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. 133
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 August 21, 1990, as supplemented on November 8, 1990 and December 3, 1990, 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. 133, 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

Richard Wessman, 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: January 2, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 133

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.

Remove	Insert
i	i
ii	ii
iii	ii
1	1
6	6
7	7
8	8
8a	---
9	9
10	10
11	---
12	---
13	---
13a	---
13b	---
14	---
15	---
16	---
17	---
18	---
19	---
20	---
21	---
22	---
23	---
24	---
25	---
26	26
27	27
29	29
36	36
37	37
38	38
39	39
40	40
---	40a*
---	40b*
---	40c*
55a	55a
71	71
72	72

*Donates new page

Remove	Insert
126	126
127	127
127A	127a
145	145
146	146
205A	205a
205A-1	---
205B	205b
205B-1	---
205B-2	---
205C	205c
205C-1	---
205C-2	---
205C-3	---
205C-4	---
205C-5	---
205C-6	---
205D	205d
205E	205e
205E-1	---
205E-2	---
205E-3	---
205E-4	---
205E-5	---
205E-6	---
205F	205f
205G	---
205H	---
206m	206m
216	216
217	217
---	217a*

*Donates new page

TABLE OF CONTENTS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

D. CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is a reload-cycle specific document, its supplements and revisions, that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual specifications.

2.1 SAFETY LIMITS

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.04.

2.1.3 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 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 SAFETY LIMIT VIOLATION

With any Safety Limit not met the following actions shall be met:

2.2.1 Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.

2.2.2 Within two hours:

A. Restore compliance with all Safety Limits, and
B. Insert all insertable control rods.

2.2.3 The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.

2.2.4 A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.

2.2.5 Critical operation of the unit shall not be resumed until authorized by the Commission.

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish a Safety Limit such that the Minimum Critical Power Ratio (MCPR) is not less than the limit specified in Specification 2.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling (i.e., MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

FUEL CLADDING INTEGRITY (2.1.1)

GE critical power correlations are applicable for all critical power calculations at pressures at or above 785 psig or core flows at or above 10% of rated flow. For operation at low pressures and low flows another basis is used as follows:

(continued)

FUEL CLADDING INTEGRITY (2.1.1) (continued)

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psig	
Max Flux:	0.1 to 1.25x10^6 lb/hr-ft^2	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to 1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	16,64 Rods in an 8x8 array 49 Rods in an 7x7 array	

MINIMUM CRITICAL POWER RATIO (2.1.2)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at which

(continued)

boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. Reference 1 includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.

The statistical analysis used to determine the MCPR safety limit is based on a model of the BWR core which simulates the process computer function. The reactor core selected for these analyses was a large 764 assembly, 251 inch reload core. Results from the large reload core analysis apply for all operating reactors for all reload cycles, including equilibrium cycles. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculational uncertainties, and statistical uncertainty associated with the critical power correlations are imposed upon the analytical representation of the core and the resulting bundle critical power ratios. Details of this statistical analysis are presented in Reference 2.

REACTOR WATER LEVEL (Shutdown Condition) (2.1.3)

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.1.0 for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

REFERENCES

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (Applicable Amendment specified in the CORE OPERATING LIMITS REPORT).

2. General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

3. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340).

 (Next page is 26)

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milli-seconds.

B. The maximum fraction of limiting power density (MFLPD) shall be less than or equal to the fraction of rated power (FRP) when thermal power is greater than or equal to 25% of rated thermal power.

1. If MFLPD is greater than FRP, adjust the APRM high flux scram and rod block trip setpoints to the relationships specified in the CORE OPERATING LIMITS REPORT within 6 hours.

2. If the required actions and associated completion times of Specification 3.1.B.1, above cannot be met, reduce thermal power to less than 25% of rated thermal power within 4 hours.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2 respectively.

B. Verify the maximum fraction of limiting power density is less than or equal to the fraction of rated power once within 12 hours after thermal power is greater than or equal to 25% of rated thermal power and every 24 hours thereafter.

TABLE 3.1.1 REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum Number 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
3	Inoperative		X	X	(5)	A
	APRM					
2	High Flux	(15)	(17)	(17)	X	A or B
2	Inoperative	(13)	X	X(9)	X	A or B
2	High Flux (15%)	≤15% of Design Power	X	X	(16)	A or B
2	High Reactor Pressure	≤1085 psig	X(10)	X	X	A
2	High Drywell Pressure	≤2.5 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 Instr. Volume	≤39 Gallons	X(2)	X	X	A
2	Main 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 (18)	X	X	X(18)	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	Turbine Control 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 (C''D)

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.

been provided to allow for bypassing of one such channel.

APRM

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 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.

To facilitate the implementation of this technique, Figure 4.1.1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M = nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of one month will be used initially until a trend is established.

Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is

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.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR 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 adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate 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

a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For 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 20×10^{-6} failure/hour. The bi-stable trip circuits are predicted to have an unsafe failure rate of less than 2×10^{-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.

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.

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

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

Group (D) devices, while similar in description to those in Group (B), are different in use because they (the analog transmitter/trip unit devices) provide alarms, trips or scram functions. An availability analysis is detailed in NEDO-21617A (12/78).

3.1 BASES (Cont'd)

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 set point for low level scram is above the bottom of the separator skirt. This level has

4.1 BASES (Cont'd)

Surveillance frequencies for the SDV system instrumentation is detailed in Amendment Number 65. NRC concurrence with this surveillance program is contained in the Safety Evaluation Report and its associated Technical Evaluation Report (TER-C-5506-66) dated 11/10/82.

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

1. Passive type indicating 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, 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.

3.1 BASES (Cont'd)

been used in transient analyses dealing with coolant inventory decrease. The results show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR 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 25 in. below the normal operating range and is thus adequate 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

4.1 BASES (Cont'd)

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.

B. The Maximum Fraction of Limiting Power Density (MFLPD) shall be checked once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the MFLPD is not expected to change significantly and thus a daily check of the MFLPD is adequate.

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.

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 set point 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 that 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 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 Basis Section 3.6.D).

High Drywell Pressure

Instrumentation (pressure switches) for the drywell are 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 cooling systems (CSCS) initiation to minimize the energy which 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 which cause an isolation of the main condenser off-gas line.

Reactor Mode Switch

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.2.3.7 FSAR.

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.

Scram Discharge Instrument Volume

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 two scram discharge volumes accommodate in excess of 39 gallons of water each and are at the low points of the scram discharge piping. No credit was taken for these volumes in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

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 which will alarm when water level reaches 4.5 gallons, initiate a control rod block at 18 gallons, and scram the reactor when 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 adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information 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 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. 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 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 $\leq$15% design power in the "Refuel" and "Startup/Hot Standby" modes and the backup IRM scram at $\leq$120/125 of full 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 scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in the CORE OPERATING LIMITS REPORT when the MFLPD is greater than the fraction of rated power (FRP). In a similar manner, the APRM rod block trip setting is adjusted downward if MFLPD exceeds FRP, thus preserving the APRM rod block safety margin.

TABLE 3.2.C-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

Trip Function	Trip Setpoint
APRM Upscale	(1) (2)
APRM Inoperative	Not Applicable
APRM Downscale	$\geq$ 2.5 Indicated on Scale
Rod Block Monitor (Flow Biased)	(1)
Rod Block Monitor Inoperative	Not Applicable
Rod Block Monitor Downscale	$\geq$ 5/125 of Full Scale
IRM Downscale	$\geq$ 5/125 of Full Scale
IRM Detector not in Startup Position	Not Applicable
IRM Upscale	$\leq$ 108/125 of Full Scale
IRM Inoperative	Not Applicable
SRM Detector not in Startup Position	Not Applicable
SRM Downscale	$\geq$ 3 counts/second
SRM Upscale	$\leq 10^5$ counts/second
SRM Inoperative	Not Applicable
Scram Discharge Instrument Volume Water Level - High	$\leq$ 18 gallons
Scram Discharge Instrument Volume - Scram Trip Bypassed	Not Applicable
Recirculation Flow Converter - Upscale	$\leq$ 120/125 of Full Scale
Recirculation Flow Converter - Inoperative	Not Applicable
Recirculation Flow Converter - Comparator Mismatch	$\leq$ 10% Flow Deviation for > 80% Rated Power, and $\leq$ 15% Flow Deviation for $\leq$ 80% Rated Power

(1) The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

(2) When the reactor mode switch is in the refuel or startup positions, the APRM rod block trip setpoint shall be less than or equal to 13% of rated thermal power, but always less than the APRM flux scram trip setting.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the Safety Limit MCPR. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

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 the condition of a MCPR less than the Safety Limit MCPR. 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 flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 107% of rated thermal power because of the APRM rod block trip setting. 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 trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. In the startup and refuel modes, the APRM rod block function is setdown below the APRM flux scram trip.

The RBM rod block function provides local protection of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are as shown in Table 3.2.C-2.

3.2 BASES (Cont'd)

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate the Reactor Building Isolation and Control System and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Four instrument channels are arranged in a 1 out of 2 twice trip logic.

Trip settings of < 100 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The alarm unit in each integrator is set to annunciate before the values specified in Specification 3.6.C are exceeded. A system whereby the time interval to fill a known volume will be utilized to provide a back-up to the flow integrators. An air sampling system is also provided to detect leakage inside the primary containment.

3.6.C Coolant Leakage (Cont'd)

power operation is permissible only during the succeeding seven days.

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

D. Safety and Relief Valves

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

   The nominal setpoint for the relief/safety valves shall be selected between 1095 and 1115 psig. All relief/safety valves shall be set at this nominal setpoint ± 11 psi. The safety valves shall be set at 1240 psig ± 13 psi.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours. Note: Technical Specifications 3.6.D.2 - 3.6.D.5 apply only when two Stage Target Rock SRVs are installed.

3. If the temperature of any safety relief discharge pipe exceeds 212°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.

4.6

D. Safety and Relief Valves

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

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

3. Whenever the safety relief valves are required to be operable, the discharge pipe temperature of each safety relief valve shall be logged daily.

4. Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

5. Notwithstanding the above, as a minimum, safety relief valves that have been in service shall be tested in the as-found condition during both Cycle 6 and Cycle 7.

3.6.D Safety Relief Valves (Con't)

4. Any safety relief valve whose discharge pipe temperature exceeds 212°F for 24 hours or more shall be removed at the next cold shutdown of 72 hours or more, tested in the as-found condition, and recalibrated as necessary prior to reinstallation. Power operation shall not continue beyond 90 days from the initial discovery of discharge pipe temperatures in excess of 212°F for more than 24 hours without prior NRC approval of the engineering evaluation delineated in 3.6.D.3.

5. The limiting conditions of operation for the instrumentation that monitors tail pipe temperature are given in Table 3.2.F.

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.

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 delta P characteristics by more than 10%.

LIMITING CONDITIONS FOR OPERATION

3.6.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%.

2. If Specification 3.6.F.1 is exceeded immediate corrective action shall be taken. If recirculation pump speed mismatch is not corrected within 30 minutes, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours unless the recirculation pump speed mismatch is brought within limits sooner.

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," Articles IWA, IWB, IWC, IWD and IWF and mandatory appendices as required by 10CFR50, Section 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).

SURVEILLANCE REQUIREMENTS

4.6.F Jet Pump Flow Mismatch

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

G. Structural Integrity

Inservice inspection of components shall be performed in accordance with the PNPS Inservice Inspection Program. The results obtained from compliance with this program will be evaluated at the completion of each ten year interval. The conclusions of this evaluation will be reviewed with the NRC.

BASES:

3.6.D and 4.6.D

Safety and Relief Valves

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

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

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

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

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

Item 3 above indicates that no credit be taken for the primary scram signal generated by closure of the main steam isolation valves. Two other scram initiation signals would be generated, one due to high neutron flux and one due to high reactor pressure. Thus item 3 will be satisfied by assuming a scram due to high neutron flux.

Relieving capacity of 40% (4 relief/safety valves) results in a peak pressure during the transient conditions used in the safety valve sizing analysis which is well below the pressure safety limit.

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

Experience in safety valve operation shows that a testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value of ± 1% is in accordance with Section III of the ASME Boiler and Pressure Vessel Code. An analysis has been performed which shows that with all safety valves set 1% higher, the reactor coolant pressure safety limit of 1375 psig is not exceeded.

BASES:

3.6.D and 4.6.D

Safety and Relief Valves

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification 3.5.D. In addition, the valves can be operated manually.

Pilgrim's experience with 2 stage safety/relief valves has demonstrated that minimum leakage exists when the tailpipe temperature is 215° Fahrenheit. Therefore, a reporting requirement triggered by a temperature of 212°F is conservative, and assures timely reporting before leakage reaches significant proportions.

3.11 REACTOR FUEL ASSEMBLY

Applicability:

The Limiting Conditions for Operation associated with fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation with both recirculation pumps operating, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the applicable limiting value specified in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLY

Applicability:

The surveillance requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

During reactor power operation, the LHGR shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

1. During power operation MCPR shall be $\geq$ the MCPR operating limit specified in the Core Operating Limits Report. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

2. The value of τ in Specification 3.11.C.2. shall be equal to 1.0 unless determined from the result of surveillance testing of Specification 4.3.C as follows:

a) τ is defined as

$$\tau = \frac{\tau_{ave} - \tau_B}{1.275 - \tau_B}$$

LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio MCPR (Cont'd)

For core flows other than rated the MCPR limits shall be the limits identified above times K_f where K_f is as shown in Figure 3.3.1 of the Core Operating Limits Report.

As an alternative method providing equivalent thermal-hydraulic protection at core flows other than rated, the calculated MCPR may be divided by K_f, where K_f is as shown in Figure 3.3.1 of the Core Operating Limits Report.

2. The operating limit MCPR values as a function of the τ are given in Table 3.3.1 of the Core Operating Limits Report where τ is given by specification 4.11.C.2.

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio MCPR (Cont'd)

b) The average scram time to the 30% insertion position is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

Where: an n = number of surveillance tests performed to date in the cycle.

Ni = number of active control rods measured in the ith surveillance test.

τi = average scram time to the 30% insertion position of all rods measured in the 1ith surveillance test.

c) = The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insertion time to the 30% position 0.945 sec.

Ni = total number of active control rod

σ = standard deviation of the distribution for average scram insertion time to the 30% position, 0.064 sec.

D. Power/Flow Relationship During Power Operation

The power/flow relationship shall not exceed the limiting values specified in the CORE OPERATING LIMITS REPORT.

If at any time during power operation it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

D. Power/Flow Relationship During Power Operation

Compliance with the power/flow relationship in Section 3.11.D shall be determined daily during reactor operation.

BASES:

3.11.A Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50, Appendix K.

The analytical method used to determine the APLHGR limiting values is described in the topical reports listed in Specification 6.9.A.4.

3.11.B Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate.

The analytical method used to determine the LHGR limiting value is described in the topical reports listed in Specification 6.9.A.4.

3.11.C Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

For any abnormal operating transient analysis with the initial condition of the reactor at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming the instrument trip settings given in Tables 3.1.1, 3.2.A and 3.2.B.

The analytical method used to determine the Operating Limit MCPR values in the CORE OPERATING LIMITS REPORT is described in the topical reports listed in Specification 6.9.A.4. By maintaining MCPR greater than or equal to the Operating Limit MCPR, the Safety Limit MCPR specified in Specification 2.1.2 is maintained in the event of the most limiting abnormal operating transient.

BASES:

3.11.D Power/Flow Relationship During Power Operation

The power/flow curve is the locus of core thermal power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

4.11.C Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Pilgrim Nuclear Power Station is located on the Western Shore of Cape Cod Bay in the Town of Plymouth, Plymouth County, Massachusetts. The site is located at approximately 41°51' north latitude and 70°35' west longitude on the Manomet Quadrangle, Massachusetts, Plymouth County 7.5 Minute Series (topographic) map issued by U.S. Geological Survey. UTM coordinates are 19-46446N-3692E.

The reactor (center line) is located approximately 1800 feet from the nearest property boundary.

5.2 REACTOR CORE

The reactor vessel core design shall be as described in the CORE OPERATING LIMITS REPORT and shall be limited to those fuel assemblies which have been analyzed with NRC-approved codes and methods and approved by the NRC in its acceptance of Amendment 22 of GESTAR II.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2.2 of the FSAR. The applicable design codes shall be as described in Table 4.2.1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2.1 of the FSAR. The applicable design codes shall be as described in Section 12.2.2.8 of the FSAR.

B. The secondary containment shall be as described in Section 5.3.2 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

6.7 Deleted

6.8 PROCEDURES

A. Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7 - 1972 and Appendix "A" of USNRC Regulatory Guide 1.33, except as provided in 6.8.B and 6.8.C below.

B. Each procedure of 6.8.A above, and changes thereto, shall be reviewed by the ORC and approved by the responsible department manager prior to implementation. These procedures shall be reviewed periodically as set forth in administrative procedures,

NOTE: ORC review and approval of procedures for vendors/contractors, who have a QA Program approved by Boston Edison Company, is not required for work performed at the vendor/contractor facility.

C. Temporary changes to procedures of 6.8.A above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's license on the unit affected.

3. The change is documented, subsequently reviewed by the ORC within 14 days of implementation, and approved by the responsible department manager.

D. Written procedures to implement the Fire Protection Program shall be established, implemented and maintained.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Commission.

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Monthly Operating Report

Routine reports of operating statistics, shutdown experience and forced reductions in power shall be submitted on a monthly basis to the Commission to arrive no later than the 15th of each month following the calendar month covered by the report.

The Monthly Operating Report shall include a narrative summary of operating experience that describes the operation of the facility, including safety-related maintenance, for the monthly report period.

6.9.A Routine Reports (Continued)

3. Occupational Exposure Tabulation

A tabulation of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g. reactor operations and surveillance inservice inspection, routine maintenance, special maintenance (including a description), waste processing, and refueling shall be submitted on an annual basis. This tabulation supplements the requirements of 20.407 of 10 CFR 20. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

4. Core Operating Limits Report

a) Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

b) The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (the approved version at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT) and in NEDO-21696, "Loss of Coolant Analysis Report for Pilgrim Nuclear Power Station," dated August 1977, (the approved version at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT).

c) The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) are met.

d) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.B Deleted