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Docket No. 50-293

JAN 7 1975

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. 6 to Facility License No. DPR-35. This amendment includes Change No. 8 to the Technical Specifications which provides a general updating of Appendix A in connection with numerous correspondence and discussions of the past year.

This amendment revises the Technical Specifications (Change No. 8) (1) to recognize a design modification in the reactor protection system, (2) to delete the extended maintenance provisions, (3) to recognize certain performance characteristics of operating equipment, and (4) to correct errata and clarify certain ambiguous statements found through use of the document.

Copies of the Safety Evaluation and Federal Register Notice relating to this action are enclosed for your information.

Sincerely,

Original Signed by
Robert Purple

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

Enclosures:
1. Amendment No. 6
(w/Change No. 8)
2. Safety Evaluation
3. Federal Register Notice

bcc: J. R. Buchanan, ORNL
Thomas B. Abernathy, DTIE

cc: See next page

OFFICE ➤	L:LWR 2-1	L:OR	OGC	L:LWR 2-1	L:OR
SURNAME ➤	SBurwell/ms	KGoller	OGC	JStolz	DZiemann
DATE ➤	11/20/74 11/25/74	1/6/75	12/20/74	1/3/75	1/6/75

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-35

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

   A. The applications for amendment by Boston Edison Company (the licensee) identified and discussed in the Commission's related Safety Evaluation comply with the standards and requirements of the Atomic Energy Act of 1954, as amended, (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. 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

   C. 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:

   "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. 9"

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

FOR THE ATOMIC ENERGY COMMISSION

for Robert A. Purple

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

Attachment:
Change No. 8 to the Technical Specifications

Date of Issuance: **JAN 7 1975**

ATTACHMENT TO LICENSE AMENDMENT NO. 6

CHANGE NO. 8 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE DPR-35

DOCKET NO. 50-293

Instructions for updating your technical specifications, Appendix A, to the Facility Operating License DPR-35.

Remove Page Nos.	Insert Page Nos.
7	7
9	9
16, 17 and 18	16, 17, 18 and 18a
27, 28, 29, 30, 31 and 32	27, 28, 29, 30, 31 and 32
37, 38, and 39	37, 38, and 39
45	45
48	48
59	59
95, 96 and 97	95, 96 and 97
105	105
107, 108, 109, 110 and 111	107, 108, 109, 110, and 111
120	120
123	123
127	127 and 127a *
147 and 148	147, 147a and 148
202	202
208, 209 and 210	208, 209 and 210

* added by change dated June 24, 1974.

Remove Page Nos.	Insert Page Nos.
212 and 213	212 and 213
219	219
229, and 230	229, and 230

B. When the reactor pressure is less than 600 psig, or core coolant flow is less than 10% of design, the reactor thermal power shall not exceed 360 MWt (18 percent of design).

C. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone.

2. APRM (15%) - When the reactor mode switch is in the refuel or startup/hot standby position, the APRM scram shall be set at $\leq 15\%$ of design power. 8

3. IRM - The IRM scram shall be set at $\leq 120/125$ of full scale.

B. <u>APRM Rod Block</u>

The APRM Control Rod Block trip set point(s) shall be biased with flow as shown on Fig. 2.1.1 and shall be less than or equal to:

$$S \leq \left[.65W + 43\right]\left[\frac{3.09}{P.F.}\right]$$

The definitions used above for the APRM scram trip apply.

C. Reactor low water level scram setting shall be ≥ 9 in. on level instruments.

D. Turbine stop valve closure scram setting shall be ≤ 10 percent valve closure.

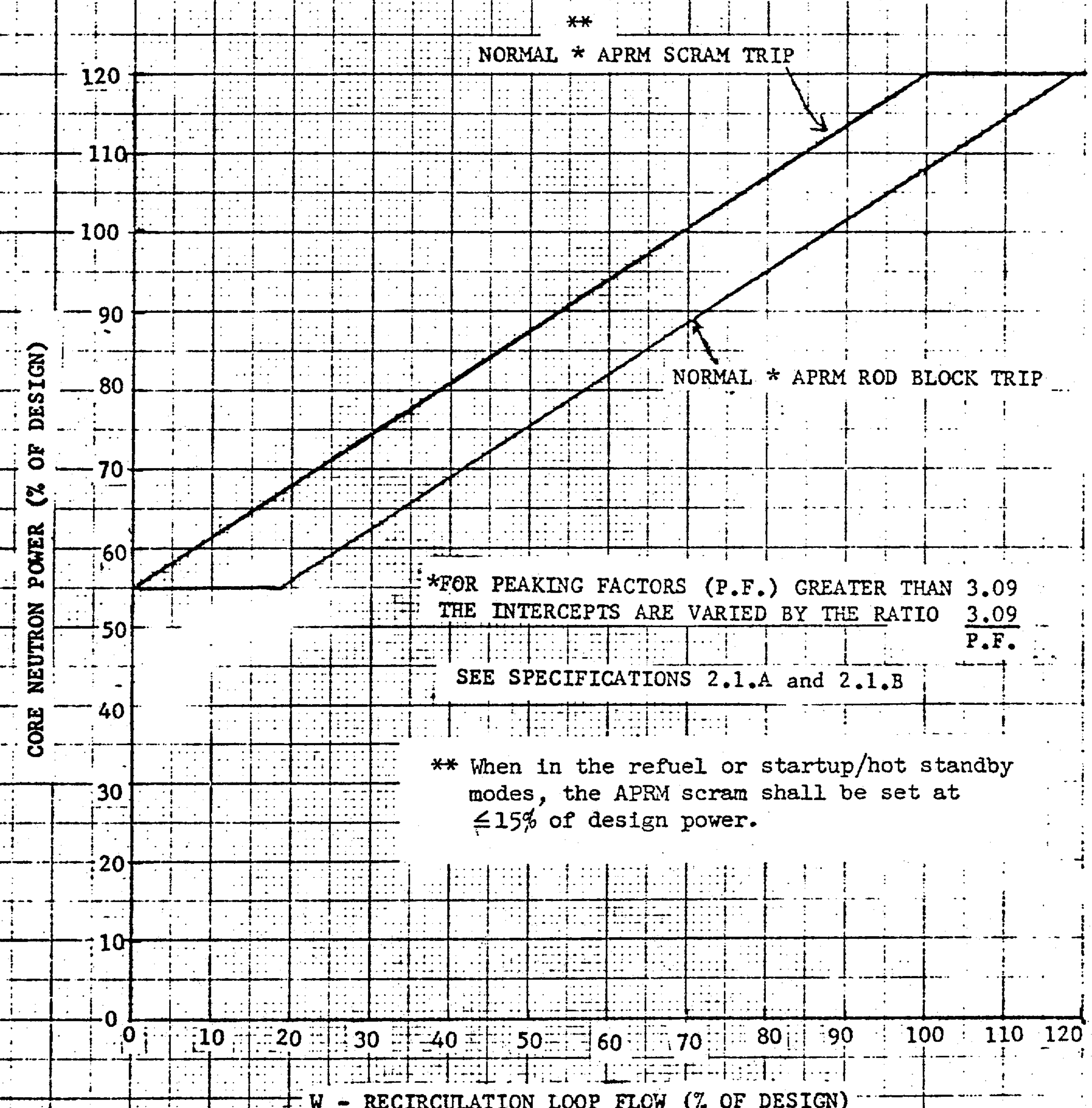
E. Turbine control valve fast closure setting shall be ≥ 150 psig control oil pressure at acceleration relay.

F. Condenser low vacuum scram setting shall be ≥ 23 in. Hg vacuum.

G. Main Steam Isolation Scram setting shall be ≤ 10 percent valve closure.

H. Main steam isolation on main steam line low pressure at inlet to turbine valves. Pressure setting shall be ≥ 880 psig.

I. Reactor low-low water level initiation of CSCS systems setting shall be at or above -49 in. indicated level.



8

8

APRM SCRAM AND ROD BLOCK TRIP LIMITING SAFETY SYSTEM SETTINGS

FIG. 2.1.1

2.1 BASES:

Steady-state operation without forced recirculation will not be permitted, except during startup testing.

In summary, the transients presented in the Safety Analysis Report were only analyzed up to the design flow control line and not above because:

1. The licensed maximum power level is 1998 MWt.

2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

3. The analysis model is demonstrated to be conservative.

4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual set points are discussed below:

A. Neutron Flux Scram

APRM 8

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of design power (1998 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients induced by disturbances, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses reported in Section 14 of the Final Safety Analysis Report demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin.

Because the flow biased scram plotted on Fig. 2.1.1 is based on recirculation loop flow and Fig. 1.1.1 for the Fuel Cladding Integrity Safety Limit is based on core flow, a one to one comparison of these curves cannot be made. Therefore Figure 2.1.3, which shows the flow biased scram as a function of core flow, has been included.

2.1 BASES:

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The fuel cladding integrity safety limit of Specification 1.1 was based on a total peaking factor of 3.09. A formula is included in Figure 1.1.1 to adjust the safety limit in the event the peaking factor exceeds this value. Likewise, the scram setting must be adjusted to ensure that the MCHFR does not become less than 1.0 in this degraded situation. The scram setting is adjusted in accordance with the formula in Specification 2.1.A above, when the combination of power and peak heat flux indicates a peaking factor greater than 3.09 (above curve PF = 3.09 on Figure 2.1.2). For operation in the startup mode while the
reactor is at low pressure, APRM scram is set at ≤ 15 percent of 8
design power. This provides an adequate thermal margin between the maximum power and the safety limit, which corresponds to 18 percent design power. The margin adequately accommodates anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures.

Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable case of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of design power, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five
percent of design power per minute, and the APRM system would be 8
more than adequate to assure a scram before power could exceed

2.1 BASES:

the safety limit. The 15% APRM scram remains active until
the mode switch is placed in the RUN position. 8

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

IRM

The IRM system consists of 8 chambers, 4 in each of the
reactor protection trip systems (A,B). Each trip system
uses 2 chambers per logic channel (A1, A2, B1, B2). The
IRM is a 5-decade instrument which covers the range of
power level between that covered by the SRM and the APRM.
The 5 decades are covered by the IRM by means of a range
switch and the 5 decades are broken down into 10 ranges,
each being one-half of a decade in size. The IRM scram
setting of 120 divisions is active in each range of the
IRM. For example, if the instrument were on range 1,
the scram setting would be 120 divisions for that range;
likewise, if the instrument were on range 5, the scram
would be 120 divisions on that range. Thus, as the IRM
is ranged up to accommodate the increase in power level,
the scram setting is also ranged up. The APRM 15% scram
prevents higher power operation without being in the RUN 8
mode. The IRM scram provides protection for changes which
occur both locally and over the entire core. The IRM,
because of the scram arrangement discussed above, thus
provides additional or backup protection to the APRM 15
percent scram in the STARTUP mode. The most significant
sources of reactivity change during the power increase
are due to control rod withdrawal. For in-sequence control
rod withdrawal, the rate of change of power is slow enough
due to the physical limitation of withdrawing control rods
that heat flux is in equilibrium with the neutron flux, and
an IRM scram would result in a reactor shutdown well before
any safety limit or the APRM 15 percent scram occurred. For
the case of a single control rod withdrawal error this
transient has been analyzed in paragraph 7.5.5.4 of the
FSAR. In order to ensure that the IRM provided adequate
protection against the single rod withdrawal error, a range
of rod withdrawal accidents was analyzed. This analysis
included starting the accident at various power levels. The
most severe case involves an initial condition in which the
reactor is just subcritical and the IRM system is not yet on
scale. This condition exists at quarter rod density.
Quarter rod density is illustrated in Figure 7.5.8 of the
FSAR. Additional conservatism was taken in this analysis by
assuming that the IRM channel closest to the withdrawn rod is
bypassed. The results of this analysis show that the reactor

2.1 BASES:

is scrammed and peak core power limited to one percent of rated power, thus maintaining heat flux within those values specified in the safety limit for this condition of plant operation. Based on the above analysis, the IRM provides 8
protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against MCHFR becoming less than unity. 8
This rod block set point, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable set point provides substantial margin from fuel damage, assuming a steady-state operation at the set point, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship; therefore, the worst case MCHFR during steady-state operation is at 108% of design power. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors greater than 3.09 exist. This assures a rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by changing the intercept point of the flow bias curve; thus, the entire curve will be shifted downward.

C. Reactor Water Low Level Scram

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram at this level adequately protects the fuel and the pressure barrier, because MCHFR remains well above 1.0 in all cases, and system pressure

Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum Number of Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action (1)
			Refuel (7)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	≤120/125 of full scale	X	X	(5)	A 8
3	Inoperative		X	X	(5)	A
	APRM					
2	High Flux	$\left[.65W + 55\right]\left[\frac{3.09}{PF}\right]$ (14) (15) (13)	(17)	(17)	X	A or B 8
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥2.5 Indicated on Scale	(11)	(11)	X(12)	A or B
2	High Flux (15%)	≤15% of Design Power	X	X	(16)	A or B 8
2	High Reactor Pressure	≤1085 psig	X(10)	X	X	A
2	High Drywell Pressure	≤2 psig	X(8)	X(8)	X	A
2	Reactor Low Water Level	≥9 In. Indicated Level	X	X	X	A
2	High Water Level in Scram Discharge Tank	≤39 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥23 In. Hg. Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	≤7X Normal Full Power Background	X	X	X	A or C
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	A or C
2	Turb. Cont. Valve Fast Closure	≥150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	Turbine Stop Valve Closure	≤10% Valve Closure	X(4)	X(4)	X(4)	A or D

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for both trip systems, the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place mode switch in the startup/hot standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 45% of design.

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

3. Permissible to bypass when reactor pressure is < 600 psig.

4. Permissible to bypass when turbine first stage pressure is less than 305 psig.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical, fuel is in the reactor vessel
and the reactor water temperature is less than 212°F, only the 8
following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM (15%) high flux scram 8

8. Not required to be operable when primary containment integrity is not required.

9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW (t).

Notes for Table 3.1.1 (Cont'd)

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 50% of the normal complement of LPRM's to an APRM.

14. W is the recirculation loop flow in percent of design. W is equal to 100 for core flow of 69 million pounds/hr. or greater. Trip level setting is in percent of design power (1998 MWt).

15. See Section 2.1.A.1.

16. The APRM (15%) high flux scram is bypassed when in the run mode.

17. The APRM flow biased high flux scram is bypassed when in the refuel or startup/hot standby modes. 8

TABLE 4.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
RPS Channel Test Switch (5)	A	Trip Channel and Alarm	Each Refueling Outage
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueli and Before Each Startup
Inoperative	C	Trip Channel and Alarm	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux	B	Trip Output Relays (4)	Once/Week (7) 8
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	Calibrate Flow Bias Signal	Once/Month (1)
High Flux (15%)	B	Trip Output Relays (4)	Once Per Week During Refueling and Before Each Startup 8
High Reactor Pressure	A	Trip Channel and Alarm	(1)
High Drywell Pressure	A	Trip Channel and Alarm	(1)
Reactor Low Water Level (6)	A	Trip Channel and Alarm	(1)
High Water Level in Scram Discharge Tank	A	Trip Channel and Alarm	Every 3 Months
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	(1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Control Valve Fast Closure	A	Trip Channel and Alarm	(1)
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 Months
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
Reactor Pressure Permissive	A	Trip Channel and Alarm	Every 3 Months

NOTES FOR TABLE 4.1.1

1. Initially once per month until exposure (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of PNPS.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required when the systems are not required to be operable or are tripped.

   If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. Test RPS channel after maintenance.

6. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the monthly functional test program.

7. This APRM testing will be performed once per week when in the run mode. If the reactor is out of the run mode for more than one week, the testing will be performed as soon as practicable after returning to the run mode. 8

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)	Minimim Frequency (2)	
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Note (4)	
APRM High Flux				
Output Signal	B	Heat Balance	Once every 3 Days	8
Flow Bias Signal	B	Internal Power and Flow Test	Each Refueling Outag	
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	

3.1 BASES (Cont'd)

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

8 The IRM system and APRM (15%) scram provide protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 39 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 39 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able

4.1 BASES (Cont'd)

purpose of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20 X 10^{-6} failure/hour. The bi-stable trip circuits are predicted to have unsafe failure rate of less than 2 x 10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each

3.1 BASES (Cont'd)

to perform its function adequately.

A source range monitor (SRM) sys-
tem is also provided to supply
additional neutron level informa-
tion during start-up but has no
scram functions. Ref. Section
7.5.4 FSAR. The APRM's cover the
"Refuel" and "Startup/Hot Standby"
modes with the APRM 15% scram, and
the power range with the flow bia-
8 sed rod block and scram. The IRM's
provide additional protection in
the "Refuel" and "Startup/Hot
Standby" modes. Thus, the IRM and
APRM 15% scram are required in the
"Refuel" and "Startup/Hot Standby"
modes. In the power range the
APRM system provides the required
protection. Ref. Section 7.5.7
FSAR. Thus, the IRM system is not
required in the "Run" mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for Startup/Hot Standby and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.1 operable in the Refuel mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system.

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. Below 305 psig turbine first stage pressure (45% of rated), the scram

4.1 BASES (Cont'd)

refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating

3.1 BASES (Cont'd)

signal due to turbine stop valve closure is bypassed because flux and pressure scram are adequate to protect the reactor.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale assures that there is proper overlap in the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation.

The provision of an APRM scram at ≤15% design power in the "Refuel and "Startup/Hot Standby" modes and the backup IRM scram at ≤120/125 of full scale assures
8 that there is proper overlap in
the neutron monitoring systems and, thus, that adequate coverage is provided for all ranges of reactor operation.

4.1 BASES (Cont'd)

devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter Table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

PNPS

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum # of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Action (2)
2 (7)	Reactor Low Water Level	≥9" indicated level (3)	A and D
1	Reactor High Pressure	≤110 psig	D
2	Reactor Low-Low Water Level	at or above -49 in. indicated level (4)	A
2	Reactor High Water Level	≤48" indicated level (5)	B
2 (7)	High Drywell Pressure	≤2 psig	A
2	High Radiation Main Steam Line Tunnel	≤7 times normal rated full power background	B
2	Low Pressure Main Steam Line	≥880 psig (8)	B
2(6)	High Flow Main Steam Line	≤120% of rated steam flow	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤170°F	B
2	Turbine Basement Exhaust Duct High Temperature	≤150°F	B
1	Reactor Cleanup System High Flow	≤300% of rated flow	C
2	Reactor Cleanup System High Temperature	≤150°F	C

PNPS

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks	
2	High Drywell Pressure	≤2 psig	1. Initiates Core Spray; LPCI; HPCI. 2. In conjunction with Low-Low Reactor Water Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS). 3. Initiates starting of Diesel Generators	
1	Reactor Low Pressure	400 psig ± 25	Permissive for Opening Core Spray and LPCI Admission valves.	8
1	Reactor Low Pressure	≤110 psig	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.	8
1	Reactor Low Pressure	400 psig ± 25	In conjunction with Low-Low Reactor Water Level initiates Core Spray and LPCI.	8
2	Reactor Low Pressure	900 psig ± 25	Prevents actuation of LPCI break detection circuit.	8

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours. 8

(4) These surveillance instruments are considered to be redundant to each other.

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification:

A. Normal System Availability

1. During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of of the following tests:

1. At least once per month each pump loop shall be functionally tested by recirculating demineralized water to the test tank.

2. At least once during each operating cycle:

   a. Check that the system relief valves trip full open at pressures less than 1800 psig, and reseat on a falling pressure greater than 1275 psig. 8

   b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39

3.4 STANDBY LIQUID CONTROL SYSTEM

B. Operation with Inoperable Components:

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an
8 operable condition within seven days.

4.4 STANDBY LIQUID CONTROL SYSTEM

gpm against a system head of 1275 psig shall be verified.

c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

   This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability. The replacement charges to be installed will be selected from the same manufactured batch as the tested charge.

d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

B. Surveillance with Inoperable Components:

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

3.4 STANDBY LIQUID CONTROL SYSTEM

8

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume - concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2. This includes the piping between the standby liquid control tank and the suction inlet to the pumps.

D. If specification 3.4.A through C cannot be met, the reactor shall be placed in a Cold Shutdown Condition with all operable control rods fully inserted within 24 hours.

4.4 STANDBY LIQUID CONTROL SYSTEM

C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Temperature: Check at least once per day.

3. Concentration: Check at least once per month. Also check concentration anytime water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.

LIMITING CONDITION FOR OPERATION

3.5.A Core Spray and LPCI Subsystems (Cont'd)

4. From and after the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days provided that during such thirty days the remaining active components of the LPCI Subsystem, the containment cooling subsystem, and all active components of both core spray subsystems and the diesel generators are operable.

5. From and after the date that the LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystem (including 2 LPCI pumps) and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. If the requirements of 3.5.A cannot be met,
8 an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENT

4.5.A Core Spray and LPCI Subsystems (Cont'd)

4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operable, the remaining active components of the LPCI Subsystem, the containment cooling subsystem, both core spray systems and the diesel generators shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

5. When it is determined that the LPCI subsystem is inoperable, both core spray subsystems, the containment cooling subsystem and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and daily thereafter.

3.5.B Containment Cooling Subsystem (Cont'd)

4. From and after the date that one containment cooling subsystem loop is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem loop is sooner made operable, provided that all active components of the other containment cooling subsystem loop, including its associated diesel generator, are operable.

5. If the requirements of 3.5.B can-
8 not be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

4.5.B Containment Cooling Subsystem (Cont'd)

4. When one containment cooling subsystem loop becomes inoperable, the operable subsystem loop and its associated diesel generator shall be demonstrated to be operable immediately and the operable containment cooling subsystem loop daily thereafter.

C. HPCI Subsystem

1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 104 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.C.2 and 3.5.C.3 below.

C. HPCI Subsystem

1. HPCI Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	Flow Rate at 1000 psig	Once/3 months
e.	Flow Rate at 150 psig	Once/operating cycle

3.5.C HPCI Subsystem (Cont'd)

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.

8 3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 104 psig within 24 hours.

4.5.C HPCI Subsystem (Cont'd)

The HPCI pump shall deliver at least 4250 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

2. When it is determined that the HPCI Subsystem is inoperable the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuation logic shall be demonstrated to be operable immediately. The RCIC system and ADS subsystem logic shall be demonstrated to be operable daily thereafter.

3.5.D Reactor Core Isolation Cooling (RCIC) Subsystem

1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor pressure is greater than 104 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 below.

4.5.D Reactor Core Isolation Cooling (RCIC) Subsystem

1. RCIC Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month

3.5.D Reactor Core Isolation Cooling (RCIC) Subsystem (Cont'd)

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.

8 3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 104 psig within 24 hours.

4.5.D Reactor Core Isolation Cooling (RCIC) Subsystem (Cont'd)

d. Flow Rate at 1000 psig — Once/3 months

e. Flow Rate at 150 psig — Once/operating cycle

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig

2. When it is determined that the RCIC subsystem is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

3.5.E Automatic Depressurization System (ADS)

1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.

4.5.E Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

   b. With the reactor at pressure, each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

3.5.E Automatic Depressurization System (ADS) (Cont'd)

2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days unless such valve is sooner made operable, provided that during such thirty days the HPCI subsystem is operable.

3. If the requirements of 3.5.E cannot be met, an orderly shutdown
8 shall be initiated and the reactor pressure shall be reduced to at least 104 psig within 24 hours.

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generator shall be operable. If this requirement cannot be met, an order-
8 ly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

4.5.E Automatic Depressurization System (ADS) (Cont'd)

2. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the other ADS valves and the HPCI subsystem shall be demonstrated to be operable immediately and at least weekly thereafter until the valve is repaired.

4.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter until the inoperable diesel is repaired.

3.5.F <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u> (Cont'd)

3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.

4. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days.

3.5.G

8

3.5.H <u>Maintenance of Filled Discharge Pipe</u>

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5.H <u>Maintenance of Filled Discharge Pipe</u>

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Every month prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point and water flow observed.

BASES:

3.5.G

8

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period except when the vessel temperatures are above 450°F. The shell flange to shell temperature differential shall not exceed 145°F.

2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6.1. The reactor vessel shall not be pressurized above 250 psig unless the reactor vessel temperatures are
8 equal to or greater than 180°F when fuel is in the reactor vessel.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 5°F.

   a. Reactor vessel shell adjacent to shell flange

   b. Reactor vessel shell flange

   c. Recirculation loops A and B

2. Reactor vessel shell temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

   Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to the requirements of ASTM E 185-66. Selected

3.6.D Safety and Relief Valves (Cont'd)

pressure shall be below 104 psig within 24 hours.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

F. Jet Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than or equal to 80%.

8

G. Structural Integrity

1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1974

6/24/74

4.6.D Safety and Relief Valves (Cont'd)

the bellows monitoring system shall be demonstrated at least once every three months.

E. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from established jet pump ΔP characteristics by more than 10%.

F. Jet Pump Flow Mismatch

Recirculation pump speeds shall be checked and logged at least once per day.

G. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

3.6.G Structural Integrity (Con't)

Edition (ASME Code, Section XI). Components of the primary system boundary whose in-service examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this code are acceptable for continued service. Plant operation with components which have in-service examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to AEC approval.

a. Components whose in-service examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

   (i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the AEC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications," of ASME Code, Section XI.

   (ii) Prior to the resumption of service, the AEC shall review the analysis and evaluation and either approve resumption of plant operation with the affected components or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph a, above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive in-service inspection. An analysis and evaluation shall be submitted to the AEC following each in-service inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications," of ASME Code, Section XI, and shall reference prior analyses submitted to the AEC to the extent applicable. Prior to resumption of service following each in-service inspection, the AEC shall review the analysis and evaluation and either approve resumption of plant operation with the affected components or require that the component be repaired or replaced.

c. Repair or replacement of components, including re-examinations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

6/24/74

BASES:

3.6.E & 4.6.E

Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism,
nozzle assembly and/or riser, would increase the cross-
sectional flow area for blowdown following the design
basis double-ended recirculation line break. Therefore, if 8
a failure occurred, repairs must be made.

A nozzle riser failure could cause the coincident failure of a
jet pump body; however, because of the lack of any substan-
tial stress in the jet pump body, the converse is not possible.
Therefore, failure of a jet pump body cannot occur without the
failure of the nozzle riser.

The following factors form the basis for the surveillance
requirements:

- A break in a jet pump decreases the flow resistance
characteristic of the external piping loop causing
the recirculation pump to operate at a higher flow
condition when compared to previous operation. 8

- The change in flow rate of the failed jet pump pro-
duces a change in the indicated flow rate of that
pump relative to the other jet pumps in that loop.
Comparison of the data with a normal relationship or
pattern provides the indication necessary to detect
a failed jet pump.

- The jet pump flow deviation pattern derived from the
diffuser to lower plenum differential pressure read-
ings will also be used to evaluate jet pump operability.

The detection technique is as follows. With the two re-
circulation pumps balanced in speed to within $\pm 5\%$, the
flow rates in both recirculation loops will be verified
by Control Room monitoring instruments. If the two flow
rate values do not differ by more than 15%, riser and noz- 8
zle assembly integrity has been verified. If they do differ
by 15% or more after correction for the difference in
pump speeds, the diffuser to lower plenum differential
pressure of all jet pumps will be compared to established
jet pump ΔP characteristics. In the event of a failed jet

BASES:

3.6.E & 4.6.E

Jet Pumps

pump nozzle (or riser), the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow. If the jet pump ΔP indications are within 10% of established jet pump ΔP characteristics, jet pump nozzle and riser integrity have been established. If
the indicated jet pump ΔP varies from the established jet pump 8
characteristics by more than 10%, indicated core flow will be compared to the core flow derived from loop flow measurements. If the difference between measured and derived core flow rate is 10% or more, a failed jet pump nozzle (or riser) is indicated and the plant shall be shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate.

8

BASES:

3.6.F and 4.6.F

Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the Pilgrim Nuclear Power Station FSAR. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. At or below 8
80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

3.10 Core Alterations

Applicability

Applies to the fuel handling and core reactivity limitations during
8 refueling and core alterations.

Objective

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification

A. Refueling Interlocks

8 During core alterations when fuel is in the vessel the reactor mode switch shall be locked in the "Refuel" position and the refuelling interlocks shall be operable.

B. Core Monitoring

8 During core alterations when fuel is in the vessel two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. **The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)**

4.10 Core Alterations

Applicability

Applies to the periodic testing of those interlocks and instrumentation used during refueling and core alterations.

Objective

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification

A. Refueling Interlocks

Prior to any fuel handling with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization, Review and Audit

A. The Pilgrim Division Head has the overall full-time responsibility for safe operation of the facility. During periods when the Division Head is unavailable, he may delegate this responsibility to one of the ORC Members. In the event of unexpected absence or un- 8
availability of the Division Head, the Process Engineer will assume the responsibility for safe operation of the facility.

B. The portion of the corporation management which relates to the operation of this plant is shown in Figure 6.1.1.

C. The functional organization for operation of the station shall be as shown in Figure 6.1.2. The normal shift complement is as stated in the notes for Figure 6.1.2.

The minimum shift complement during the reactor operating and cold shutdown modes shall be as stated in note 4, for Figure 6.1.2.

D. The qualifications with regard to educational and experience backgrounds of key supervisory and professional personnel at the time of initial core loading or appointment to the active position shall meet the requirements as described in the American National Standards Institute N18.1 - 1971, "Selection and Training of Personnel for Nuclear Power Plants". The key supervisory and professional personnel are indicated by an asterisk (*) on Figure 6.1.2.

E. Organizational units for the review and audit of facility operations shall be constituted and have the responsibilities and authorities outlined below:

1. Nuclear Safety Review and Audit Committee (NSRAC)

a. Membership

The NSRAC shall consist of eight members. As a group, they will collectively provide experience in reactor operation, reactor engineering, chemistry and radiochemistry, metallurgy and radiation damage, instrumentation and control systems, radiological safety, mechanical and electrical systems and any other appropriate fields required by the unique characteristics of the facility. The Chairman and Vice Chairman and other members shall be appointed by the President of the Boston Edison Company or such other person as he shall designate. No more than two members shall be selected from the normal operating organization. The NSRAC will obtain advice and counsel of scientific or technical personnel employed by the Company or other organization whenever the Committee considers it necessary to obtain further scientific or technical assistance in carrying out its responsibilities.

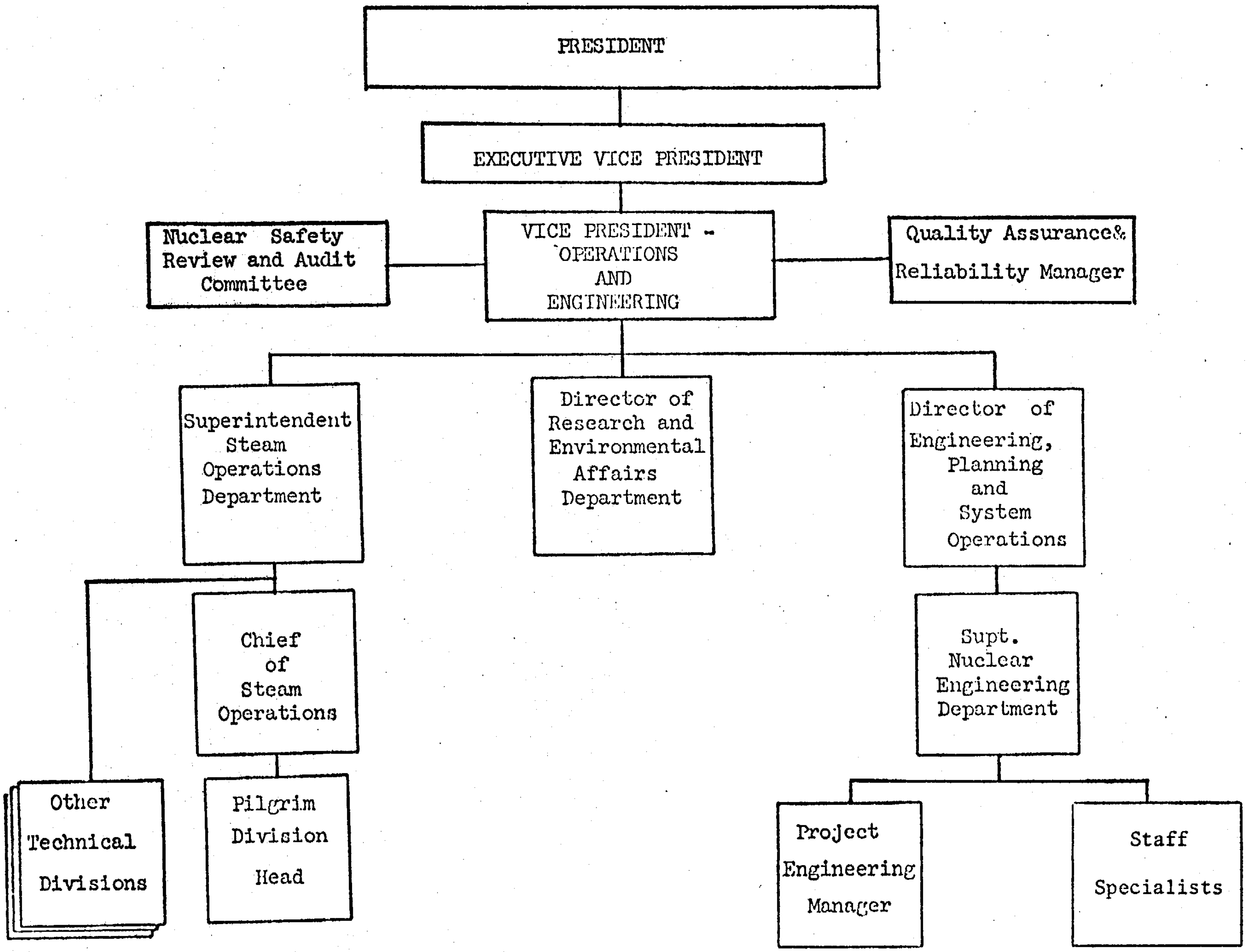
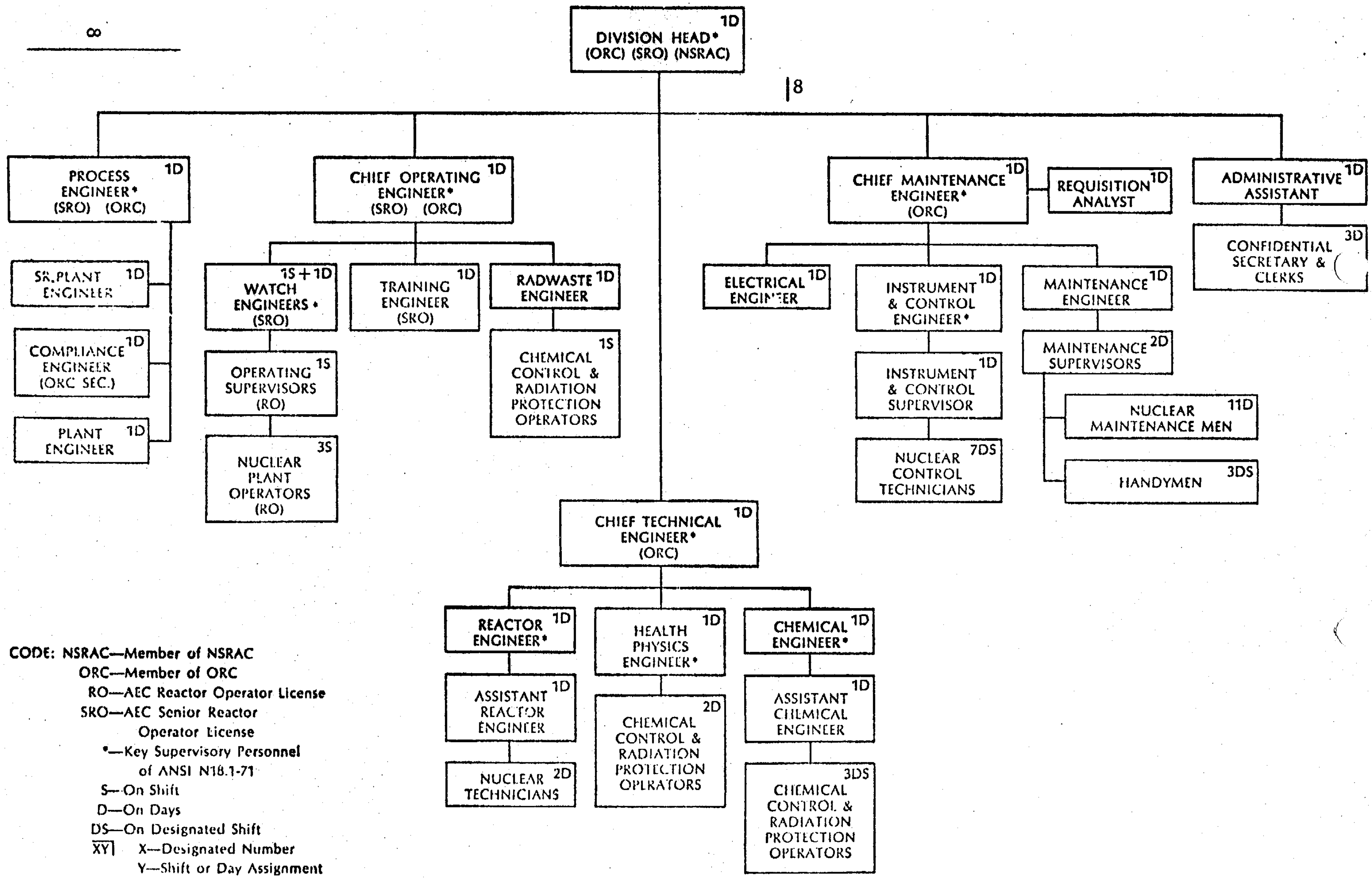


FIGURE 6.1.1 BOSTON EDISON COMPANY LINE ORGANIZATION



PILGRIM DIVISION ORGANIZATION

FIGURE 6.1.2

b. Meeting Frequency

The NSRAC shall meet on call by the Chairman and at least semi-annually.

c. Quorum

Chairman or Vice Chairman plus three members.

d. Responsibilities

(1) Review proposed changes to the station operating license including Technical Specifications, emergency plan, unreviewed safety questions and Final Safety Analysis Report.

(2) Review proposed changes or modifications to station procedures, systems or equipment referred to NSRAC by the Operations Review Committee (ORC), the Pilgrim Division Head or by members of NSRAC.

(3) Review meeting minutes of the ORC to determine if matters considered by that committee involve unreviewed safety questions or change in the Technical Specifications incorporated in the license.

(4) Review reports regarding nuclear safety considerations
that are originated by either the Steam Operations 8
Department or the Nuclear Engineering Department.

(5) Arrange for, and evaluate the results of, semiannual audits of nuclear safety related aspects of station operation to verify that operation complies with the nuclear safety related terms, conditions and intent of any license, permit and other applicable regulations.

(6) Perform special reviews and investigations as requested 8
by the Vice President-Operations and Engineering.

(7) Review all abnormal occurrences which require reporting to Federal and State agencies.

(8) Provide advice and counsel, as requested by station management, on problems of nuclear safety.

(9) Review ORC and station handling of reported or suspected violations of AEC regulations or orders, license requirements or technical specifications and make recommendations as needed in regard to such violations.

e. Authority

The NSRAC shall be advisory to the Vice President-Operations and Engineering. The NSRAC shall have authority to obtain access to the station operating record files and to station personnel to perform NSRAC functions.

f. Records

(1) Minutes shall be recorded for all meetings of
NSRAC. Copies of the minutes shall be forwarded
to the Vice President-Operations and Engineering,
Superintendent of Steam Operations Department,
the Pilgrim Division Head, Superintendent of Nuclear 8
Engineering Department, Quality Assurance and
Reliability Manager, committee members and others
the Chairman or Vice-Chairman may designate.

(2) Reports of all reviews and audits including
evaluation of results by the Committee shall be
made in writing to the Vice President-Operations
and Engineering, with copies to the Superintendent
of Steam Operations Department, Pilgrim Division 8
Head, Superintendent of Nuclear Engineering Department, and Quality Assurance & Reliability Manager.

(3) The findings of all reviews by the Committee of
ORC and station handling of violations of license
requirements, AEC regulations or technical specifications and any recommendations made in regard to
such violations shall be reported in writing to
the Vice President-Operations and Engineering,
with copies to the Superintendent of Steam Oper- 8
ations Department, Pilgrim Division Head, Superintendent of Nuclear Engineering Department, and
Quality Assurance & Reliability Manager.

g. Procedures

A written charter for Committee operation shall be prepared and maintained. This charter shall include Committee responsibilities, method of submission of presentations to the Committee, use of subcommittees, committee member approval of Committee actions, distribution of meeting minutes, Committee membership, quorum requirements, meeting agenda and scheduling of meetings.

2. Operations Review Committee (ORC)

a. Membership

The ORC shall consist of the Pilgrim Division Head,
Process Engineer, Chief Operating Engineer, Chief
Technical Engineer and the Chief Maintenance
Engineer. The Division Head shall be the Chairman 8
and the Process Engineer shall be the Vice Chairman.

TABLE 6.1.3
PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR 14.4(b)(4)
Facepiece, full 7/	NP	100	21B 30 CFR 14.4(b)(5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline Respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c)(2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c)(2) Type C(i)
Facepiece, full 7/	D	100	19B 30 CFR 12.2(c)(2) Type C(ii) 8
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c)(2) Type C(iii)
Hood	CF	5/	6/
Suit	CF	5/	6/
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 7/	D	100	13E 30 CFR 11.4(b)(2)(i) 8
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b)(2)(ii)
Facepiece, full	R	1,000	13E 30 CFR 11.4(b)(1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR 12.2(e) or applicable schedules as listed above

1/, 2/, 3/, 4/, 5/, 6/, 7/, (These notes are on the following pages)

NOTES: 1. Each integrated leak rate test of the primary containment shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report as described in the AEC Guide on Containment Testing dated January 16, 1966, shall include data, analysis and interpretations of the results which demonstrate compliance in meeting the specified leak rate limits.

2. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

3. The report shall be submitted within 90 days after completion of each test. Test periods shall be based on the commercial service date as the starting point.

3. A report shall be submitted to the Commission along with the routine operating report delineated in 6.6.D.1 of these specifications, specifying total quantities of radioactive material released to unrestricted areas in liquid and gaseous effluents during the previous six months and such other information on releases as may be required to estimate exposures to the public resulting from effluent releases. If quantities of radioactive material released during the reporting period are unusual for normal reactor operations, including expected operational occurrences, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the Licensee or others, the Commission may from time to time require the Licensee to take such action as the Commission deems appropriate.

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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. 6 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 Unit 1, located in Plymouth, Massachusetts. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications (1) to recognize a design modification in the reactor protection system, (2) to delete the extended maintenance provisions, (3) to recognize certain performance characteristics of operating equipment, and (4) to correct errata and clarify certain ambiguous statements found through use of the document.

The applications for the amendment, which are identified in the Commission's related Safety Evaluation, comply 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 identified and discussed in the Commission's related Safety Evaluation, (2) Amendment No. 6 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, North Street, 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 7th day of JAN 1975

FOR THE ATOMIC ENERGY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch No. 2
Directorate of Licensing

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING
SUPPORTING AMENDMENT NO. 6 TO LICENSE NO. DPR-35
(CHANGE NO. 8 TO THE TECHNICAL SPECIFICATIONS)
BOSTON EDISON COMPANY
PILGRIM NUCLEAR POWER STATION, UNIT 1
DOCKET NO. 50-293

INTRODUCTION

On September 15, 1972, the Atomic Energy Commission issued Amendment No. 1 to Facility Operating License No. DPR-35 to the Boston Edison Company (BECO) for the Pilgrim Nuclear Power Station. This license amendment permits operation at steady state power levels not to exceed 1998 megawatts thermal, in accordance with the Technical Specifications contained in Appendix A attached to the license. Since Amendment No. 1 to DPR-35, the Technical Specifications have been amended as follows:

1. to incorporate additional requirements resultant from the Regulatory staff review of the fuel densification matter,
2. to extend the closure time on a containment isolation valve,
3. to condition several requirements to recognize conditions not anticipated during the core inspection and refueling operation,
4. to permit the loading of 8 x 8 fuel assemblies,
5. to reflect modified assumptions used in the Station transient analyses,
6. to authorize operation of the station using 8 x 8 fuel assemblies,
7. to add surveillance requirements on the drywell vacuum breakers.
8. to update the requirements for inservice inspection of the primary system boundary, and
9. to revise the bases for reactivity limitations.

Other changes to the Technical Specifications, Appendix A, have been proposed and discussed between representatives of the BECO and the Regulatory staff over the past year. Some of these changes were initially proposed at the meeting held May 24, 1973, and have been the subject of correspondence since that date. Other changes are of much more recent origin. The need for and causes of these changes are:

a. BECO has made modifications to the station to incorporate improvements in more recent BWR designs. In no case have the modifications resulted in a decrease in the safety features of the station.

b. We have determined that modification of the Technical Specifications in certain instances is necessary to make our intentions clear.

c. The Pilgrim Station staff has accumulated experience about the operating characteristics of the instrumentation, controls and process equipment installed in the station.

d. Reorganizations within BECO have resulted in the transfer of certain functions to other organizational divisions.

e. The staff of the Pilgrim Station and the Commission have found through use that a number of statements, phrases, and words need changes to correct errata, to make their meaning and intent clear to all parties, and to more accurately reflect operation conditions and inspection procedures.

EVALUATION

Our evaluation of the proposed changes to the Technical Specifications for the Pilgrim Nuclear Power Station is as follows:

A. APRM 15% Power Scram

During the course of the meeting held on May 24, 1973, representatives of BECO proposed a series of revisions to the Technical Specifications changing the low power scram trip at 15% rated power from the IRM system to the APRM system. Consequent to this discussion, BECO submitted a description of these changes in the Reactor Protective System (RPS) by a letter dated August 7, 1973. Our review of this information concluded that this alteration in the RPS changes it to the system design installed at other recently licensed BWRs; e.g., Browns Ferry, Peach Bottom, Cooper. Our review further concluded that the modification would be equally acceptable installed in Pilgrim because the modification provides the same function on Pilgrim as on these other comparable plants. And, the modification provides for the replacement of one trip circuit with an equivalent trip circuit. We also compared the Technical Specification changes proposed for the Pilgrim Station and found them consistent with the recently issued Technical Specifications on other BWRs and have concluded that these changes do not result in any reduction in the protection provided by the RPS and are therefore acceptable. The changes in the Technical Specifications have been made directly on the respective pages as marked in the attachment to this license amendment. This modification in the RPS resulted in changes to pages 7, 9, 16, 17, 18, 18a, 27, 28, 29, 30, 31, 37, 38, and 39 of the Technical Specifications.

B. Deletion of Extended Maintenance Provisions

In accordance with Regulatory policy for existing Technical Specifications, the Limiting Condition for Operation 3.5.G, Extended Maintenance, and all references to it are hereby deleted. This change eliminates the permission to continue station operation for periods greater than specified elsewhere throughout the Technical Specification when minimal safety equipment is not operable. No safety limits or trip settings are affected. this deletion resulted in changes to pages 59, 96, 97, 105, 107, 108, 109, 110, 111, 120, 229, and 230 as marked in the attachment to the license amendment.

C. Surveillance Requirements for Jet Pump Operability

Surveillance Requirement 4.6.E requires a sequence of daily checks to verify that all jet pumps are operable. Three checks are used to determine whether a jet pump might have experienced failure of the nozzle assembly hold down mechanism, the nozzle assembly, or the riser. Such a failure would be indicated by all three checks outside their respective allowable limits. One of the checks is dependent upon a comparison of the pressure differential between the diffuser and lower plenum developed by each of the jet pumps. A significant change in the pressure differential from values previously characteristic of a pump would be indicative of a structural change in the pump. In the course of developing these surveillance requirements for the Pilgrim Station, the tacit assumption was made that all 20 jet pumps would develop substantially the same pressure differential. Specification 4.6.E.3 was consequently based upon the detection of a variant from the mean pressure differential exhibited by all of the jet pumps. In the case of the Pilgrim Station, operating experience showed that this pressure differential varied significantly between the individual jet pumps. This variation is described in a letter from the Boston Edison Company dated October 9, 1973. Members of the Pilgrim Station staff have advised us that these differences in the pressure differential are attributed to anomalies in the feedwater distribution to the jet pumps and is not indicative of a jet pump failure. This letter also enclosed a detailed test procedure for checking the operability of these jet pumps against a record of their respective operating characteristics as established by testing. We have therefore modified the Specification 4.6.E.3 to base the detection of a jet pump failure upon a variation from its established pressure differential characteristics.

This change provides for the observed variation in normal operational characteristics of these jet pumps, and more accurately states the surveillance requirement for the detection of a failure in the jet pump. We have also revised the Bases for Specifications 3.6.E and 4.6.E to more accurately track the conduct of the three checks used to identify a jet pump failure. Our review of these matters concluded that these changes do not result in any reduction in the protection provided by these surveillance requirements, and are acceptable. These revisions resulted in changes to pages 127, 147, and 147a of the Technical Specifications as marked in the attachment to the license amendment.

D. Surveillance Requirements for the Relief Valves on the Standby Liquid Control System

At the meeting held on May 24, 1973, representatives of BECO proposed a revision to the Technical Specifications to clarify the surveillance requirements for the relief valves on the Standby Liquid Control System. The Technical Specification 4.4.A.2.a previously required the licensee to "check that the setting of the system relief valves is 1425 ± 50 psig". That requirement was not specific as to whether the relief valves should only open or both open and reseat within the ± 50 psig tolerance band. Operating experience further showed that it was difficult to maintain the valve settings within these stringent limits as described in BECO's letter dated August 7, 1973. It should be noted that the ± 50 psig tolerance was arbitrarily selected at the time the Technical Specifications were drafted and have no safety-related significance. The change provided by this revision to the Technical Specifications separates the two actions (opening and reseating) It sets the full open pressure at 1800 psig, the design pressure for the SBLC system, thereby providing full protection against overpressurization of the SBLC system. The lower reseating limit provides assurance that the SBLC system will reseal and be capable of injecting boron solution into the primary coolant system at any pressure condition that might be experienced by the primary coolant system in which the SBLC system may be required. The analyses of abnormal operational transients given in FSAR Section 14 show that a high primary coolant system pressure is of such short duration (in the order of seconds) that the SLCS is neither used nor effective in the course of the transient. The analyses given in FSAR Section 14 and the analysis for safety valve sizing given in BECO letters of October 16, 1973, February 15, 1974, and March 4, 1974, all show that the peak primary coolant system pressure does not exceed 1275 psig, the lower reseating setpoint. We conclude that this change more accurately describes the design requirements for the SBLC system and does not reduce the safety performance of this system. Therefore, we conclude that the change is acceptable. The change is marked on page 95 of the attachment to the license amendment.

E. BECO Corporate and Station Organization Changes

By letter dated June 18, 1973, BECO described a revision to their company line organization as previously shown in Figure 6.1.1 of the Technical Specifications. This letter also requested certain changes in the responsibilities and records of the Nuclear Safety Review and Audit Committee (NSRAC) to recognize the described organizational change. Responsibilities for review and reporting remain undiminished.

By letter dated November 12, 1973, BECO described further changes to both the company line organization and the Pilgrim Division organization. The significant changes were the elimination of the position of Assistant Division Head and the establishment of a Process Engineer position with supporting staff.

By letter dated April 8, 1974, BECO notified the Commission of the establishment of a Senior Plant Engineer position reporting to the Process Engineer. This position was established to improve certain administrative work within the station staff.

We have reviewed the changes to the organization as described in these two letters and find them consistent with our requirements for corporate support of licensed nuclear facilities and station staffing. Our review concluded that these organization changes do not result in any loss of authority or diminishment of responsibilities relative to the safe operation of the plant. Therefore, we find the organizational changes acceptable. These organizational changes result in revisions to pages 208, 209, 210, 212, and 213 of the Technical Specifications as marked in the attachment to the license amendment.

F. Requirements Conditional Upon Fuel Being in the Reactor Vessel

The Technical Specifications need clarification in two other instances on whether the requirements are conditional upon fuel being in the reactor vessel. Note 7 on page 28 and specifications 3.10.A and 3.10.B on page 202 of the Technical Specifications are revised to make them applicable to the condition "when fuel is in the vessel." We concluded that the addition of this conditional qualification does not reduce the safety provisions for this equipment and that it is not the intent to require that equipment be maintained in an operable condition when its operation can serve no safety function. We have included the phrase "during refueling and core alterations" to the applicability statement for specification 3.10 to make our intent clear and consistent with the applicability statement for specification 4.10 on the same page. These changes are indicated by a revision mark on pages 28 and 202 of the attachment to the license amendment.

G. Revised Protection Factors for Respirators

Recent tests by the AEC Respiratory Studies Group at Los Alamos Scientific Laboratory of full facepiece respirators operating in the demand mode as currently approved by the U.S. Bureau of Mines have indicated that the protection factor for particulates and vapors and gases is considerably less than the expected factor of 500. As a result of these tests, technical specifications containing allowances for the use of respiratory protective equipment shall be changed to reflect a protection factor of no more than 100 for full facepiece respirators operating in the demand mode for atmosphere-supplying respirators, both air-line and self-contained breathing apparatus. This change will increase the level of protection for station workers.

Accordingly, the Technical Specifications are hereby changed as set forth below:

> In Section 6.2.B reduce the protection factors given in Table 6.1.3, Protection Factors for Respirators, Section II, "Atmosphere-supplying respirators," Subsection 1. "Air-line respirators: Facepiece, full - Demand mode," and Subsection 2. "Self-contained breathing apparatus (SCBA): Facepiece full - Demand mode," for Column 3, "Particulates and vapors and gases except tritium oxide," from 500 to 100.

This change is shown on page 219 of the attachment to the license amendment.

H. Miscellaneous Changes for Clarification and Correction of Errata

A number of ambiguous statements and errors have been identified in the Technical Specifications attached to Amendment 1 of the Facility Operating License DPR-35 as issued on September 15, 1972. The following items are revised to improve the understanding and use of the document. In each case, the revision is marked on the noted page of the attachment to the license amendment.

1. Change to clarify the meaning of a statement in the Bases: On page 18 under B. APRM Control Rod Block change "becoming less than an MCHFR of unity" to "MCHFR becoming less than unity. (Amended on revision page 18a.)

2. Change to clarify associated items in a table and correct errata: On page 32, Table 4.1.2. For the columns <u>Group</u>, <u>Calibration Test</u>, and <u>Minimum Frequency</u> lower the entries level with "APRM High Flux" one line to make them level with "Output Signal". Under the column <u>Minimum Frequency</u> and level with "Flow Bias Signal" change:"Every 3 Months" to "Each Refueling Outage." This change makes the table consistent with the frequency of calibration described in the Bases on pages 37 and 38.

3. Removal of unnecessary instrument trip setting: When the RHR is operating in the shutdown cooling mode, the reactor primary coolant system is operating at a pressure of less than 110 psig. Since the portion of the RHR system outside containment is not designed to full reactor operating pressure and temperature, the containment isolation valves on the RHR outlet line are designed to close automatically whenever the primary coolant system exceeds the 110 psig pressure. The sensor initiating this action was previously required to be set at 100 $\pm$ 10 psig, or in the range between 90 psig and 110 psig. This change deletes any requirements for the setting to be greater than 90 psig. Our review of this matter finds that there is no safety reason for the lower tolerance and the inclusion of the lower tolerance was an error in writing the initial specification. Further, we conclude that any setting equal or below 110 psig provides equal protection to the low pressure portions of the RHR system. This specification is therefore changed from "100 $\pm$ 10 psig" to "$\leq$ 110 psig". This change occurs in the second item of Table 3.2.A on page 45 and the third item on Table 3.2.B on page 48, and is indicated by the revision mark in the right hand margin.

4. Changes to tolerances provided for instrument trip settings: Table 3.2.B provides trip level settings for initiating or controlling the core and containment cooling systems. Three of the trip settings on page 48 have been found to have errors in the assignment of the tolerance bands within which these instruments must be set and maintained. These are items 2, 4, and 5 on page 48. These tolerance bands were erroneously given as $\pm$ 10 psig. This small tolerance

is not within the capability of these instruments. Table 7.4.4 of the FSAR specifies that the two instruments with the 400 psig trip setting have an accuracy of ± 2%, and the instrument with the 900 psig trip setting should have an accuracy of ± 1%. Since instrument accuracy is normally specified as percent of full scale, we conclude that the instruments should have an accuracy of about ± 30 psig and ± 15 psig respectively. We have also discussed maintainability of these settings with the licensee. Based on our review of these instruments and our discussions with the licensee, we are changing all three tolerance settings to ± 25 psig. These changes are indicated by the revision marks in the right hand margin of revised page 48 of the Technical Specifications.

We have also reviewed the impact of these wider tolerance bands on the operation of the emergency core cooling systems and conclude that their performance will not be altered in any way which might change the capability of these systems to satisfy our ECCS criteria.

5. Clarification of testing requirements for ADS valves: At the meeting held on May 24, 1973, representatives of BECO identified an inadvertent omission in the Surveillance Requirement 4.5.E.2 (on page 110). When it is determined that one valve of the ADS is inoperable, the correct test on the other ADS valves is to demonstrate that the ADS subsystem actuation logic is operable. This is demonstrated by the correctly stated Surveillance Requirement 4.5.C.2, given for a similar operational situation. Therefore, we have amended Surveillance Requirement 4.5.E.2 to insert the phrase "the ADS subsystem actuation logic for." Our review of this matter concludes that this omission was an oversight and that the revision states the requirement as originally intended. Further, we conclude that the revision does not compromise the safety provisions of the specification and is acceptable. This change in the Technical Specifications is shown as marked on page 110 of the attachment to the license amendment.

6. Clarification of thermal and pressurization limits. Specification 3.6.A.2 on page 123 ended with the phrase "and fuel is in the reactor vessel." The specification was not clear whether fuel is required to be in the reactor vessel when the vessel is pressurized above 250 psig, or whether the reactor vessel temperatures must be equal to or greater than 180°F when pressurized above 250 psig when fuel is in the reactor vessel. Our review concludes the latter was intended. The last phrase is changed to "when fuel is in the reactor vessel." This change is shown on the revised page 123 of the attachment to the license amendment.

7. Inclusion of limitation on recirculation pump speeds at 80% power. Specification 3.6.F.1 on page 127 and the Bases on page 148 are unclear on the limitations on the recirculation pump speeds when the station is operating at 80% power. The conditions at "greater than" and "less than" 80% power were described; the conditions at 80% power were not mentioned. In order to define the limitations at 80% power, these pages are revised to include this condition with the limitation in effect for less than 80% power. These changes are shown on the revised pages 127 and 148 of the attachment to the license amendment.

CONCLUSION

The changes to Technical Specifications described above do not compromise the safe operation of the Pilgrim Nuclear Power Station. We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does 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) 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.

Original Signed by Original Signed by

Spottswood B. Burwell, Project Manager
Light Water Reactors Branch 2-1
Directorate of Licensing

Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch 2-1
Directorate of Licensing

Dated: JAN 7 1975