that the calculated peak pressure at the bottom of the vessel is 106 psi below the 1375 psig allowed by ASME Boiler and Pressure Vessel Code, Section III, and is acceptable.

The proposed changes to the bases of Specification 2.1, 2.2, and 3.3 revise these sections to reflect the assumptions used in the Boston Edison transient reanalysis which established that adequate safety margins have been established and are therefore acceptable.

Item No. 2

The proposed change to Technical Specification 3.3 increases the required control rod insertion rate for the Pilgrim Station control rods. This change helps offset some of the reactivity reduction associated with the scram reactivity curve changes. This change is consistent with the assumptions used in the transient analysis and is acceptable because it results in an increase in the margin of reactor safety.

Item No. 3

This change increases the required boron concentration for the standby liquid poison system to assure that the design criteria of 5%Δk/k shutdown margin is maintained. This change maintains the margin of reactor safety and is therefore acceptable.

Item No. 4

The previous Technical Specifications authorized operation of the facility for up to thirty days with the safety function of one of the relief/safety valves inoperable and up to seven days with the safety function of two relief/safety valves inoperable. The proposed change would delete the authorization to operate with two relief/safety valves inoperable and authorize operation of the facility for up to seven days with the safety function of one relief/safety valve inoperable. The reanalysis of the response of Pilgrim 1 to the safety valve sizing transient assumes the operation of all four relief/safety valves and demonstrates a 106 psi margin to the ASME Code pressurization limit. We concur with the deletion of the authorization to operate with two relief/safety valves out of service but because no analysis was submitted with one relief/safety valve inoperable, the proposed provision for allowing one relief/safety valve to be inoperable for seven days has not been authorized.

CONCLUSION

The staff concludes that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) that such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

James J Shea for
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: July 23, 1974

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-293

BOSTON EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-35 issued to the Boston Edison Company which revised Technical Specifications for operation of the Pilgrim Nuclear Power Station, located in Plymouth County, Massachusetts. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications of the Pilgrim Nuclear Power Station by:

1. Revising the bases which support certain technical specifications relating to overpressure transients to reflect assumptions used when reanalyzing these transients to take into account a change in the scram reactivity curve.
2. Increasing the required scram reactivity insertion rate.
3. Increasing the sodium pentaborate solution concentration in the standby liquid control system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

For further details with respect to this action, see (1) the applications for amendment dated October 16, 1973, February 15, 1974, and March 4, 1974, (2) Amendment No. 3 to License No. DPR-35, with any attachments, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Plymouth Public Library on North Street in Plymouth, Massachusetts 02360.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 23rd day of July, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Dennis L. Ziemann

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