March 25, 1993

Docket No. 50-293

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

Dear Mr. Boulette:

SUBJECT: ISSUANCE OF AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M82782)

The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated February 7, 1992.

This amendment revises Reactor Protection System (RPS) instrumentation surveillance test intervals from 1 month to 3 months, provides for 12- and 6-hour allowable out-of-service times for repairs and tests, and deletes the water level perturbation requirement. Changes to Control Rod Block and Primary Containment Isolation System instrumentation common to RPS are also included, as well as appropriate bases changes.

In addition, some administrative changes are included to clarify nomenclature, correct a typographical error and provide information to operators.

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

Sincerely,
/S/
Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 147 to License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

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

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See next page

cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated February 7, 1992 complies 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 facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 147 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 25, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 147

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages have been provided.*

Remove	Insert
27	27
28	28
29	29
30	30
31	31
32	32
33	33
34	34
35	35
36	36
37	37
38	38
39	39
40	40
41	41
45	45
46	46
----	46a
54	54
54a	54a
58b	58b
63	63
67	67
74	74
76	76
77	77

PNPS Table 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Operable Inst. Channels Per Trip System (1)		Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action(1)
Minimum	Avail.			Refuel (7)	Startup/Hot Standby	Run	
1	1	Mode Switch in Shutdown		X	X	X	A
1	1	Manual Scram		X	X	X	A
		IRM					
3	4	High Flux	≤120/125 of full scale	X	X	(5)	A
3	4	Inoperative		X	X	(5)	A
		APRM					
2	3	High Flux	(15)	(17)	(17)	X	A or B
2	3	Inoperative	(13)	X	X(9)	X	A or B
2	3	High Flux (15%)	≤15% of Design Power	X	X	(16)	A or B
2	2	High Reactor Pressure	≤1085 psig	X(10)	X	X	A
2	2	High Drywell Pressure	≤2.5 psig	X(8)	X(8)	X	A
2	2	Reactor Low Water Level	≥9 In. Indicated Level	X	X	X	A
		SDIV High Water Level:	<39 Gallons	X(2)	X	X	A
2	2	East					
2	2	West					
2	2	Main Condenser Low Vacuum	≥23 In. Hg Vacuum	X(3)	X(3)	X	A or C
2	2	Main Steam Line High Radiation	≤7X Normal Full Power Background (18)	X	X	X(18)	A or C
4	4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	A or C
2	2	Turbine Control Valve Fast Closure	≥150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	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 trip function (e.g., high drywell pressure, reactor low water level, etc.). An instrument channel, satisfying minimum operability requirements for a trip system, may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

   An inoperable channel and/or trip system need not be placed in the tripped condition if this would cause a full scram to occur. When a trip system can be placed in the tripped condition without causing a full scram to occur, place the trip system with the most inoperable channels in the tripped condition, per the table below. If both systems have the same number of inoperable channels, place either trip system in the tripped condition, per the table below.

Condition	Required Action	Completion Time
a. With less than the minimum required operable channels per trip function in one trip system.	Place associated trip system in trip or *	12 hours
b. With less than the minimum required operable channels per trip function, in both trip systems.	Place one trip system in trip or *	6 hours
c. If full scram trip capability is not available for a given trip function	Restore RPS trip capability or *	1 hour

   * Initiate the actions required by Table 3.1.1 and specified in Actions A through D below for that function:

     A. Initiate insertion of operable rods and complete insertion of all operable rods within four (4) hours.

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

     C. Reduce turbine load and close main steam line isolation valves within eight (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 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. 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).
10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
11. Deleted
12. Deleted
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. Deleted
15. The APRM high flux trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT, but shall in no case exceed 120% of rated thermal power.
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.
18. Within 24 hours prior to the planned start of hydrogen injection with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the injection of hydrogen. The background radiation level and associated trip setpoints may be adjusted based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to withdrawing control rods at reactor power levels below 20% rated power.

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

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

NOTES FOR TABLE 4.1.1

1. Deleted

2. Deleted

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. Deleted

7. This APRM testing will be performed once every 3 months when in the RUN mode and within 24 hours after entering RUN mode, if not performed within the previous seven days.

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

Instrument Channel	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	Comparison to APRM on Controlled Shutdowns	Note (4)
	Full Calibration	Once/operating cycle
APRM High Flux		
Output Signal	Heat Balance	Once every 3 Days
Flow Bias Signal	Calibrate Flow Comparator and Flow Bias Network	Each Refueling Outage
	Calibrate Flow Bias Signal (1)	Every 3 Months
LPRM Signal	TIP System Traverse	Every 1000 Effective Full Power Hours
High Reactor Pressure	Note (7)	Note (7)
High Drywell Pressure	Note (7)	Note (7)
Reactor Low Water Level	Note (7)	Note (7)
High Water Level in Scram Discharge Tanks	Note (7)	Note (7)
Turbine Condenser Low Vacuum	Note (7)	Note (7)
Main Steam Line Isolation Valve Closure	Note (6)	Note (6)
Main Steam Line High Radiation	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive	Note (7)	Note (7)
Turbine Control Valve Fast Closure	Standard Pressure Source	Every 3 Months
Turbine Stop Valve Closure	Note (6)	Note (6)
Reactor Pressure Permissive	Note (7)	Note (7)

NOTES FOR TABLE 4.1.2

1. Adjust the flow bias trip reference, as necessary, to conform to a calibrated flow signal.

2. Calibration tests are not required when the systems are not required to be operable or are tripped.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Maximum frequency required is once per week.

5. Response time is not a part of the routine instrument channel test, but will be checked once per operating cycle.

6. Physical inspection and actuation of these position switches will be performed during the refueling outages.

7. Calibration of these devices will be performed during refueling outages.

   To verify transmitter output, a daily instrument check will be performed. Calibration of the associated analog trip units will be performed concurrent with functional testing as specified in Table 4.1.1.

BASES:

3.1 The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference FSAR Section 7.2). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic (i.e., an input signal on either one or both of the subchannels will cause a trip system trip). The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure, and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved (i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor). Three APRM instrument channels are provided for each protection trip system.

For some trip functions (e.g. MSIV or Turbine Stop Valve Position Switches), the loss of one instrument may lead to degradation of both trip systems. In these cases, a 6 hour LCO must be entered.

A source range monitor (SRM) system is also provided to supply additional neutron level information during refuel and startup (Reference FSAR Section 7.5.4).

3.1 BASES (Cont'd)

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

The APRM's cover the Refuel and Startup/Hot Standby modes with the APRM 15% scram, and the power range with the flow-biased 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 (Reference FSAR Section 7.5.7). 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 shifting to the Refuel mode during reactor power operation does not diminish the capability of the reactor protection system.

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to startup the unit. Below 305 psig turbine first stage pressure (45% of rated), the scram signal due to turbine stop valve closure or fast closure of turbine control valves is bypassed because flux and pressure scram are adequate to protect the reactor. If the scram signal due to turbine stop valve closure or fast closure of turbine control valves is bypassed at lower powers, less conservative MCPR and MAPLHGR operating limits may be applied as specified in the CORE OPERATING LIMITS REPORT.

Average Power Range Monitor (APRM)

APRM's #1 and #3 operate contacts in one subchannel and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5, and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel.

3.1 BASES (Cont'd)

The 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 abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses 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, the use of flow-referenced scram trip provides even additional margin.

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 proper margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

Analyses of the limiting transients show that no scram adjustment is required to assure the minimum critical power ratio (MCPR) is greater than the safety limit MCPR when the transient is initiated from MCPR above the operating limit MCPR.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides proper thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is sufficient to accommodate anticipated maneuvers associated with power plant 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 backed up by the rod worth minimizer.

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 rated power, the rate of power rise is very slow. Generally the heat flux is in the 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 rated power per minute, and the APRM system would be more than adequate to assure a scram before power could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 880 psig.

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

Intermediate Range Monitor (IRM)

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. 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 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on Range 5, the scram would be 120/125 of full scale 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 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 is exceeded.

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. 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 is scrammed and peak core power limited to one percent of rated power, thus maintaining MCPR above the safety limit MCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

Reactor Low Water Level

The setpoint 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 show that scram at this level properly protects the fuel and the pressure barrier, because MCPR

3.1 BASES (Cont'd)

remains well above the safety limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 15 inches below the normal operating range and is thus sufficient to avoid spurious scrams.

Turbine Stop Valve Closure

The turbine stop valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the safety limit MCPR even during the worst case transient that assumes the turbine bypass is closed.

Turbine Control Valve Fast Closure

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the acceleration relay. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. MCPR remains above the safety limit MCPR.

Main Condenser Low Vacuum

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram setpoint is selected to initiate a scram before the closure of the turbine stop valves is initiated.

Main Steam Line Isolation Valve Closure

The low pressure isolation of the main steam lines at 880 psig (as specified in Table 3.2.A) was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 785 psig requires the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram and APRM 15% scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire

3.1 BASES (Cont'd)

range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

High Reactor Pressure

The high reactor pressure scram setting is chosen slightly above the maximum normal operating pressure to permit normal operation without spurious scram, yet provide a wide margin to the ASME Section III allowable reactor coolant system pressure (1250 psig, see Bases Section 3.6.D).

High Drywell Pressure

Instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the Core Standby Cooling Systems (CSCS) initiation to minimize the energy that must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

Main Steam Line High Radiation

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors that cause an isolation of the main condenser off-gas line.

Reactor Mode Switch

The reactor mode switch actuates or bypasses the various scram functions appropriate to the particular plant operating status (Reference FSAR Section 7.2.3.9).

Manual Scram

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.

3.1 BASES (Cont'd)

Scram Discharge Instrument Volume

The control rod drive scram system is designed so that all of the water that is discharged from the reactor by a scram can be accommodated in the discharge piping. The two scram discharge volumes have a capacity of 48 gallons of water each and are at the low points of the scram discharge piping.

During normal operation the scram discharge volume system is empty; however, should it fill with water, the water discharged to the piping could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, redundant and diverse level detection devices in the scram discharge instrument volumes have been provided. From a reference zero established by analysis, the instruments will alarm at a water level less than 4.5 gallons, initiate a control rod block before the 18 gallon water level, and scram the reactor before the water level 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 to perform its function properly.

4.1 BASES

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with General Electric Company Topical Report NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," as approved by the NRC and documented in the safety evaluation report (NRC letter to T. A. Pickens from A. Thadani dated July 15, 1987).

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).

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating every three days using heat balance data and by calibrating individual LPRM's every 1000 effective full power hours using TIP traverse data.

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PNPS TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Operable Instrument Channels Per Trip System (1) Minimum	Available	Trip Function	Trip Level Setting	Action (2)
2(7)	2	Reactor Low Water Level	≥9" indicated level (3)	A and D
1	1	Reactor High Pressure	≤110 psig	D
2	2	Reactor Low-Low Water Level	at or above -49 in. indicated level (4)	A
2	2	Reactor High Water Level	≤48" indicated level (5)	B
2(7)	2	High Drywell Pressure	≤2.5 psig	A
2	2	High Radiation Main Steam Line Tunnel (9)	≤7 times normal rated full power background	B
2	2	Low Pressure Main Steam Line	≥880 psig (8)	B
2(6)	2	High Flow Main Steam Line	≤140% of rated steam flow	B
2	2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤170°F	B
2	2	Turbine Basement Exhaust Duct High Temperature	≤150°F	B
1	1	Reactor Cleanup System High Flow	≤300% of rated flow	C
2	2	Reactor Cleanup System High Temperature	≤150°F	C

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function. An instrument channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter; or, where only one channel exists per trip system, the other trip system shall be operable.

2. Action

   If the minimum number of operable instrument channels cannot be met for one of the trip systems of a trip function, the appropriate conditions listed below shall be followed:

   > If placing the inoperable channel(s) in the tripped condition would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within one hour (twelve hours for Reactor Low Water Level, High Drywell Pressure, and Main Steam Line High Radiation) or initiate the action required by Table 3.2.A for the affected trip functions.
   >
   > If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to operable status within two hours (six hours for Reactor Low Water Level, High Drywell Pressure, and Main Steam Line High Radiation) or initiate the Action required by Table 3.2.A for the affected trip function.

   If the minimum number of operable instrument channels cannot be met for both trip systems, place at least one trip system (with the most inoperable channels) in the tripped condition within one hour or initiate the appropriate Action required by Table 3.2.A listed below for the affected trip function.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

3. Instrument set point corresponds to 128.26 inches above top of active fuel.
4. Instrument set point corresponds to 77.26 inches above top of active fuel.
5. Not required in Run Mode (bypassed by Mode Switch).
6. Two required for each steam line.
7. These signals also start SBGTS and initiate secondary containment isolation.
8. Only required in Run Mode (interlocked with Mode Switch).
9. Within 24 hours prior to the planned start of hydrogen injection with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the injection of hydrogen. The background radiation level and associated trip setpoints may be adjusted based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to withdrawing control rods at reactor power levels below 20% rated power.

PNPS TABLE 3.2.C-1

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Instrument Channels per Trip Function Minimum	Available	Required Operational Conditions	Notes
APRM Upscale (Flow Biased)	4	6	Run	(1)
APRM Upscale	4	6	Startup/Refuel	(1)
APRM Inoperative	4	6	Run/Startup/Refuel	(1)
APRM Downscale	4	6	Run	(1)
Rod Block Monitor (Power Dependent)	2	2	Run, with limiting control rod pattern, and reactor power > LPSP	(2) (5)
Rod Block Monitor Inoperative	2	2	Run, with limiting control rod pattern, and reactor power > LPSP	(2) (5)
Rod Block Monitor Downscale	2	2	Run, with limiting control rod pattern, and reactor power > LPSP	(2) (5)
IRM Downscale	6	8	Startup/Refuel, except trip is bypassed when IRM is on its lowest range	(1)
IRM Detector not in Startup Position	6	8	Startup/Refuel, trip is bypassed when mode switch is placed in run	(1)
IRM Upscale	6	8	Startup/Refuel	(1)
IRM Inoperative	6	8	Startup/Refuel	(1)
SRM Detector not in Startup Position	3	4	Startup/Refuel, except trip is bypassed when SRM count rate is $\geq$ 100 counts/second or IRMs on Range 3 or above (4)	(1)
SRM Downscale	3	4	Startup/Refuel, except trip is bypassed when IRMs on Range 3 or above (4)	(1)

PNPS TABLE 3.2.C-1 (Con't)

Trip Function	Operable Instrument Channels per Trip Function Minimum	Available	Required Operational Conditions	Notes
SRM Upscale	3	4	Startup/Refuel, except trip is by-passed when the IRM range switches are on Range 8 or above (4)	(1)
SRM Inoperative	3	4	Startup/Refuel, except trip is by-passed when the IRM range switches are on Range 8 or above (4)	(1)
Scram Discharge Instrument Volume Water Level - High	2	2	Run/Startup/Refuel	(3)
Scram Discharge Instrument Volume-Scram Trip Bypassed	1	1	Run/Startup/Refuel	(3)
Recirculation Flow Converter - Upscale	2	2	Run	(1)
Recirculation Flow Converter - Inoperative	2	2	Run	(1)
Recirculation Flow Converter - Comparator Mismatch	2	2	Run	(1)

TABLE 3.2.F (Cont'd)
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Parameter	Type Indication and Range	Notes
1	RI 1001-609 RR 1001-608	Reactor Building Vent	Indicator/Multipoint Recorder 10^{-1} to 10^{4} R/hr	(4) (7)
1	RI 1001-608 RR 1001-608	Main Stack Vent	Indicator/Multipoint Recorder 10^{-1} to 10^{4} R/hr	(4) (7)
1	RI 1001-610 RR 1001-608	Turbine Building Vent	Indicator/Multipoint Recorder 10^{-1} to 10^{4}R/hr	(4) (7)

PNPS TABLE 4.2.C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
APRM - Downscale	Once/3 Months	Once/3 Months	Once/Day
APRM - Upscale	Once/3 Months	Once/3 Months	Once/Day
APRM - Inoperative	Once/3 Months	Not Applicable	Once/Day
IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Inoperative	(2) (3)	Not Applicable	(2)
RBM - Upscale	Once/3 Months	Once/6 Months	Once/Day
RBM - Downscale	Once/3 Months	Once/6 Months	Once/Day
RBM - Inoperative	Once/3 Months	Not Applicable	Once/Day
SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
SRM - Inoperative	(2) (3)	Not Applicable	(2)
SRM - Detector Not in Startup Position	(2) (3)	Not Applicable	(2)
SRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Detector Not in Startup Position	(2) (3)	Not Applicable	(2)
Scram Discharge Instrument Volume Water Level-High	Once/3 Months	Refuel	Not Applicable
Scram Discharge Instrument Volume-Scram Trip Bypassed	Once/3 Months	Not Applicable	Not Applicable
Recirculation Flow Converter	Not Applicable	Once/Cycle	Once/Day
Recirculation Flow Converter-Upscale	Once/3 Months	Once/3 Months	Once/Day
Recirculation Flow Converter-Inoperative	Once/3 Months	Not Applicable	Once/Day
Recirculation Flow Converter-Comparator Off Limits	Once/3 Months	Once/3 Months	Once/Day
Recirculation Flow Process Instruments	Not Applicable	Once/Cycle	Once/Day

Logic System Functional Test (4) (6)

System Logic Check	Once/18 Months

NOTES FOR TABLES 4.2.A THROUGH 4.2.G

1. Initially once per month until exposure hours (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.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations of IRMs and SRMs shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and main steam line high radiation are not included on Table 4.2.A since they are tested on Tables 4.1.1 and 4.1.2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. Calibration of analog trip units will be performed concurrent with functional testing. The functional test will consist of injecting a simulated electrical signal into the measurement channel. Calibration of associated analog transmitters will be performed each refueling outage.

BASES:

4.2 The instrumentation listed in Table 4.2.A thru 4.2.H will be functionally tested and/or calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Conservatively assuming that those instruments which have their contacts arranged in 1 out of n logic cannot be used during a testing sequence, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where:

i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^{3} \text{ hours}$$

$$= 40 \text{ days}$$

For additional margin a test interval of once per month will be used initially.

(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

4.2 BASES (Cont'd)

is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more unusual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the refueling area ventilation duct which initiate building isolation and Standby Gas Treatment operation are arranged in two 1 out of 2 logic systems. The bases given above apply here also and were used to arrive at the functional testing frequency. Based on experience with instruments of similar design, a testing interval of once every three months has been found adequate.

4.2 BASES (Cont'd)

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The instrumentation which is required for the recirculation pump trip and alternate rod insertion systems incorporate analog transmitters. The transmitter calibration frequency is once per refueling outage, which is consistent with both the equipment capabilities and the requirements for similar equipment used at Pilgrim. The Trip Unit Calibration and Instrument Functional Test is specified at monthly, which is the same frequency specified for other similar protective devices. An instrument check is specified at once per day; this is considered to be an appropriate frequency, commensurate with the design applications and the fact that the recirculation pump trip and alternate rod insertion systems are backups to existing protective instrumentation.

Control Rod Block and PCIS instrumentation common to RPS instrumentation have surveillance intervals and maintenance outage times selected in accordance with NEDC-30851P-A, Supplements 1 and 2 as approved by the NRC and documented in SERs (letters to D. N. Grace from C. E. Rossi dated September 22, 1988 and January 6, 1989).

A logic system functional test interval of 18 months was selected to minimize the frequency of safety system inoperability due to testing and to minimize the potential for inadvertent safety system trips and their attendant transients. Based on industry experience and BWR Standard Technical Specifications, an 18 month testing interval provides adequate assurance of operability for this equipment.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.147 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated February 7, 1992, Boston Edison Company (BECo/the licensee) submitted a request to amend license DPR-35, of Pilgrim Nuclear Power Station for proposed changes to the Technical Specifications (TS) (Ref.1). The proposed TS changes would affect the Reactor Protection System (RPS) surveillance test intervals (STI) and allowable out-of-service times for repair to RPS instrument channels in accordance with 10 CFR 50.90. Changes to Control Rod Block and Primary Containment Isolation Systems (PCIS) instrumentation common to RPS are also proposed, delete the water level perturbation requirement, as well as appropriate bases changes. Administrative changes to clarify nomenclature, correct a typographical error and provide information to operators were also provided.

2.0 BACKGROUND

On July 8, 1983, the NRC staff issued Generic Letter (GL) 83-28, requesting that all licensees and applicants respond to the generic issues raised by analysis of the Salem anticipated transient without scram (ATWS) events (Ref. 2). The GL requested that licensees and applicants review existing STIs for RPS instrumentation required by TS to assure that current and proposed intervals for such testing are consistent with achieving high RPS availability. In late 1983, the staff issued NUREG-1024, "Technical Specifications - Enhancing the Safety Impact," which recommended that TS surveillance requirements and action statements be reviewed to assure that they have an adequate technical basis and minimize risk (Ref. 3). Probabilistic risk assessment (PRA) analysis may be used as a basis for TS changes. The BWR Owners Group (BWROG) formed a TS improvement Committee in late 1983 to develop a program that used PRA to identify improvements to STIs specified in BWR TS. The BWROG commissioned General Electric Company (GE) to perform a generic analysis which could be used by individual BWR plants.

3.0 EVALUATION

The licensee proposes to change the TS by increasing the RPS instrument functional test interval and calibration of analog trip units from 1-month intervals to every 3 months. Similar changes to test PCIS instruments common to RPS instrumentation are being proposed. The proposed changes can result in reduced potential for human error rates during testing and reduce the potential for component wear-out caused by testing. This change should result in less reactor scrams and man-hours per year required to perform such testing.

Changes to the TS include a repair/test time interval of 12 and 6 hours. These surveillance and out-of-service times were selected to allow reasonable repair/test times without placing induced stress on plant personnel that can contribute to human error. These proposed changes are supported by studies made by GE for BWROG.

The BWROG submitted to the NRC for approval the GE Topical Reports (TR) NEDC-30844, which was a response to GL 83-28 and NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reaction Protection Systems." Topical Reports NEDC-30844 and NEDC-30851P were approved by the NRC on July 15, 1987 (Ref. 4). The approval of the TRs required each licensee to assure that these generic TR enveloped their facilities.

In September 1987, an analysis specific to Pilgrim Nuclear Power Station (PNPS), MDE-31 30851P-A was performed by GE to confirm the applicability of the generic study to PNPS. This analysis was reviewed by BECo to ensure that design configuration differences from the generic study were properly evaluated. This analysis indicated that there was a slight increase in RPS failure frequency during the increased surveillance interval; however, this was more than offset by the expected reduction in inadvertent plant scrams and challenges to safe shutdown systems, reduced equipment test cycling wear, and diversion of personnel required to perform the tests. On April 27, 1988, the NRC issued a letter to the BWROG providing guidance and clarification regarding the NRC's requirement for confirmation of instrument drift allowance on trip setpoint (Ref. 5).

Prior to the April 27, 1988, NRC letter to the BWROG, BECo addressed instrument drift in a letter dated February 22, 1988. This letter supplied additional information to their December 23, 1985, proposed TS change concerning the out-of-service-time for testing and calibrating the RPS and PCIS systems. A Topical Report MDE-31-0286 compared the PNPS RPS instrumentation configuration, including the recently installed BECo Analog Trip System (ATS), against the BWR generic model. GE advised BECo that instrument drift was included for the ATS equipment only, and is based on equipment vendor information. The drift data in the generic analysis is applicable to PNPS (Ref. 6). The licensee stated that RPS instrument analog trip units setpoint calculations assume a drift over a 6-month period; therefore, setpoint changes to account for drift are not necessary.

The licensee proposes to delete the requirements to perturb reactor water level after functional testing to minimize the potential for an inadvertent plant transient. The licensee states that this testing is not needed because reactor water level channel checks verify proper instrument valve lineup and sensor response. The use of instrument check is consistent with Standard Technical Specifications, acceptable IEEE-279 on-line sensor check methods, and PNPS design. Note 7 of Table 4.1.2 of the TS requires a daily instrument check to verify transmitter output and calibration of the associated analog trip units will be performed concurrent with the functional testing every 3 months.

The staff concludes that the proposed changes to the TSs, submitted by the licensee on February 7, 1992, for the RPS surveillance test intervals, and allowable out-of-service times for surveillance, and repair, are acceptable based upon GE's Topical Reports NEDC-30844, NEDC-30851P and plant specific report MDE-31-0286. We also find acceptable the removal of the requirement for water level perturbation which is a means for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation because IEEE Standard 279, Requirement 4.9 allows also that this availability determination can be accomplished by cross checking between channels that bear a known relationship to each other and have read-outs available. The administrative changes were not technical, but corrective and clarifying in nature and did not affect the technical bases of the amendment.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 7807). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Frederick P. Paulitz

Date: March 25, 1993

7.0 REFERENCES

1. Letter from R. A Anderson, SVP, BECo to NRC "Proposed Technical Specification Change to Reactor Protection System Surveillance Test Intervals and Allowable Out-of Service Times" dated February 7, 1992.

2. Letter from D.E. Eisenhut, NRC, to all Licensees of Operating Reactors... "Requested Action Based on Generic Implications of Salem ATWS Events, (Generic Letter 83-28)" dated July 8, 1983.

3. NUREG-1024. "Technical Specification - Enhancing the Safety Impact", dated November 1983.

4. Letter from A.C. Thadani, NRC, to T.A. Pickens, BWR Owners Group "General Electric Company Topical Reports NEDC-30844,'BWR Owners Group Response to NRC Generic Letter 83-28', and NEDC-30851P, 'Technical Specification Improvement Analysis for BWR Reactor Protection Systems'" dated July 15, 1987.

5. Letter, C.E. Rossi, NRC to R.F. Janecek (BWROG), "Staff Guidance for Licensee Determination that the Drift Characteristics for Instrumentation Used in RPS Channels are Bounded by NEDC-30851 Assumptions When the Functional Test Interval is Extended from Monthly to Quarterly" dated April 27, 1988.

6. Letter from R.G. Bird, SVP, BECO, to NRC, "Supporting Information for Proposed Technical Specification Changes for Tables 3.1.1 and 3.2.A" dated February 22, 1988.