Attachment A

Technical Specification Change to APRM Setpoints

Proposed Change

Reference is made to Pilgrim Station Operating License No. DPR-35 Appendix A. The bases for Specifications 2.1.A.1 and 2.1.B contain information pertaining to the APRM High Flux Scram (Run Mode) trip setting and the APRM Rod Block (Run Mode) trip setting as follows:

2.1.A.1 Bases (Page 16)

"The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than 3.06 (8X8 array)."

2.1.B Bases (Pages 17 & 18)

"As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking exceeds (3.06) for 8 X 8 fuel, thus preserving the APRM rod block safety margin."

The desired changes consist of:

In 2.1.A.1 bases delete "3.06 (8X8 array)" and add the following:

"....the design value. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain adjust."

In 2.1.B bases delete "3.06" and replace with "the design value," and add "As with the scram setting, this may be accomplished by adjusting the APRM gain."

Reason for Change

Pilgrim Nuclear Power Station Technical Specifications require the adjustment of the APRM high flux scram and APRM rod-block settings whenever the actual maximum total peaking factor in the reactor core exceeds a specified value. The requirement to adjust these settings ensures that adequate safety margins are maintained for all core power distributions. The current Technical Specifications do not address the method to be used to accomplish the setting adjustment of the APRM circuits.

This amendment would clarify the Technical Specification requirement by changing the bases to identify the method to be used to adjust the APRM circuit settings.

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Safety Considerations

Changing the gains in the APRM amplifiers is equivalent to adjusting the set points because in either case, the circuits would perform their safety function in an identical manner for identical conditions in the reactor core. These changes do not present any hazard considerations not described or implied in the license application as amended. These changes have been reviewed by the Nuclear Safety Review and Audit Committee and reviewed and approved by the Operations Review Committee.

Clarification

Changing the gains of the APRM amplifiers effectively changes the scram setpoints. The technical justification of this method is presented below to demonstrate that the Technical Specification requirement is satisfied exactly. To this end, a realistic set of conditions will be used as an example:

Core Power	20%
Recirculation Flow	40%
Core TPF	5

As per Specification 2.1.A.1, the scram setpoint is:

$$S = (0.65\ W + 55) \left[\frac{PF}{MTPF} \right]$$

$$\text{For 8X8 fuel } S = \left[(0.65)(40) + 55\right] \frac{3.06}{5} = 49.572$$

Therefore, the core power must increase a factor of 49.572/20 = 2.4786 to cause a scram. (Note that the power increase cannot be caused by a flow change or affect peaking factor to stay within the constraints of the Technical Specification.) This scram requirement is accomplished by changing the APRM gain so that the APRM indicates a factor of TPF/3.06 greater than actual core power.

For this example, after the APRM's have been adjusted, they will have a scale factor of 0.612.

$$APRM = (\text{Actual Core Power})_X \frac{TPF}{3.06}$$

$$= (20\%) \frac{(5)}{(3.06)} = 32.68$$

Scram occurs when APRM indicates:

$$APRM = .65\ (40) + 55 = 81\%$$

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Therefore, power must increase a factor of 81/32.68 = 2.4786 to cause a scram exactly satisfying Technical Specification requirements.

The actual value of the peaking factors for 8X8 fuel was removed and replaced with "the design value" to save future Technical Specification changes of the bases as these values change.

Schedule of Change

This change will be put into effect upon receipt of approval by the commission.

Boston Edison Company proposes that pursuant to 10CFR170 this is a Class III amendment.

Attachment B

Technical Specification Change concerning duration of Integrated Primary Containment Leak Rate Test (ILRT)

Proposed Change

Reference is made to Pilgrim Station Operating License No. DPR-35 Appendix A, specification 4.7.A.2a p. 153. The desired change would consist of replacing the existing paragraph:

> The test duration shall not be less than 24 hours for integrated leak rate measurements, but shall be extended to a sufficient period of time to verify, by measuring the quantity of air required to return to the starting point (or other methods of equivalent sensitivity), the validity and accuracy of the leak rate results.

With the following:

> The test duration shall not be less than 12 hours for integrated leak rate measurements, but shall be extended to a sufficient period of time to verify, by measuring the quantity of air required to return to the starting point (or other methods of equivalent sensitivity), the validity and accuracy of the leak rate results.

Reason for Change

The current specification requires 24 hours of outage critical path time be expended for leak rate testing, due to the fact that the original wording of this specification was made at a time when leak rate measurements and calculations were accomplished through manual methods, and a 24 hour test was considered mandatory for accurate results. However, at present, enhancements in computer data aquisition and data reduction have since invalidated the preceeding mandate, and a test duration of 12 hours is a more realistic approach for compliance to the technical specification.

Safety Considerations

Reducing the minimum duration of the IPCLRT from 24 hours to 12 hours does not present any hazards to the public health and safety. This change has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Clarification

1. The functional integrity of the primary containment can be demonstrated independent of the test period, as long as equilibrium has been obtained in the measured variables prior to commencement of the test. This is consistent with 10CFR50 Appendix J.

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2. A recalculation of the 1976 IPCLRT at Pilgrim has been conducted and statistical leak rate has met acceptance criteria on the basis of a 12 hour test. In fact, the 12 hour calculation results are more conservative than the 24 hour test. Calculated results of mass point leakage rates including a 95% upper confidence limit of .561%/day versus .412%/day for the 12 hour test and the 24 hour test, respectively, meet the acceptance criterion of .715%/day.

3. Data acquisition, reduction and calculational capability have been enhanced since the original IPCLRT was conducted in 1972. Data collection and alaysis for the original IPCLRT in 1972 was at a frequency of 3 data sets per hour, present capability as used in the 1976 test is 6 data sets per hour. The increased yield of data coupled with off-line computer analyses has given enhanced validity to statistical test results.

4. NRC regulations for IPCLRT, 10CFR50, Appendix J, specifies that statistical leak rate be adjusted to that which would be obtained for a 24 hour test, but does not require an actual 24 hour test period.

5. ANSI N45.4-1972 specifies that a test period of shorter than 24 hours may be used if "It can be demonstrated to the satisfaction of those responsible for the acceptance of the containment that a leak rate can be accurately determined during a shorter test period.

Schedule of Change

This change will be put into effect upon receipt of approval by the Commission. Boston Edison Company proposes that pursuant to 10CFR Part 170 this is a Class III amendment.

ATTACHMENT C

Technical Specification Change
RE: T. S. 3.6.I

Introduction

Pursuant to 10 CFR 50.59 Boston Edison hereby proposes the following changes to Appendix A of License No. DPR-35.

Proposed Change

Table 3.6.1 (page 137c) is a list of all safety related snubbers. This change consists of (a) deleting snubbers SS-2-10-17, SS-2-10-18, SS-3-3-1 and (b) changing the designation of snubber SS-6-10-1- to SS-6-10-10 and snubber SS-2-20-5 to SS-2-30-5, (c) changing the elevation of SS-2-20-1, SS-2-20-2, SS-2-20-3, SS-2-20-4 to 42' and (d) changing the prefix SS to S for snubbers located outside the drywell. Attachment A is the proposed Table 3.6.1 that reflects these changes.

Reason for Change

Snubbers SS-2-10-17 and SS-2-10-18 and SS-3-3-1 are not in the plant. SS-2-10-17 and SS-2-10-18 were removed when the reactor recirculation pump discharge valve 4" bypass lines were removed. This change was reported in the 1976 annual 10 CFR 50.59 report. SS-3-3-1 was removed when the control rod drive system return line was removed. This change was reported in the 1978 annual 10 CFR 50.59 report. Thus these changes update the Technical Specification to account for previously documented plant modification. The remaining changes are changes in snubber nomenclature and are made for administrative clarity.

Safety Considerations

These Technical Specification changes relfect plant modifications whose safety considerations have been previously documented. Hence the health and safety of the public is adequately protected when these proposed Technical Specifications become effective. This change has been reviewed and approved by the Operations Review Committee and the Nuclear Safety and Review Committee.

Schedule for Change

This change will be put into effect upon receipt of approval from the Commission.

Fee Determination

Boston Edison Company proposes that pursuant to 10 CFR 170 this is a Class II Amendment since it reflects changes that are adminsitrative in nature.

ATTACHMENT A

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-1-10-1	Main Steam Line	42'			X (Drywell)	
SS-1-10-2	Main Steam Line	42'			X (Drywell)	
SS-1-10-3	Main Steam Line	42'			X (Drywell)	
SS-1-10-4	Main Steam Line	42'			X (Drywell)	
SS-1-10-5	Main Steam Line	42'			X (Drywell)	
SS-1-10-6	Main Steam Line	42'			X (Drywell)	
SS-1-10-7	Main Steam Line	42'			X (Drywell)	
SS-1-10-8	Main Steam Line	42'			X (Drywell)	
SS-1-10-9	Main Steam Line	42'			X (Drywell)	
SS-1-10-10	Main Steam Line	42'			X (Drywell)	
SS-1-10-11	Main Steam Line	42'			X (Drywell)	
SS-1-10-12	Main Steam Line	42'			X (Drywell)	
SS-6-10-6	Feedwater Sys.	41'			X (Drywell)	
SS-6-10-7	Feedwater Sys.	41'			X (Drywell)	
SS-6-10-8	Feedwater Sys.	44'			X (Drywell)	
SS-6-10-9	Feedwater Sys.	41'			X (Drywell)	
SS-6-10-10	Feedwater Sys.	44'			X (Drywell)	
SS-10-30-1	RHR System	52'			X (Drywell)	
SS-10-20-2	RHR System	52'			X (Drywell)	
SS-10-20-3	RHR System	52'			X (Drywell)	
SS-10-20-4	RHR System	52'			X (Drywell)	
SS-10-30-5	RHR System	24'			X (Drywell)	
SS-10-30-6	RHR System	24'			X (Drywell)	
SS-10-20-7	RHR System	24'			X (Drywell)	
SS-10-20-8	RHR System	24'			X (Drywell)	
SS-10-3-9	RHR System	87'			X (Drywell)	
SS-10-3-10	RHR System	90'			X (Drywell)	
SS-2-20-1	Recir. System	42'	X		X (Drywell)	
SS-2-20-2	Recir. System	42'	X		X (Drywell)	
SS-2-20-3	Recir. System	42'	X		X (Drywell)	
SS-2-20-4	Recir. System	42'	X		X (Drywell)	
SS-2-30-5	Recir. System	15'	X		X (Drywell)	
SS-2-30-6	Recir. System	15'	X		X (Drywell)	
SS-2-30-7	Recir. System	15'	X		X (Drywell)	

ATTACHMENT A (cont.)

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-2-30-8	Recir. System	15'	X		X (Drywell)	
SS-2-30-9	Recir. System	11'	X		X (Drywell)	
SS-2-30-10	Recir. System	11'	X		X (Drywell)	
SS-2-30-11	Recir. System	27'	X		X (Drywell)	
SS-2-30-12	Recir. System	27'	X		X (Drywell)	
SS-2-30-13	Recir. System	27'	X		X (Drywell)	
SS-2-30-14	Recir. System	27'	X		X (Drywell)	
SS-2-30-15	Recir. System	27'	X		X (Drywell)	
SS-2-30-16	Recir. System	27'	X		X (Drywell)	
SS-2-20-19	Recir. System	16'	X		X (Drywell)	
SS-2-20-20	Recir. System	16'	X		X (Drywell)	
SS-2-20-21	Recir. System	19'	X		X (Drywell)	
SS-2-20-22	Recir. System	16'	X		X (Drywell)	
SS-2-50-23	Recir. System	17'	X		X (Drywell)	
SS-2-20-24	Recir. System	18'	X		X (Drywell)	
SS-2-20-25	Recir. System	16'	X		X (Drywell)	
SS-2-50-26	Recir. System	16'	X		X (Drywell)	

ATTACHMENT A (cont.)

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-6-10-1	Feedwater System	42'			X (Drywell)	
SS-6-10-2	Feedwater System	42'			X (Drywell)	
SS-6-10-3	Feedwater System	42'			X (Drywell)	
SS-6-10-4	Feedwater System	42'			X (Drywell)	
SS-6-10-5	Feedwater System	42'			X (Drywell)	
SS-13-3-1	RCIC	38'			X (Drywell)	
SS-13-3-2	RCIC	38'			X (Drywell)	
SS-14-3-1	Core Spray	65'			X (Drywell)	
SS-14-3-2	Core Spray	65'			X (Drywell)	
SS-14-3-3	Core Spray	65'			X (Drywell)	
SS-14-3-4	Core Spray	65'			X (Drywell)	
SS-23-10-1	H.P.C.I.	42'			X (Drywell)	
SS-23-10-2	H.P.C.I.	42'			X (Drywell)	
S-23-3-30	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-31	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-10-32	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-33	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-10-34	H.P.C.I.	-6'				X H.P.C.I. Quadrant
S-23-10-35	H.P.C.I.	-6'				X H.P.C.I. Quadrant
S-23-3-36	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-37	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-10-3-43	RHR	-3'06"				X RHR Pump Room
S-10-20-44	RHR	-3'06"				X RHR Pump Room
S-30-3-45	RBCCW	83'5"				X Reactor Building
S-10-10-46	RHR	6"				X Torus Compartment

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Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

ATTACHMENT D

ADMINISTRATIVE TECHNICAL SPECIFICATION CHANGE

Proposed Change

Reference is made to Pilgrim Nuclear Power Station - Unit #1 Technical Specification Appendix A, Section 6, Figure 6.2.2, titled "Pilgrim #1 Station Organization". The changes would consist of restructuring the chart (Exhibit A) to accommodate organizational changes at the Station. Also, Section 6.5.A.2 "ORC Composition" will be updated to reflect another ORC member.

Reason for Change

Recently, Boston Edison Company has initiated changes to improve the Health Physics Program at Pilgrim and as a result new positions have been created. The subject positions are titled as follows:

Chief Radiological Engineer

Senior ALARA Engineer
ALARA Engineer
ALARA Health Physics Technicians

Senior Waste Management Engineer

Prior to the creation of the Chief Radiological Engineer the Health Physics Program was under the responsibility of the Chief Technical Engineer. It is Boston Edison's intention, by means of these new positions, to increase Station management's attention toward the Health Physics Program.

Safety Considerations

Since the proposed changes affect only the Administrative Control Section of the Technical Specifications there are no safety-related changes involved. These changes have been reviewed by the Nuclear Safety Review and Audit Committee and have been reviewed and approved by the Operations Review Committee.

Schedule of Change

The described changes to the organization are presently in effect.

Boston Edison Company proposes that pursuant to 10CFR Part 170 this is a Class II Amendment.

Attachments: Exhibit A
Exhibit B

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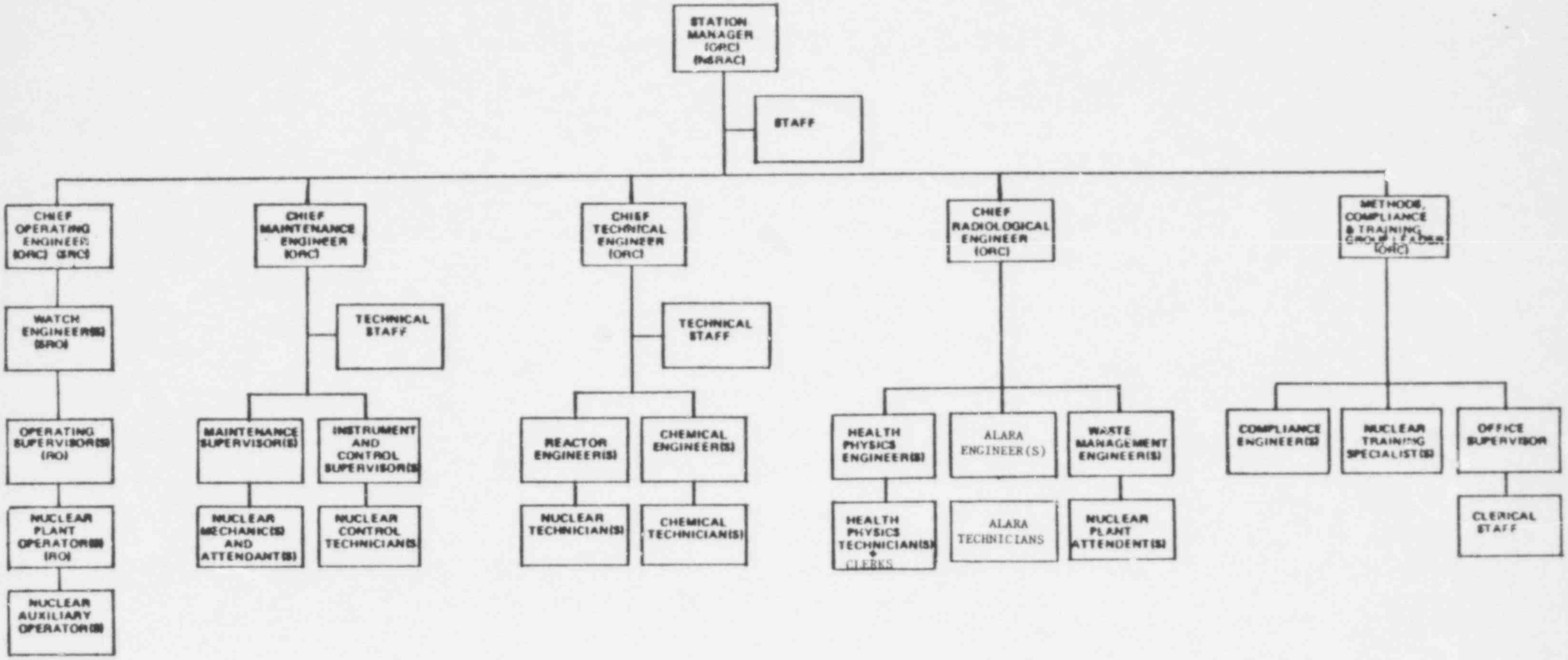
EXHIBIT A



CODE: NSRAC - MEMBER OF NSRAC
ORC - MEMBER OF ORC
RO - NRC REACTOR OPERATOR LICENSE
SRO - NRC SENIOR REACTOR OPERATOR LICENSE

PILGRIM I STATION ORGANIZATION
FIGURE 6.2.2

POOR ORIGINAL

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6.5 REVIEW AND AUDIT

A. OPERATIONS REVIEW COMMITTEE (ORC)

1. FUNCTION

The ORC shall function to advise the Pilgrim Station Manager on all matters related to nuclear safety.

2. COMPOSITION

The ORC shall be composed of the:

Chairman:	Station Manager
Member:	Methods, Compliance & Training Group Leader
Member:	Chief Operating Engineer
Member:	Chief Technical Engineer
Member:	Chief Maintenance Engineer
Member:	Chief Radiological Engineer

3. ALTERNATES

Alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in an ORC quorum at any one time.

4. MEETING FREQUENCY

The ORC shall meet at least once per calendar month and as convened by the ORC Chairman.

5. QUORUM

A quorum of the ORC shall consist of the Chairman and two members including alternates.

6. RESPONSIBILITIES

The ORC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto that affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Nuclear Operations Manager and to the NSRAC Chairman.

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ATTACHMENT E

Proposed Technical Specification Change
to Table 3.2.B, Auto Blowdown Timer

Proposed Change

Reference is made to Pilgrim Station Operating License No. DPR-35, Appendix A, Table 3.2.B, page 49. The desired change would consist of replacing the Trip Level Setting Values of the Auto Blowdown Timer from the current 120 seconds ± 5 sec. with the following values: ≥ 90, ≤ 120 seconds.

Reason for Change

A recent review of ECCS analysis input parameters has shown that the upper limit of the ADS blowdown timer (125 sec) permitted by the present Technical Specifications is higher than the value used in past and present ECCS analyses (120 sec). This disagreement results in a 3 to 5° nonconservatism in peak clad temperature in the LOCA analysis. Even though this temperature difference is well within the margins existing at PNPS, it would be desirable to have the Technical Specification and LOCA input entirely compatible. A change of input to the LOCA analysis would require a complete reanalysis which would not only be expensive but would unnecessarily reduce operating margins. The proposed Technical Specification change is therefore the preferable action to assure compatibility between the Technical Specification and LOCA analysis.

Safety Considerations

Reducing the response time of the ADS will increase the effectiveness of the ECCS by allowing LPCI to be operable earlier in the LOCA. The timer limit of 90 sec. is long enough to allow HPCI's to start (designed for 25 sec.) or to allow the operator to cancel the ADS signal if the main control room information indicates the signal is false or is not needed. Since the input to the LOCA analysis will remain unchanged, this Technical Specification change will not affect the accident analysis.

Schedule of Change

This change will be put into effect upon receipt of approval from the Commission.

Boston Edison Company proposes that pursuant to 10 CFR Part 170 this is a Class III amendment.

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ATTACHMENT F

Technical Specification Change Concerning In Sequence Criticals

Proposed Change

Reference is made to Pilgrim Station Operating License No. DPR-35 Appendix A, Specification 3.3.A.1 and corresponding bases p. 87. The desired change would consist of adding conditions to the bases (attached) to allow shutdown margin demonstration by a method other than that currently specified in our Technical Specifications.

Reason For Change

The two rod method (pulling the strongest worth rod and a diagonally adjacent rod to a specified position) produces a highly peaked flux distribution which maximizes the worth of the second rod. This high worth can lead to sudden unexpected criticals with fast periods. Conversely, the dispersed uniform withdrawal sequence is designed specifically to minimize the rod worths; thus the probability of a high reactivity insertion incident will be substantially reduced.

Safety Considerations

This Technical Specification change will not compromise the health and safety of the public since it offers a more conservative approach to demonstrating the shutdown margin, yet still retains the use of the two rod demonstration if such is required in the future. This change has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Schedule of Change

This change will be put into effect upon receipt of approval by the Commission.

Boston Edison Company proposes that pursuant to 10 CFR Part 170 this is a Class III Amendment.

BASES:

A. Reactivity Limitation

1. The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk at the time of the test, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in %Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter.

Bases Change
Dated 10/25/74

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. A new value of R must be determined for each full cycle.

The 0.25%Δk in the expression R + 0.25%Δk is provided as a finite, demonstrable, subcriticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25%Δk in reactivity, or by an insequence, xenon-free cold critical measurement to demonstrate at least R + 0.25% Δk in reactivity with the most reactive control rod fully withdrawn. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R + 0.25% Δk margin beyond this.

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ATTACHMENT G

Technical Specification Change To "IRM Channel Calibration"

Ref. A) USNRC Region I letter to Boston Edison Company dated September 21, 1978. Acknowledging response to Inspection 50-293/78-18

Proposed Change

Reference is made to Pilgrim Station Operating License No. DPR-35 Appendix A, Table 4.1.2. The changes would consist of adding further calibration and testing frequencies to the IRM High Flux Instrument Channel Section (see attachment).

Reason for Change

This change as requested by the Commission through an I&E Inspection and subsequent phone call (Ref. A) will serve to clearly specify the functional and calibration test requirements of the Nuclear Instrumentation Intermediate Range Monitoring (IRM) System.

Safety Considerations

This change does not present any hazard considerations not described or implied in the license application as amended. This change has been reviewed by the Nuclear Safety Review and Audit Committee and reviewed and approved by the Operations Review Committee.

Schedule of Change

This change will be put into effect upon receipt of approval by the Commission.

Boston Edison proposes that this request is exempt from any fee determinations since this amendment request was initiated by the Commission as clarification to the affected Technical Specifications.

TABLE 4.1.2
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Note (4)
		Full Calibration	once/operating cycle
APRM High Flux			
Output Signal	B	Heat Balance	Once every 3 Days
Flow Bias Signal	B	Internal Power and Flow Test	Each Refueling Outage
LPRM Signal	B	TIP System Traverse	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume	A	Note (6)	Note (6)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line Isolation Valve Closure	A	Note (6)	Note (6)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 Months
Turbine Stop Valve Closure	A	Note (6)	Note (6)
Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 Months

(ATTACHMENT)

ATTACHMENT H

Technical Specification Change
To Remove the Power Restriction Requirement
For Testing MSIV Closure Time

Reference: a) Boston Edison Company letter to NRC dated March 22, 1978 "Proposed Technical Specification Change to the Main Steam Line High Flow Setpoint."

b) NRC letter to Boston Edison Company dated September 19, 1978, "Amendment No. 34 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station, Unit No. 1"

Proposed Change

Reference is made to Pilgrim Nuclear Power Station Unit #1 Operating License No. DPR-35, Appendix A, Specification 4.7.D.1.b(2) p. 160. The desired change would consist of removing the current power restriction requirement of 50% reactor power when verifying closure time of Main Steam Isolation Valves.

Reason For Change

In our letter, Reference (a) we proposed a change to the Technical Specification for Pilgrim Nuclear Power Station, Unit No. 1, related to the trip level setting for the High Flow Main Steam Line instruments. The purpose of that modification was to allow testing of the MSIV's and Turbine Stop Control Valves at higher (full) power levels, thus eliminating the need to reduce power for testing purposes. This request was granted by the Commission per Reference (b). However, our letter Reference (a), failed to include all the Technical Specifications that would be affected by the subsequent amendment, and as such now submit this proposed change.

Safety Considerations

This proposed change was intended to be included in Amendment No. 34 and therefore the Safety Evaluations that accompanied Reference (a) and (b) satisfy all concerns related to safety.

This change has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Schedule of Change

This change will be put into effect upon receipt of approval of the Commission.

Fee Determination

Boston Edison Company proposes that pursuant to 10 CFR 170 this is a Class II Amendment.