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

BOSTON EDISON CJMPANY

DOCKET NO. 50-?93

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. DPR-35

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

   A. The submittals by Boston Edison Company (the licensee) dated May 1, 1975; September 1, November 12, November 15, 1976; July 20, August 8, August 24, 1977; February 1, March 22, 1978; September 27, December 12, December 31, 1979; February 5, March 28, April 3, April 7, April 17, April 24, and April 29, 1980, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The 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;

   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:

8005230588

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 42 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Branch Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Revise Appendix A As Follows:

Remove	Insert
6	6
7	7
8	8
-	8a
9	9
12	12
13	13
13a	-
13b	-
16	16
17	17
18	18
26	26
27	27
29	29
40	40
43	43
-	44a
45	45
48	48
50	50
-	50a
51	51
-	53a
54	54
55	55
-	59a
61	61
-	66a
67	67
71	71
73	73
77	77
91	91
103	103
104	104
106	106
107	107
108	108

Remove	Insert
109	109
124	124
125	125
126	126
127	127
137e	137e
141	141
142	142
146	-
158	158
-	158a
-	158b
-	158c
161	161
165	165
172	172
173	173
174	174
-	174a
-	174b
177	177
178	178
180	180
181	181
194	194
-	194a
195	195
197	197
199	199
200	200
205a	205a
205a-1	205a-1
205b	205b
205c	205c
205c-1	-
205c-2	205c-2
205c-3	205c-3
205c-4	205c-4
205c-5	205c-5
205c-6	205c-6
205d	205d
205e	-
-	205e-1
-	205e-2
-	205e-3
-	205e-4
-	205e-5
205f	-
206m	206m

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Reactor Pressure > 800 psia and Core Flow >10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit. A MCPR of 1.07 is hereinafter referred to as the Safety Limit MCPR.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia and/or Core Flow ≤ 10%)

When the reactor pressure is ≤ 800 psia or core flow is less than or equal to 10% of rated, the steady state core thermal power shall not exceed 25% of design thermal power.

C. Power Transient

The safety limit shall be assumed to be exceeded when scram is known to have been accomplished by a means other than the expected scram signal unless analyses demonstrate that the fuel cladding integrity safety limits defined in Specifications 1.1A and 1.1B were not exceeded during the actual transient.

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limits from being exceeded.

Specification:

A. Neutron Flux Scram

The limiting safety system trip settings shall be as specified below:

1. Nuetron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq .65W + 55\%$ 2 loop

Where:

S = Setting in percent of rated thermal power (1998 MWt)

W = Percent of drive flow to produce a rated core flow of 69 M lb/hr.

1.1 SAFETY LIMIT

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

2.1 LIMITING SAFETY SYSTEM SETTING

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.65W + 55\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{2 Loop}}$$

Where,

FRP = fraction of rated thermal power (1998 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8 and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. <u>APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)</u>

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. <u>IRM</u>

The IRM flux scram setting shall be $\leq$120/125 of scale.

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

The APRM rod block trip setting shall be:

$$S_{RB} \leq 0.65W + 42\% \quad \underline{\text{2 Loop}}$$

Where,

S_{RB} = Rod block setting in percent of rated thermal power (1998 MWt)

W = Percent of drive flow required to produce a rated core flow of 69M lb/hr.

In the event of operating with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S_{RB} \leq (0.65\ W + 42\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{2\ Loop}$$

Where,

FRP = fraction of rated thermal power

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8 and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

C. Reactor low water level scram setting shall be $\geq$ 9 in. on level instruments.

D. Turbine stop valve closure scram setting shall be $\leq$ 10 percent valve closure.

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

F. Condenser low vacuum scram setting shall be $\geq$ 23 in. Hg. vacuum.

G. Main steam isolation scram setting shall be $\leq$ 10 percent valve closure.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

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

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

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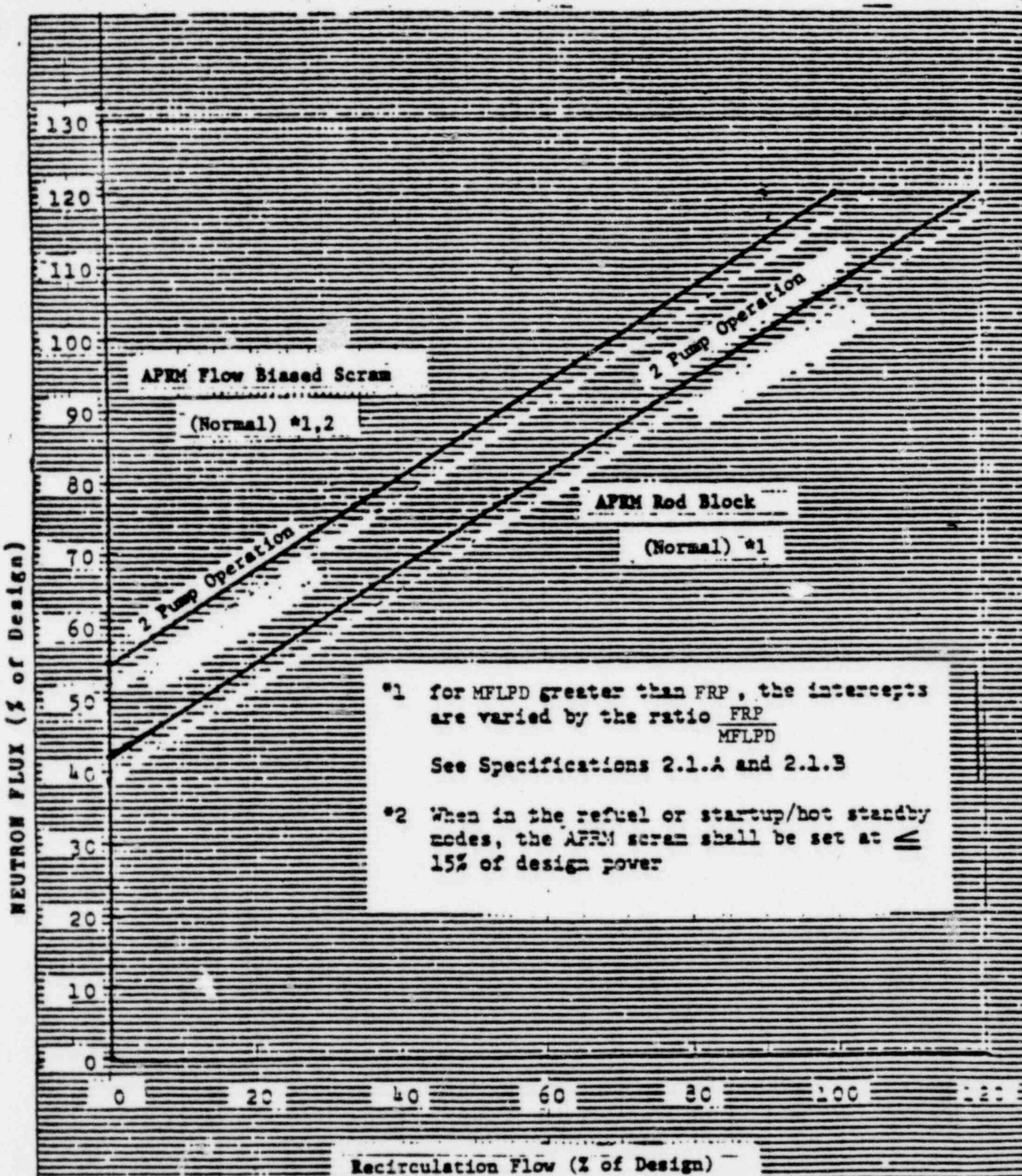

Figure 2.1.1

APRM Scram and Rod Block Trip Limiting Safety System Settings

The required input to the statistical model are the uncertainties listed on Table 5-1, Reference 3, the nominal values of the core parameters listed in Table 5-2, Reference 3, and the relative assembly power distribution shown in Figures 5-1 and 5-1A of Reference 3. Tables 5-2A and 5-2B, Reference 3, show the R-factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The basis for the uncertainties in the core parameters are given in NEDO 20340[2] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Pilgrim Nuclear Power Station Unit 1 during any fuel cycle would not be as severe as the distribution used in the analysis.

B. Core Thermal Power Limit (Reactor Pressure < 800 psig or Core Flow <10% of Rated)

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psig or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is established by other means. This is done by establishing a limiting condition of core thermal power operation with the following basis.

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. 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 the 3.35 MWt bundle power corresponds to a core thermal power of more than 50%. Therefore a core thermal power limit of 25% for reactor pressures below 800 psia, or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1A or 1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closures of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Pilgrim Unit 1 has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient.

D. Reactor Water Level (Shutdown Condition)

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.

References

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

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

3. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P.

(DELETED)

(DELETED)

2.1 BASES:

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 Specification 2.1.A.1 when the MFLPD is greater than the fraction of rated power (FRP).

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 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 either one or 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

2.1 **BASES:**

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.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less 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

2.1 BASES:

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.

C. Reactor Water Low Level Scram Trip Setting (LL1)

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results 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.

D. Turbine Stop Valve Closure Scram Trip Setting

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 ≤ 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.

E. Turbine Control Valve Fast Closure Scram Trip Setting

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.

F. Main Condenser Low Vacuum Scram Trip Setting

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.

3.1 LIMITING CONDITION 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:

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.

4.1 SURVEILLANCE REQUIREMENTS

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. Daily during reactor power operation, the maximum fraction of limiting power density shall be checked and the scram and APRM Rod Block settings given by equations in Specification 2.1.A.1 and 2.1.B shall be calculated if maximum fraction of limiting power density exceeds the fraction of rated power.

REACTOR PROTECTION SYSTEM (SCRAM) . TRUMENTATION REQUIREMENT

Minm. 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	* (14) (15)	(17)	(17)	X	A or B
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥ 2.5 Indicated on Scale	(11)	(11)	X(12)	A or B
2	High Flux (15%)	≤ 15% of Design Power	X	X	(16)	A or B
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 Tank	≤ 39 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	≤ 7X Normal Full Power Background	X	X	X	A or C
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3) (6)	X(3) (6)	X(6)	A or C
2	Turb. Cont. Valve Fast Closure	≥ 150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure	X(4)	X(4)	X(4)	A or D

*APRM high flux scram setpoint $\leq (.65W + 55)\left[\frac{FRP}{MFLPD}\right]$ Two recirc. pump operation

NOTES FOR TABLE 3.1.1 (Cont'd)

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

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

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

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

14. W is percent of drive flow required to produce a rated core flow of 69 Mlb/hr. Trip level setting in percent of design power (1998 MWt).

15. See Section 2.1.A.1.

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

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

4.1 BASES (Cont'd)

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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Control Rod Block Actuation	C. Control Rod Block Actuation
1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.C.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C. System logic shall be functionally tested as indicated in Table 4.2.C.
2. The minimum number of operable instrument channels specified in Table 3.2.C for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any thirty day period.	
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. Steam Air Ejector Off-Gas System	1. Steam Air Ejector Off-Gas System
(a) Except as specified in (b) below, both steam air ejector off-gas system radiation monitors shall be operab'e during reactor power operation. The trip settings for the monitors shall be set at a value not to exceed the equivalent of the stack release limit specified in Specification 3.8.B.1. The time delay setting for closure of the steam air ejector isolation valves shall not exceed 15 minutes.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2.D.
(b) From and after the date that one of the two steam air ejector radiation monitors is made or found to be inoperable,	

EXHIBIT A

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
G. Recirculation Pump Trip/Alternate Rod Insertion Initiation The recirculation pump trip system causes a pump trip on a signal of high reactor pressure or low-low reactor water level when the mode select switch is in the RUN mode. The alternate rod insertion system provides for initiating control rod insertion whenever the mode switch is in the RUN, STARTUP or SHUTDOWN mode. The limiting conditions for operation for the instrumentation are listed in Table 3.2-G.	G. Recirculation Pump Trip/Alternate Rod Insertion Surveillance for instrumentation which initiates Recirculation Pump Trip and Alternate Rod Insertion shall be as specified in Table 4.2-G.

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TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

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

PNPS

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	High Drywell Pressure	$\leq$2.5 psig	1. Initiates Core Spray; LPCI; HPCI. 2. In conjunction with Low-Low Reactor Water Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS). 3. Initiates starting of Diesel Generators.
1	Reactor Low Pressure	400 psig $\pm$ 25	Permissive for Opening Core Spray and LPCI Admission valves.
1	Reactor Low Pressure	$\leq$110 psig	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
1	Reactor Low Pressure	400 psig $\pm$ 25	In conjunction with Low-Low Reactor Water Level initiates Core Spray and LPCI.
2	Reactor Low Pressure	00 psig $\pm$ 25	Prevents actuation of LPCI break detection circuit.

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting [1]	Remarks
2	Startup Transformer Loss of Voltage	0V with 1.1 Sec Time Delay 3094V with 18 Sec Time Delay	1. Trips Startup Transformer to Emergency Bus Breaker. 2. Locks out automatic closure of Startup Transformer to Emergency Bus. 3. Initiates starting of Diesel Generators in conjunction with loss of auxiliary transformer. 4. Prevents simultaneous starting of CSCS components. 5. Starts load shedding logic for Diesel Operation in conjunction with (a) Low Low Reactor Water Level and Low Reactor Pressure or (b) High drywell pressure or (c) Core Standby Cooling System components in service in conjunction with Auxiliary Transformer breaker open.

1. These trip setpoints define the range of trip settings selected from the appropriate relay curve

PNPS

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting [2]	Remarks
2	Startup Transformer Degraded Voltage	3745V ± 2% with 9.2 ± 0.5 sec. time delay	1. Trips Startup Transformer to Emergency Bus Breaker. 2. Locks out automatic closure of Startup Transformer to Emergency Bus. 3. Initiates starting of Diesel Generators in conjunction with loss of auxiliary transformer. 4. Prevents simultaneous starting of CSCS components. 5. Starts load shedding logic for Diesel Operation in conjunction with (a) Low Low Reactor Water Level and Low Reactor Pressure or (b) High drywell pressure or (c) Core Standby Cooling System components in service in conjunction with Auxiliary Transformer breaker open.

2. Settings subject to change after installation and testing of relays.

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TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
1	RHR (LPCI) Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	NA	Monitors availability of power to logic systems and valves.
1	HPCI Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	NA	Monitors availability of power to logic systems.
2	Recirculation Pump A d/p	$\leq$2 psid	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop.
2	Recirculation Pump B d/p	$\leq$2 psid	
2	Recirculation Jet Pump Riser d/p A>B	$0.5<p<1.5$ psid	
1	Core Spray Sparger to Reactor Pressure Vessel d/p	-1($\pm$1.5) psid	Alarm to detect core spray sparger pipe break.

TABLE 3.2.B.1

INSTRUMENTATION THAT MONITORS BUS UNDERVOLTAGE

Minimum # of Operable Instrument Channels Per Trip System	Function	Setting	Remarks
1	Emergency Bus (1) Undervoltage Annunciation	3850 ± 2% with 9.2 ± 0.1 % Second Time Delay	Alerts Operator to possible degraded voltage conditions

NOTE

(1) In the event that the alarm system is determined inoperable, commence logging safety related bus voltage every 1/2 hour until such time as the alarm is restored to operable status.

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TABLE 3.2.C

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum # of Operable Instrument Channels Per Trip Systems (1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	$(0.65W + 42)\left[\frac{FRP}{MFLPD}\right]$ (2)
2	APRM Downscale	2.5 indicated on scale
1 (7)	Rod Block Monitor (Flow Biased)	$(0.65W + 42)\left[\frac{FRP}{MFLPD}\right]$ (2)
1 (7)	Rod Block Monitor Downscale	5/125 of full scale
3	IRM Downscale (3)	5/125 of full scale
3	IRM Detector not in Startup Position	(8)
3	IRM Upscale	$\leq$108/125 of full scale
2 (5)	SRM Detector not in Startup Position	(4)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; If this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W is percent of drive flow required to produce a rated core flow of 69 M lb/hr. Trip level setting is in percent of design power (1998 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is ≥ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when the IRM range switches are on range 8 or above.

7. The trip is bypassed when the reactor power is ≤30%.

8. This function is bypassed when the mode switch is placed in Run.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP AND ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Mode Select Requirements (2)
	High Reactor Dome Pressure	1175 ± 15 PSIG	A/B
	Low-Low Reactor Water Level	≥ 78.5" above the top of the active fuel	A/B

NOTES

1. Minimum number of operable trip systems shall be two.

2. With the number of OPERABLE trip systems less than that required by note (1), restore the inoperable trip system to operable status within 14 days or be in at least cold shutdown within 24 hours with the mode switch positioned in either:

   A. STARTUP, REFUEL, or SHUTDOWN mode for recirculation pump trip system.

   OR

   B. REFUEL mode for alternate rod insertion system.

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TABLE 4.2.B
MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

	Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1)	Reactor Water Level	(1)	Once/3 months	Once/day
2)	Drywell Pressure	(1)	Once/3 months	None
3)	Reactor Pressure	(1)	Once/3 months	None
4)	Auto Sequencing Timers	NA	Once/operating cycle	None
5)	ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6)	a. Loss of Voltage	Monthly	Once/operating cycle	Once/12 hrs.
	b. Degraded Voltage Relays	Monthly	Once/operating cycle	Once/12 hrs.
7)	Trip System Bus Power Monitors	Once/operating cycle	NA	Once/day
8)	Recirculation System d/p	(1)	Once/3 months	Once/day
9)	Core Spary Sparger d/p	NA	Once/operating cycle	Once/day
10)	Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
11)	Steam Line High Temp. (HPCI & RCIC)	(1)	Once/3 months	None
12)	Safeguards Area High Temp.	(1)	Once/3 months	None
13)	HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
14)	HPCI Suction Tank Levels	(1)	Once/3 months	None
15)	Degraded Voltage Alarms	Monthly	Once/Operating Cycle	Once/12 hrs.

Table 4.2-G

Minimum Test and Calibration Frequency for ATWS RPT/ARI Instrumentation

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
1. Reactor High Pressure	(1)	Once/Operating Cycle-Transmitter Once/3 months - Trip unit	Once/day Once/day
2. Reactor Low-Low Water Level	(1)	Once/Operating Cycle-Transmitter Once/3 months - Trip unit	Once/day Once/day

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.

   These instrument channels will be calibrated using simulated electrical signals once every three months.

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 high radiation main steam line tunnel are not included on Table 4.2.A since they are tested on Table 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.

3.2 BASES (Cont'd)

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.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the Safety Limit MCPR.

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 set at 2.5 indicated on scale.

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

3.2 BASES (Cont'd.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip/alternate rod insertion systems are consistent with the "Monticello RPT/ARI" design described in NEDO 25016 (Reference 1) as referenced by the NRC as an acceptable design (Reference 2) for RPT. Reference 1 provides both system descriptions and performance analyses. The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding providing a negative reactivity feedback. High pressure sensors and low water level sensors initiate the trip. The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated in this unlikely postulated event. Requiring the trip to be operable only when in the RUN mode is therefore conservative. The low water level trip function includes a time delay of nine (9) seconds ± one (1) second to avoid increasing the consequences of a postulated LOCA. This delay has an insignificant effect on ATWS consequences.

Alternate rod insertion utilizes the same initiation logic and functions as RPT and provides a diverse means of initiating a reactor scram. ARI uses sensor diverse from the reactor protection system to depressurize the scram pilot air header, which in turn causes all control rods to be inserted.

References

1. NEDO-25016, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant," September 1976.

2. NUREG 0460, Volume 3, December 1978.

4.2 BASES (Cont'd)

instruments of similar design, a testing interval of once every three months has been found adequate.

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 and are a new, improved line of BWR instrumentation. The calibration frequency is once per operating cycle which is consistent with both the equipment capabilities and the requirements for similar equipment used by other reactor vendors. The calibration frequency of the trip units is proposed to be quarterly, the same as other similar protective instrumentation. Likewise, the test frequency is specified at monthly like that of other protective instrumentation. A sensor check is proposed 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.

BASES:

During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's less than the Safety Limit MCPR. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than the Safety Limit MCPR.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested during each cycle as a periodic check against deterioration of the control rod performance.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and suppression pool cooling subsystems.

Objective

To assure the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Subsystems

1. Both core spray subsystems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 below.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the Surveillance Requirements of the core and suppression pool cooling subsystems which are required when the corresponding Limiting Condition for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Subsystem

1. Core Spray Subsystem Testing.

	Item	Frequency
a.	Simulated Automatic Actuation test.	Once/Operating Cycle
b.	Pump Operability	Once/month and Once/cycle from the Alternate Shutdown Panel
c.	Motor Operated Valve Operability	Once/month and Once/cycle from the Alternate Shutdown Panel
d.	Pump flow rate Each pump shall deliver at least 3600 gpm against a system head corresponding to a reactor vessel pressure of 104 psig.	
e.	Core Spray Header Δ p Instrumentation	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE EQUIPMENT
3.5.A Core Spray and LPCI Subsystems (cont'd)	4.5.A Core Spray and LPCI Subsystems (cont'd)
	Check — Once/day Calibrate — Once/3 months Test — Once/3 months
2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.	2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.
3. The LPCI Subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4, 3.5.A.5 and 3.5.F.5.	3. LPCI Subsystem Testing shall be as follows: a. Simulated Automatic Actuation Test — Once/Operating Cycle b. Pump Operability — Once/month and Once/cycle from the Alternate Shutdown Panel c. Motor Operated valve operability — Once/Month and Once/cycle from the Alternate Shutdown Panel d. Pump Flow — Once/3 months Three LPCI pumps shall deliver 14,400 gpm against a system head corresponding to a vessel pressure of 20 psig

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.B Containment Cooling Subsystem	4.5.B Containment Cooling Subsystem
1. Except as specified in 3.5.B.2, 3.5.B.3, and 3.5.F.3 below, both containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Condition.	1. Containment Cooling Subsystem Testing shall be as follows:

	Item	Frequency
a.	Pump & Valve Operability	Once/3 months and Once/cycle from the Alternate Shutdown Station
b.	Pump Capacity Test Each RBCCW pump shall deliver 1700 gpm at 70 ft. TDH. Each SSWS pump shall deliver 2700 gpm at 55 ft. TDH.	After pump maintenance and every 3 months
c.	Air test on drywell and torus headers and nozzles	Once/5 years

3.5.B Containment Cooling Subsystem (Cont'd)

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

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

C. HPCI Subsystem

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

4.5.B Containment Cooling Subsystem (Cont'd)

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

C. HPCI Subsystem

1. HPCI Subsystem testing shall be performed as follows:

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

3.5.C <u>HPCI Subsystem</u> (Cont'd)

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

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

3.5.D <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>

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

4.5.C <u>HPCI Subsystem</u> (Cont'd)

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

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

4.5.D <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>

1. RCIC Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month and Once/cycle from the Alternate Shutdown Station
c.	Motor Operated Valve Operability	Once/month and Once/cycle from the Alternate Shutdown Station

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.D Reactor Core Isolation Cooling (RCIC) Subsystem (Cont'd)	4.5.D Reactor Core Isolation Cooling (RCIC) Subsystem (Cont'd)

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

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

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

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

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

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

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

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

3.5.E Automatic Depressurization System (ADS)

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

4.5.E Automatic Depressurization System (ADS)

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

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

      This test shall also be performed from the Alternate Shutdown Station within the same time frame.

   b. With the reactor at pressure, each relief valve shall be manually opened until a corresponding change in reactor pressure or main turbine bypass valve positions indicate that steam is flowing from the valve

LIMITING CONDITION FOR OPERATION

3.6.A Thermal and Pressurization Limitations (Cont'd)

3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 50°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

6. Thermal-Hydraulic Stability

Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

B. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour, except as specified in 3.6.B.3:

Conductivity...2 umho/cm

Chloride ion...0.1 ppm

SURVEILLANCE REQUIREMENTS

4.6.A Thermal and Pressurization Limitations (Cont'd)

neutron flux specimens shall be removed at the frequency required by 10 CFR50 Appendix H and tested to experimentally verify or adjust the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 3.6.2.

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

B. Coolant Chemistry

1. a. A reactor coolant sample shall be taken at least every 96 hours and analyzed for radioactivity content.

   b. Isotopic analysis of a reactor coolant sample shall be made at least once per month.

2. During startups and at steaming rates less than 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for chloride content.

LIMITING CONDITION FOR OPERATION

3.6.B Coolant Chemistry (Cont'd)

3. For reactor startups and for the first 24 hours after placing the reactor in the power operating condition, the following limits shall not be exceeded.

   Conductivity...10 umho/cm

   Chloride ion...0.1 ppm

4. Except as specified in 3.6.B.3 above, the reactor coolant water shall not exceed the following limits when operating with steaming rates greater than or equal to 100,000 pounds per hour.

   Conductivity...10 umho/cm

   Chloride ion...1.0 ppm

5. If Specification 3.6.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 24 hrs. and Cold Shutdown within the next 8 hours.

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor

SURVEILLANCE REQUIREMENTS

4.6.B Coolant Chemistry (Cont'd)

3. a. With steaming rates of 100,000 pounds per hour or greater, a reactor coolant sample shall be taken at least every 96 hours and analyzed for chloride ion content.

   b. When all continuous conductivity monitors are inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.C Coolant Chemistry (Cont'd)

power operation is permissible only during the succeeding seven days.

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

D. Safety and Relief Valves

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

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

4.6

D. Safety and Relief Valves

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

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.D **Safety and Relief Valves** (Cont'd) pressure shall be below 104 psig within 24 hours.	
E. Jet Pumps 1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	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 P characteristics by more than 10%.
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%.	F. Jet Pump Flow Mismatch Recirculation pump speeds shall be checked and logged at least once per day.
G. Structural Integrity 1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1974	G. Structural Integrity The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with AEC.

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

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

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

BASES:

3.6.B Coolant Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the higher limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so high concentrations of chlorides are not considered harmful during these periods.

In the case of BWR's where no additives are used and where neutral PH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor clean-up system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the clean-up system to re-establish the purity of the reactor coolant. During start-up periods, which are in the category of less than 1% reactor power, conductivity may exceed 2 u mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 u mho/cm (other than short term spikes), samples will be taken to assure that the chloride concentration is less than 0.1 ppm.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meters inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

BASES:

4.6.B Coolant Chemistry

The iodine radioactivity will be monitored by reactor water sample analysis. The total iodine activity would not be expected to change over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is an indication of the trend of the iodine activity in the reactor coolant. Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses to determine major contributors to activity can be performed by a gamma scan.

(DELETED)

Amendment No. 42

3.7.B Standby Gas Treatment System and Control Room With Efficiency Air Filtration System

1. Standby Gas Treatment System

a. Except as specified in 3.7.B.1.c below, both trains of the standby gas treatment system and the diesel generators required for operation of such trains shall be operable at all times when secondary containment integrity is required or the reactor shall be shutdown in 36 hours.

b. (1.) The results of the in-place cold DOP tests on HEPA filters shall show $\geq$99% DOP removal. The results of halogenated hydrocarbon tests on charcoal adsorber banks shall show $\geq$99% halogenated hydrocarbon removal.

(2.) The results of the laboratory carbon sample analysis shall show >95% methyl iodide removal at a velocity within 10% of system design, 0.5 to 1.5 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H. and $\geq$190°F.

c. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, continued reactor operation or fuel handling is permissible only during the succeeding seven days providing that within 2 hours and daily thereafter, all active components of the other standby gas treatment train shall be demonstrated to be operable.

d. Fans shall operate within $\pm$10% of 4000 cfm.

4.7.B Standby Gas Treatment System and Control Room High Efficiency Air Filtration System

1. Standby Gas Treatment System

a (1.) At least once every 18 months, it shall be demonstrated that pressure drop across the combined high efficiency filters and charcoal adsorber banks is less than 8 inches of water at 4000 cfm.

(2.) At least once every 18 months, demonstrate that the inlet heaters on each train are operable and are capable of an output of at least 14 kW. Perform an instrument functional test on the humidistats controlling the heaters.

(3.) The tests and analysis of Specification 3.7.B.1.b.2 shall be performed at least once every 18 months or following painting, fire or chemical release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal adsorbers

(4.) At least once every 18 months, automatic initiation of each branch of the standby gas treatment system shall be demonstrated, with Specification 3.7.B.1.d satisfied.

(5.) Each train of the standby gas treatment system shall be operated for at least 15 minutes per month.

(6.) The tests and analysis of Specification 3.7.B.1.b.(2) shall be performed after every 720 hours of system operation.

3.7.B (Continued)

e. Except as specified in 3.7.B.1.c, both trains of the standby gas treatment system shall be operable during fuel handling operations. If the system is not operable fuel movement shall not be started (any fuel assembly movement in progress may be completed).

4.7.B (Continued)

b. (1.) Inplace cold DOP testing shall be performed on the HEPA filters after each completed or partial replacement of the HEPA filter bank and after any structural maintenance on the HEPA filter system housing which could affect the HEPA filter bank bypass leakage.

(2.) Halogenated hydrocarbon testing shall be performed on the charcoal adsorber bank after each partial or complete replacement of the charcoal adsorber bank or after any structural maintenance on the charcoal adsorber housing which could affect the charcoal adsorber bank bypass leakage.

3.7.B (Continued)

2. Control Room High Efficiency Air Filtration System

a. Except as specified in Specification 3.7.B.2.c below, both trains of the Control Room High Efficiency Air Filtration System used for the processing of inlet air to the control room under accident conditions and the diesel generator(s) required for operation of each train of the system shall be operable whenever secondary containment integrity is required and during fuel handling operations.

b. (1.) The results of the in-place cold DOP tests on HEPA filters shall show $\geq$99% DOP removal. The results of the halogenated hydrocarbon tests on charcoal adsorber banks shall show $\geq$99% halogenated hydrocarbon removal when test results are extrapolated to the initiation of the test.

(2.) The results of the laboratory carbon sample analysis shall show $\geq$95% methyl iodide removal at a velocity within 10% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H., and $\geq$125^{o}F.

c. From and after the date that one train of the Control Room High Efficiency Air Filtration System is made or found to be incapable of supplying filtered air to the control room for any reason, reactor operation or refueling operations are permissible only during the succeeding 7 days. If the system is not made fully operable within 7 days, reactor

4.7.B (Con ued)

2. Control Room High Efficiency Air Filtration System

a. At least once every 18 months the pressure drop across each combined filter train shall be demonstrated to be less than 3 inches of water at 1000 cfm.

b. (1.) The tests and analysis of Specification 3.7.B.2.b shall be performed once every 18 months or following painting, fire or chemical release in any ventilation zone communicating with the system while the system is operating.

(2.) Inplace cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing which could affect the HEPA filter bank bypass leakage.

(3.) Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing which could affect the charcoal adsorber bank bypass leakage.

(4.) Each train shall be operated with the heaters in automatic for at least 15 minutes every month.

(5.) The test and analysis of Specification 3.7.B.2.b.(2) shall be performed after every 720 hours of system operation.

3.7.B (Continued)

shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours and irradiated fuel handling operations shall be terminated within 2 hours. (Fuel handling operations in progress may be completed).

d. Fans shall operate within $\pm$ 10% of 1000 cfm.

4.7.B (Continued)

c. At least once every 18 months the following shall be demonstrated:

(1) Automatic initiation of the control room high efficiency air filtration system.

(2) Operability of heaters at rated power.

3. Perform an instrument functional test on the humidistate controlling the heaters.

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
1	Main Steam Line isolation valves	4	4	$3 \leq T \leq 5$	0	GC
1	Main steam line drain isolation valves	1	1	30	C	SC
1**	Reactor Water sample line isolation valves	1	1	10	C	SC
2	Drywell purge supply isolation valves		2	15	C(1) 0(1)	SC GC
2	Suppression chamber purge supply isolation valves		2	15	0	GC
2	Nitrogen purge isolation valve		1	10	0	GC
2	Nitrogen makeup isolation valve		1	10	C	SC
2	Suppression chamber nitrogen makeup isolation valve		1	10	C	SO
2	Drywell purge exhaust isolation valves		2	15	C	SC
2	Drywell exhaust isolation valves		2	10	C	SC
2	Suppression chamber purge exhaust isolation valves		2	15	C	SC
2	Suppression chamber exhaust isolation valves		2	10	C	SC

NOTES FOR TABLE 3.7.1 (Cont'd)

2. High reactor vessel pressure

3. High drywell pressure

GROUP 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:

1. HPCI steam line high flow

2. High temperature in the vicinity of the HPCI steam line

3. Low reactor pressure

GROUP 5: Isolation valves in the RCIC system are closed upon any one of the following signals:

1. RCIC steam line high flow

2. High temperature in the vicinity of the RCIC steam line

3. Low reactor pressure

GROUP 6: Actions in Group 6 are initiated by any one of the following:

1. Reactor low water level

2. Cleanup area high temperature

3. Cleanup inlet high flow

*The RHRS shutdown cooling injection isolation valves require a Group 2 signal plus high reactor vessel pressure.

**The Reactor Water Sample Line Isolation Valves initiate on a Group 1 signal plus high drywell pressure.

BASES

3.7.B.1 and 4.7.B.1 - Standby Gas Treatment System

The Standby Gas Treatment System is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Upon containment isolation, both standby gas treatment fans are designed to start to bring the reactor building pressure negative so that all leakage should be in leakage. After a preset time delay, the standby fan automatically shuts down so the reactor building pressure is maintained approximately ¼ inch of water negative. Should one system fail to start, the redundant system is designed to start automatically. Each of the two trains has 100% capacity.

High Efficiency Particulate Air (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA filter efficiency of at least 99 percent removal of cold DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of at least 95 percent for expected accident conditions. The specified efficiencies for the charcoal and particulate filters is sufficient to preculde exceeding 10 CFR 100 guidelines for the accidents analyzed. The analysis of the loss of coolant accident assumed a charcoal adsorber efficiency of 95% and TID 14844 fission product source terms, Hence, installing two banks of adsorbers and filters in each train provides adequate margin. A 14 kW heater maintains relative humidity below 70% in order to ensure the efficient removal of methyl iodide on the impregnated charcoal adsorbers. Considering the relative simplicity of the heating circuit, the test frequency of once per 18 months is adequate to demonstrate operability.

Air flow through the filters and charcoal adsorbers for 15 minutes each month assures operability of the system. Since the system heaters are automatically controlled, the air flowing through the filters and adsorbers will be $\leq$ 70% relative humidity and will have the desired drying effect.

Tests of impregnated charcoal identical to that used in the filters indicate that shelf life of five years leads to only minor decreases in methyl iodide removal efficiency. Hence, the frequency of laboratory carbon sample analysis is adequate to demonstrate acceptability. Since adsorbers must be removed to perform this analysis, this frequency also minimizes the system out of service time as a result of surveillance testing. In addition, although the halogenated hydrocarbon testing is basically a leak test, the adsorbers have charcoal of known efficiency and holding capacity for elemental iodine and/or methly iodide, the testing also gives an indication of the relative efficiency of the installed system.

The required Standby Gas Treatment System flow rate is that flow, less than or equal to 4000 CFM which is needed to maintain the Reactor Building at a 0.25 inch of water negative pressure under calm wind conditions. This capability is adequately demonstrated during Secondary Containment Leak Rate Testing performed pursuant to Technical Specification 4.7.C.1.c.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters or adsorbers, thus reducing their reserve capacity too quickly. The filter testing is performed pursuant to appropriate procedures reviewed and approved by the Operations Review Committee pursuant to Section 6 of these Technical Specifications. The in-place testing of charcoal filters is performed by injecting a halogenated hydrocarbon into the system upstream of the charcoal adsorbers. Measurements of the concentration upstream and downstream are made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. A similar procedure substituting dioctyl phthalate for halogenated hydrocarbon is used to test the HEPA filters.

Pressure drop tests across filter and adsorber banks are performed to detect plugging or leak paths through the filter or adsorber media. Considering the relatively short times the fans will be run for test purposes, plugging is unlikely and the test interval of once per 18 months is reasonable.

System drains and housing gasket doors are designed such that any leakage would be inleakage from the Standby Gas Treatment System Room. This ensures that there will be no bypass of process air around the filters or adsorbers.

Only one of the two Standby Gas Treatment Systems (SBGTS) is needed to maintain the secondary containment at a 0.25 inch of water negative pressure upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling activities may continue while repairs are being made. In the event one SBGTS is inoperable, the redundant system will be tested daily. This substantiates the availability of the operable system and justifies continued reactor or refueling operations.

If both trains of SBGTS are inoperable, the plant is brought to a condition where the SBGTS is not required.

BASES

3.7.B.2.b and 4.7.B.2.b - Control Room High Efficiency Air Filtration System

The Control Room High Efficiency Air Filtration System is designed to filter intake air for the control room atmosphere during conditions when normal intake air may be contaminated. Following manual initiation, the Control Room High Efficiency Air Filtration System is designed to position dampers and start fans which divert the normal air flow through charcoal adsorbers before it reaches the control room.

High Efficiency Particulate Air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. A second bank of HEPA filters is installed downstream of the charcoal filter.
The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of cold DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Tests of impregnated charcoal identical to that used in the filters indicate that shelf life of five years leads to only minor decreases in methyl iodide removal efficiency. Hence, the frequency of laboratory carbon sample analysis is adequate to demonstrate acceptability. Since adsorbers must be removed to perform this analysis, this frequency also minimizes the system out of service time as a result of surveillance testing. In addition, although the halogenated hydrocarbon testing is basically a leak test, the adsorbers have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the testing also gives an indication of the relative efficiency of the installed system.

Determination of the system pressure drop once per operating cycle provides indication that the HEPA filters and charcoal adsorbers are not clogged by excessive amounts of foreign matter and that no bypass routes through the filters or adsorbers had developed. Considering the relatively short times the systems will be operated for test purposes, plugging is unlikely and the test interval of once per operating cycle is reasonable.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters or adsorbers, thus reducing their reserve capacity too quickly. The filter testing is performed pursuant to appropriate procedures reviewed and approved by the Operations Review Committee pursuant to Section 6 of these Technical Specifications. The in-place testing of charcoal filters is performed by injecting a halogenated hydrocarbon into the system upstream of the charcoal adsorbers. Measurements of the concentration upstream and downstream are made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. A similar procedure substituting dioctyl phthalate for halogenated hydrocarbon is used to test the HEPA filters.

If both trains of the system are found to be inoperable, there is no immediate threat to the control room and reactor operation or fuel handling may continue for a limited period of time while repairs are being made. If at least one train of the system cannot be repaired within seven days, the reactor will be brought to a condition where the Control Room High Efficiency Air Filtration System is not required.

Air flow through the filters and charcoal adsorbers for 15 minutes each month assures operability of the system. Since the system heaters are automatically controlled, the air flowing through the filters and adsorbers will be $\leq$ 70% relative humidity and will have the desired drying effect.

3.7.C - Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water negative pressure within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

3.8 RADIOACTIVE MATERIALS

Applicability:

Applies to the controlled release of radioactive liquids and gases from the facility.

Objective:

To define the limits and conditions for the release of radioactive effluents to the environs to assure that any radioactive releases are as low as practicable and would not result in radiation exposures greater than a few percent of natural background exposures and, in any event, within the limits of 10 CFR Part 20 for instantaneous release rates.

Specification:

A. Liquid Effluents

1. The instantaneous gross radioactivity release concentration in liquid effluents from the station shall not exceed the values specified in 10 CFR Part 20, Appendix B, for unrestricted areas.

2. The release rate of radioactive liquid effluents, excluding tritium and noble gases, shall not exceed 10 curies during any calendar quarter, without specific approval from the Commission.

4.8 RADIOACTIVE MATERIALS

Applicability:

Applies to the periodic test and record requirements and sampling and monitoring methods used for facilities effluents.

Objective:

To ensure that radioactive liquid and gaseous releases from the facility are maintained within the limits specified by Specifications 3.8.A and 3.8.B.

Specification:

A. Liquid Effluents

1. Facility records shall be maintained of the radioactive concentrations and volume before dilution of each batch of liquid effluent released, and of the average dilution flow and length of time over which each discharge occurred.

2. Prior to release of each batch of liquid effluent, a sample shall be taken from that batch and analyzed for gross radioactivity (B, γ) and a complete gamma spectrum analysis to demonstrate compliance with 3.8.A. using the circulating water flow rate at the time of discharge.

3.8.A **Liquid Effluents** (Cont'd)

3. The annual average concentration of tritium prior to dilution in a natural body of water shall not exceed 1×10^{-5} uCi/cc.

4. During release of radioactive wastes, the following conditions shall be met:

   a. The minimum dilution water required to satisfy 3.8.A.1 shall be met.

   b. The gross activity monitor and recorder on the radwaste effluent line shall be operable.

   c. The effluent control monitor shall be set to alarm and automatically close the waste discharge valve prior to exceeding the limits specified in 3.8.A.1 above.

   d. Liquid waste activity and flow rate shall be continuously monitored and recorded during release.

5. The equipment installed in the liquid radioactive waste system shall be maintained and shall be operated to process, as a minimum, all liquids prior to their discharge when the activity released during any calandar quarter exceeds 1.25 curies.

6. When the release rate of radioactive liquid effluents, excluding tritium and noble gases, exceed 2.5 curies during any calendar quarter, the licensee shall notify the Director, Directorate of Licensing within 30 days identifying the causes and describing the proposed program of action to reduce such release rates.

4.8.A **Liquid Effluents** (Cont'd)

3.

4. A monthly proportional composite liquid waste sample, including an aliquot of each batch released during the month, shall be analyzed for tritium, Sr-89, Sr-90, and gross alpha radioactivity.

5. At least one representative liquid waste batch per month shall be analyzed for dissolved fission and activated gases.

6.

7. The liquid effluent radiation monitor shall be calibrated at least quarterly by means of a check source and annually with a known radioactive source. Each monitor, as described, shall also have an instrument channel test monthly and a sensor check daily.

8. The status and performance of automatic isolation valves and discharge tank selection valves and results of independent liquid waste samples shall be checked and logged.

3.8.B Airborne Effluents (Cont'd)

When the release rate exceeds 0.05 Ci/sec for a period of greater than 48 hours, the licensee shall notify the Director, Directorate of Licensing, in writing within 10 days.

4. During release of gaseous wastes, the following conditions shall be met:

   a. The gross activity monitor, the iodine activity monitor and particulate activity monitor shall be operable.

   b. The minimum air flow shall be maintained.

   c. Isolation devices capable of limiting gaseous release rates from the main stack to within the values specified in 3.8.B.1 above shall be operable.

5. One reactor building exhaust vent and one plant stack monitoring system shall be operable, and the off-gas radiation monitors shall be operable or operating whenever steam pressure is available to the air ejectors. If these requirements are not satisfied, a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the hot shutdown condition within 10 hours in the case of the stack monitor or 10 days in the case of the building vent monitor.

6. The containment shall not be purged except through the standby gas treatment system while the reactor is in the RUN mode.

4.8.B Airborne Effluents (Cont'd)

   a. Within one month after the date of initial criticality of the reactor.

   b. At least monthly thereafter.

   c. Following each refueling, process change or other occurrence which could alter the mixture of radionuclides.

4. The release rate of tritium in the gaseous effluents shall be determined on the basis of a representative sample collected and analyzed for tritium at least quarterly.

5. Samples of offgas effluents shall be taken at least every 96 hours and a ratio of long-lived to short-lived radioactivity determined. When these samples indicate a change in this ratio of greater than 20% from the ratio established by the previous monthly isotopic analysis, a new isotopic analysis shall be performed.

6. Facility records of iodine and particulate releases with half lives greater than eight days shall be maintained on the basis of all filter cartridges counted. These filters shall be analyzed for I-131 (charcoal), gross radioactivity (β,γ) and a complete gamma spectrum (particulate). These filters shall be analyzed weekly when the iodine or particulate release rate is less

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.B Airborne Effluents (Cont'd)	4.8.B Airborne Effluents (Cont'd)

than the annual average release rate given in 3.8.B.2 above, otherwise the cartridges shall be removed and analyzed daily until a steady release level has been established.

7. A weekly charcoal filter from each release point shall be analyzed for I-133 and I-135 at least quarterly.

8. A weekly particulate filter from each release point shall be analyzed for gross alpha radioactivity at least quarterly. A composite of a months' filters from each release point shall be analyzed for Sr-89 and Sr-90 at least quarterly.

9. When the average daily gross radioactivity release rate equals or exceeds that given in 3.8.B.3 or increases by 50% over the previous day, the iodine and particulate cartridge shall be analyzed to determine the release rate increase for iodines and particulates.

10. All waste gas monitors shall be calibrated at least quarterly by means of a built-in check source and annually with a known radioactive source. Each monitor shall have an instrument channel test at least monthly and sensor check at least daily.

11. At least annually, automatic initiation and closure of waste gas system shall be verified.

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

The reactor shall not be made critical unless all of the following conditions are satisfied:

1. At least one offsite transmission line and the startup transformer are available and capable of automatically supplying auxiliary power to the emergency buses.

2. An additional source of offsite power consisting of one of the following:

 a. A transmission line and shutdown transformer capable of supplying power to the emergency 4160 volt buses.

 b. The main transformer and unit auxiliary transformer available and capable of supplying power to the emergency 4160 volt buses.

3. Both diesel generators shall be operable. Each diesel generator shall have a minimum of 19,800 gallons of diesel fuel on site.

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment Surveillance

1. Diesel Generators

 a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one hour period at rated load.

 During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.

 Also, once per operating cycle the diesel generator shall be manually started and loaded from the Alternate Shutdown Station

 b. Once per operating cycle the condition under which the diesel generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.

1. **Verifying de-energization of the emergency buses and load shedding from the emergency buses.**

2. **Verifying the diesel starts from ambient condition on the auto-start signal energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequence and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.**

The results shall be logged.

C. **Once per operating cycle with the diesel loaded per 4.9.A.1.b verify that on diesel generator trip secondary (off-site) a-c power is automatically connected to the emergency service buses and emergency loads are energized through the load sequencer in the same manner as described in 4.9.A.1.b.1.**

The results shall be logged.

2. Secondary Off-Site Power

A. A test will be performed once per operating cycle to verify that the shutdown transformer breakers will close on to the safety related buses within 12 to 14 seconds.

3.9.A Auxiliary Electrical Equipment

4. 4160 volt buses A5 and A6 are energized and the associated 480 volt buses are energized.

5. The station and switchyard 125 and 250 volt batteries are operable. Each battery shall have an operable battery charger.

6. Emergency Bus under Voltage Annunciation System is operable.

   Emergency buses A5 & A6 shall not be operated below 3745 during normal operation.

B. Operation with Inoperable Equipment

Whenever the reactor is in Run Mode or startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.B.4 and 3.9.B.5.

1. From and after the date that incoming power is not availble from the startup or shutdown transformer, continued

4.9.A Auxiliary Electrical Equipment Surveillance

c. Once a month the quantity of diesel fuel available shall be logged.

d. Once a month a sample of diesel fuel shall be checked for quality in accordance with ASTM D270-1975. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-77 and logged.

2. Station and Switchyard Batteries

a. Every week the specific gravity, the voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

c. Once each operating cycle, the stated batteries shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be determined after the discharge and logged.

3. Emergency Under Voltage Annunciation System

a. Once each operating cycle, calibrate the alarm sensor.

b. Once each 31 days perform a channel functional test on the alarm system.

c. In the event the alarm system is determined inoperable under 3.b above, commence logging safety related bus voltage every 30 minutes until such time as the alarm is restored to operable status.

3.9.B Operation with Inoperable Equipment

following conditions are satisfied and the AEC is notified within 24 hours of the occurrence and the plans for restoration of the inoperable components:

a. The startup transormer and both offsite 345 kV transmission lines are available and capable of automatically supplying auxiliary power to the emergency 4160 volt buses.

b. A transmission line and associated shutdown transformer are available and capable of automatically supplying auxiliary power to the emergency 4160 volt buses.

5. From and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding three days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specification 3.5.F is satisfied. The AEC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

6. With emergency bus voltage below 3745 during normal operation, transfer the safety related buses to the diesel generators and be in at least Hot shutdown within the next 4 hours and in Cold shutdown within the following 12 hours.

BASES (Continued)

3.9

can be used for either 125 volt battery, (2) a 250 volt d-c back-up battery charger is supplied. Thus, on loss of normal battery charger, the back-up charger can be used. The 125 volt battery system shall have a minimum of 105 volts at the battery terminals to be considered operable. The 250 volt battery system shall have a minimum of 210 volts at the battery terminals to be considered operable.

Automatic second level undervoltage protection is installed on the startup transformer and is available when safety related loads are being supplied from this source. During normal operation, the unit auxiliary transformer supplies safety related buses. Automatic second level undervoltage protection is not installed on the unit auxiliary transformer. Safety related loads are in use during normal operation and thus are not provided with automatic second level undervoltage protection. BECo is conducting new grid studies with the intent of providing automatic second level undervoltage protection.* We expect these studies to be completed and any necessary modifications to be installed prior to startup from reload 5. During Cycle 5, the Safety Bus Degraded Voltage Alarm System will be relied upon in conjunction with operator action, to preclude operation with a degraded bus voltage condition.

* for the unit auxiliary transformer and the startup transformer.

BASES:

4.9

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel generator will be manually started, connected to the bus and load picked up. The diesel generator should be loaded to a least 75% of rated load to prevent fouling of the engine. It is expected that the diesel generator will be run for one to two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required.

Each diesel generator has one air compressor and two air receivers for starting, one air compressor and three receiver tanks for turbo-charger assist in starting and loading. It is expected that the air compressors will run only infrequently. During the monthly check of the diesel generator, one receiver in each set of receivers will be drawn down below the point at which the corresponding compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

The diesel generator fuel consumption rate at full load is approximately 193 gallons per hour. Thus, the monthly load test of the diesel generators will test the operation and ability of the fuel oil transfer pumps to refill the day tank and will check the operation of these pumps from the emergency source.

The test of the diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e., it will check diesel generator starting and closure of diesel generator breaker and sequencing of load on the diesel generator. The diesel generator will be started by simulation of a loss-of-coolant accident. In conjunction with this an undervoltage condition will be imposed to simulate a loss of off-site power. The timing sequence will be checked to assure that the diesel generators can operate the core spray pumps at rated power within thirty seconds and the LPCI pumps at rated power within forty-three seconds. Additionally, with the Diesel Generator operating as described above the capability of supplying power the the Emergency Bus will be further substantiated. This will be accomplished by tripping the Diesel Generator Breaker and verifying that secondary offsite power is connected to the Emergency Bus and emergency loads are energized through the load sequencer.

Periodic tests between refueling outages verify the ability of the diesel generator to run at full load and the core and containment cooling pumps to deliver full flow. Periodic testing of the various components, plus a functional test once-a-cycle, is sufficient to maintain adequate reliability.

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure. In addition, the checks described also provide adequate indication that the batteries have the specified ampere hour capability.

3.11 REACTOR FUEL ASSEMBLY

Applicability

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

Objective

The Objective of the 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 shown in Figures 3.11-1 through 3.11-5. The top curves are applicable for core flow greater than or equal to 90% of rated core flow. When core flow is less than 90% of rated core flow, the lower curves shall be limiting. 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 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	SURVEILLANCE REQUIREMENTS
B. Linear Heat Generation Rate (LHGR) During reactor power operation the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 kw/ft for 8x8 and P8x8R fuel. 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.	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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. Minimum Critical Power Ratio (MCPR)	C. Minimum Critical Power Ratio (MCPR)
During power operation MCPR shall be $\geq$ 1.35 for 8x8 and P8x8R fuel. 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.	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.
For core flows other than rated the MCPR shall be $\geq$ 1.35 for 8x8 and P8x8R fuel times K_f, where K_f is as shown in Figure 3.11-8.	
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 3.11-8.	

BASES

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

This specifications assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The limiting value for APLHGR is this LHGR of the highest powered rod divided by its local peaking factor.

The calculational procedure used to establish the APLHGR limit for each fuel type is based on a loss-of-coolant accident analysis. The emergency core cooling system (ECCS) evaluation models which are employed to determine the effects of the loss of coolant accident (LOCA) in accordance with 10CFR50 and Appendix K are discussed in Reference 1. The models are identified as LAMB, SCAT, SAFE, REFLOOD, and CHASTE. The LAMB Code calculates the short term blowdown response and core flow, which are input into the SCAT code to calculate blowdown heat transfer coefficients. The SAFE code is used to determine longer term system response and flows from the various ECC systems. Where appropriate, the output of SAFE is used in the REFLOOD code to calculate liquid levels. The results of these codes are used in the CHASTE code to calculate fuel clad temperatures and maximum average planar linear heat generation rates (MAPLHGR) for each fuel type.

The significant plant input parameters are given in Reference 2. MAPLHGR's for the present fuel types were calculated by the above procedure and are included in Reference 3. The curves in Figures 3.11-1 through 3.11-5 were generated by multiplying the values in Reference 3 by factors given in Reference 4. These multipliers were developed assuming no core spray heat transfer credit in the LOCA analysis.

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REFERENCES

1. General Electric BWR Generic Reload Fuel Application, NEDE-24011-P.

2. Loss of Coolant Accident Analysis Report for Pilgrim Nuclear Power Station, NEDO-21696, August 1977.

3. "Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4", NEDO-24224, November 1979.

4. "Supplement 1 to Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4" NEDO-24224-1 March 1980.

BASES:

3.11C MINIMUM CRITICAL POWER RATIO (MCPR)

Operating Limit MCPR

For any abnormal operating transient analysis evaluation with the initial condition of the reactor being 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 instrument trip setting given in Specification 2.1. The required operating limit MCPR at steady state conditions in Specification 3.11.C was chosen conservatively at a value higher than MCPR's of past analysis with the objective of establishing an operating limit MCPR which is fuel type and cycle independant.

The difference between the specified Operating Limit MCPR in Specification 3.11C and the Safety Limit MCPR in Specification 1.1A defines the largest reduction in critical power ratio (CPR) permitted during any anticipated abnormal operating transient. To ensure that this reduction is not exceeded, the most limiting transients are analized for each reload and fuel type (8x8 and P8x8R) to determine that transient which yields the largest value of Δ CPR. This value, when added to the Safety Limit MCPR must be less than the minimum operating limit MCPR's of Specification 3.11.C. The result of this evaluation is documented in the "Supplemental Reload Licensing Submittal" for the current reload.

The evaluation of a given transient begins with the system input parameters shown in Tables 5-4, 5-6 and 5-8 of NEDE-24011-P(1), Supplemented by reload unique inputs given in the current Supplemental Reload Licensing Submittal. These values are input to a GE core dynamic behavior transient computer program described in NEDO-10802(2). Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new mthod of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566(3). The principal result of this evaluation is the reduction in MCPR caused by the transient.

Two codes are used to analyze the rod withdrawal error transient, The first code simulates the three dimensional BWR core nuclear and thermal-hydraulic characteristics. Using this code a limiting control rod pattern is determined; the following assumptions are included in this determination:

(1) The core is operating at full power in the xenon-free condition.

(2) The highest worth control rod is assumed to be fully inserted.

(3) The analysis is performed for the most reactive point in the cycle.

(4) The control rods are assumed to be the worst possible pattern without exceeding thermal limits.

(5) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the maximum allowable linear heat generation rate.

(6) A bundle in the vicinity of the highest worth control rod is assumed to be operating the minimum allowable critical power ratio.

The three-dimensional BWR code then simulates the core response to the control rod withdrawal error. The second code calculates the Rod Block Monitor response to the rod withdrawal error. This code simulates the Rod Block Monitor under selected failure conditions (LPRM) for the core rod use (calculated by the 3-dimensional BWR simulation code) for the control rod withdrawal.

The analysis of the rod withdrawal error for Pilgrim Unit 1 considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor. A summary of the analytical methods used to determine the nuclear characteristics is given in Section 5.2.1.5 of NEDE-24011-P.

MCPR LIMITS FOR CORE FLOWS OTHER THAN RATED

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR given in Specification 3.11C will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.11-8[4] were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-8[4] are conservative for the Pilgrim Unit 1 operation because the operating limit MCPR given in Specification 3.11C is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

4.11.C MINIMUM CRITICAL POWER RATIO (MCPR) - SURVEILLANCE REQUIREMENT

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. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. 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.

REFERENCES

1. General Electric BWR Generic Reload Fuel Application, NEDE-24011-P.
2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).
3. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.
4. Letter from J. E. Howard, Boston Edison Company to D. L. Ziemann USNRC, dated October 31, 1975.

(DELETED)

FIGURE 3.11-1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE 8DB219L

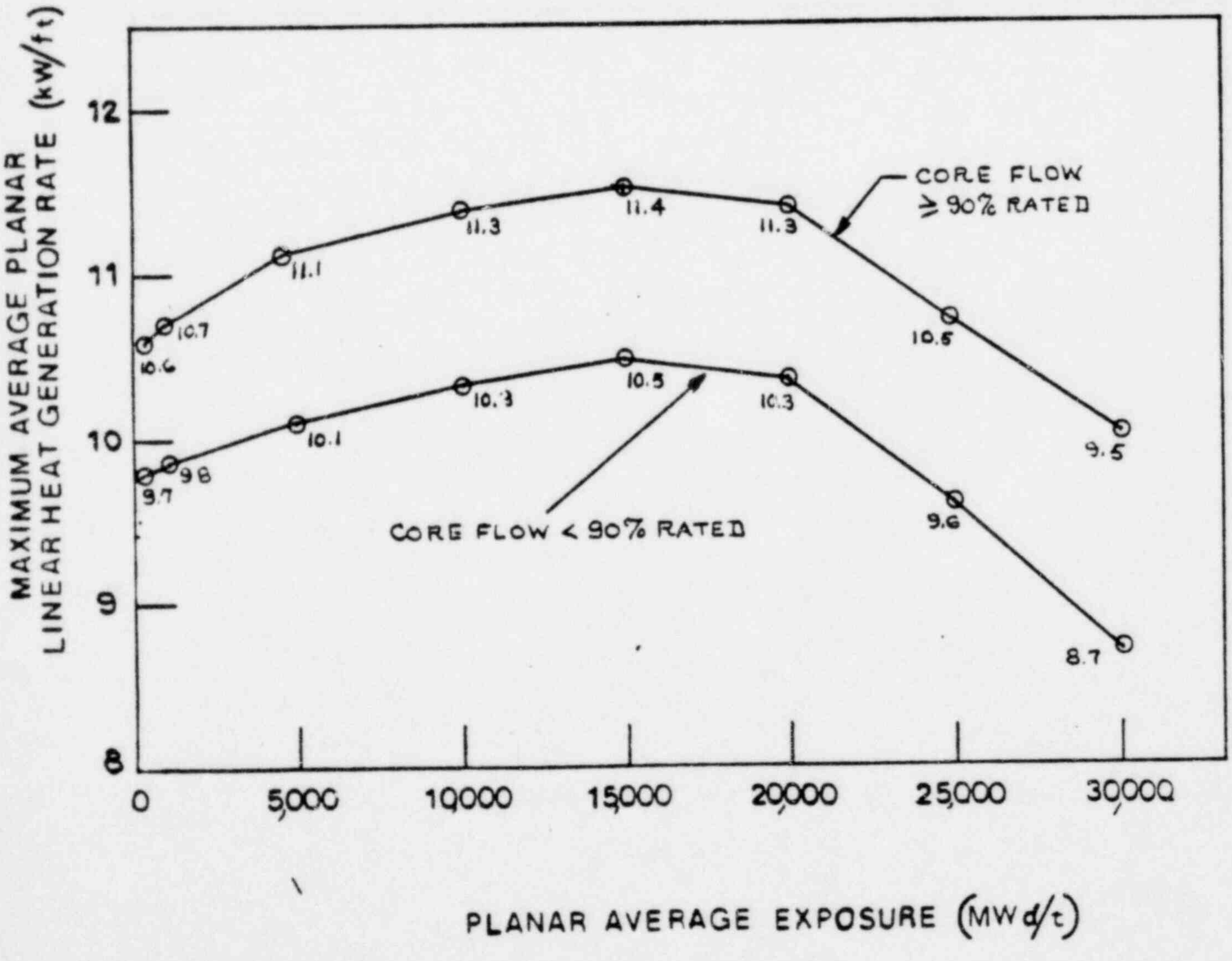


FIGURE 3.11-2

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE 8DB219H

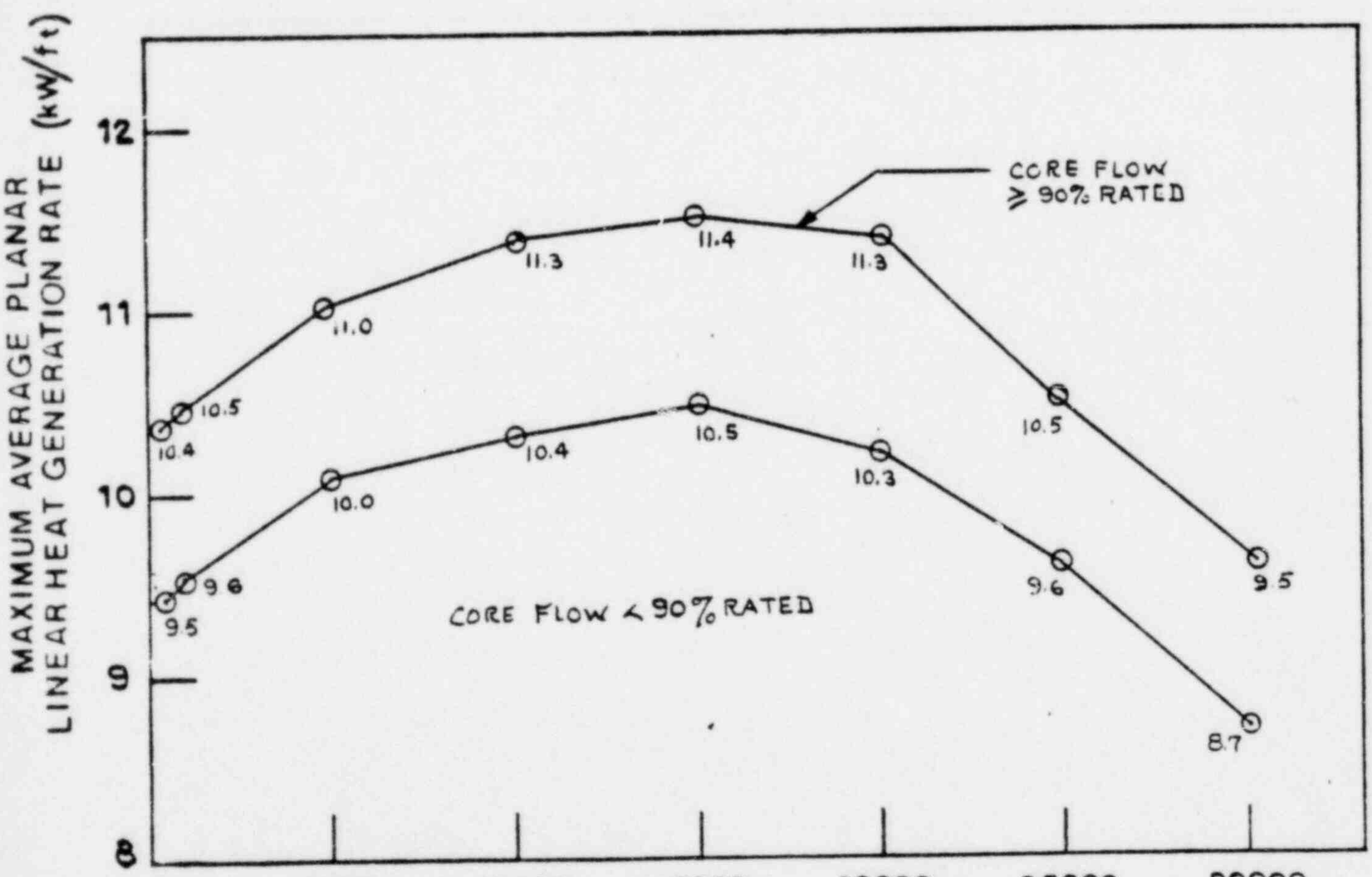


FIGURE 3.11-3
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE
VERSUS
PLANAR AVERAGE EXPOSURE
FUEL TYPE 8DB262

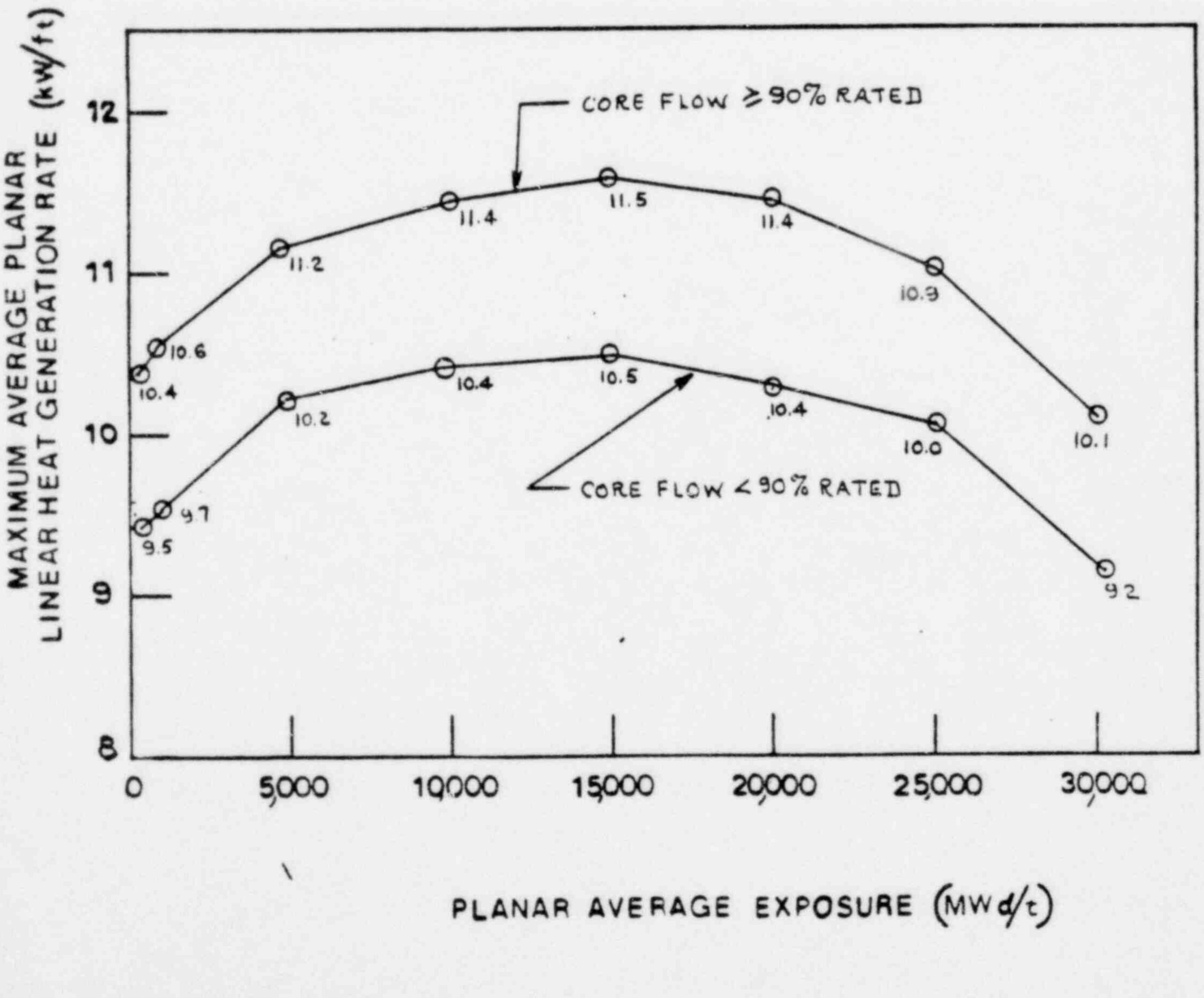


FIGURE 3.11-4

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE P8DRB265L

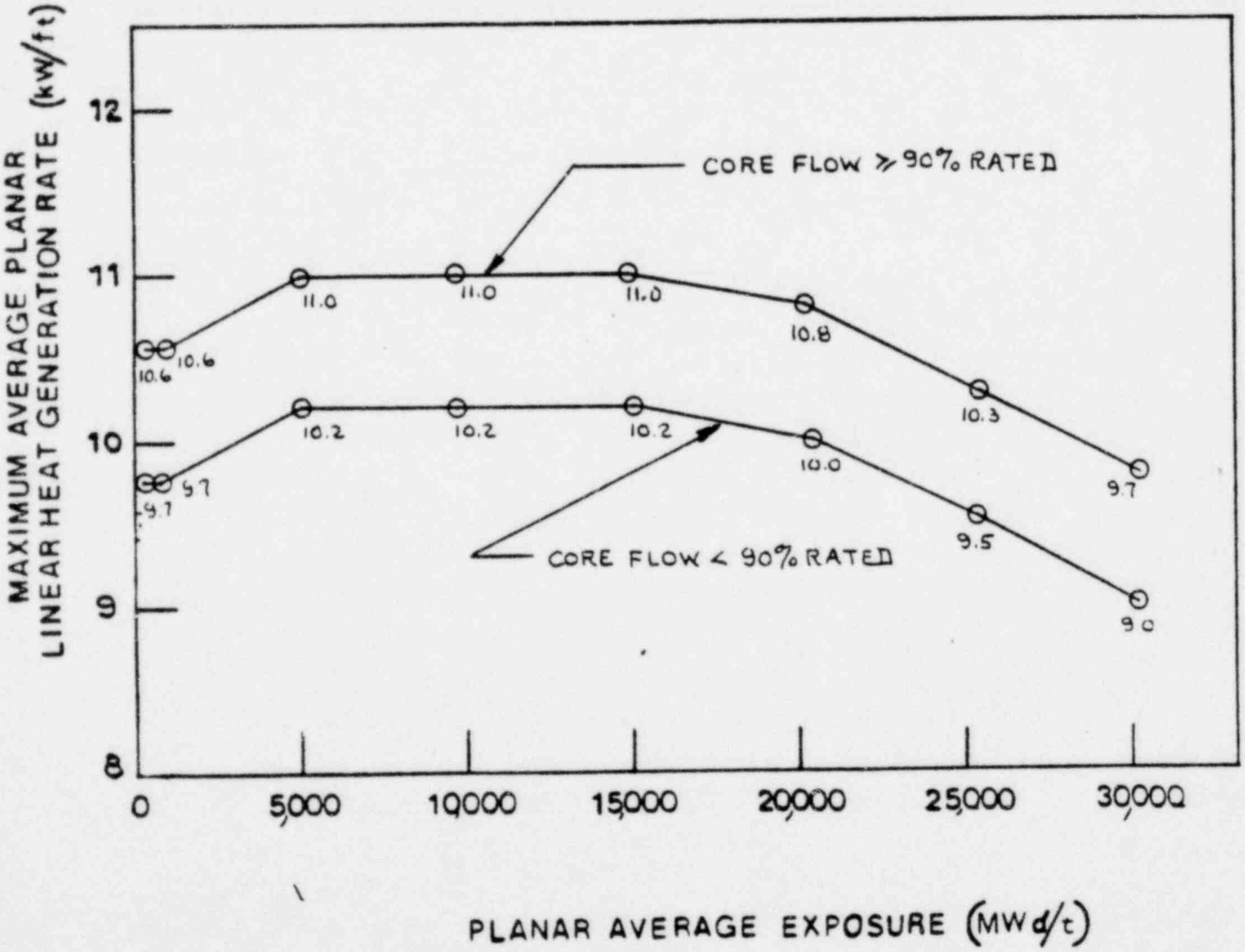


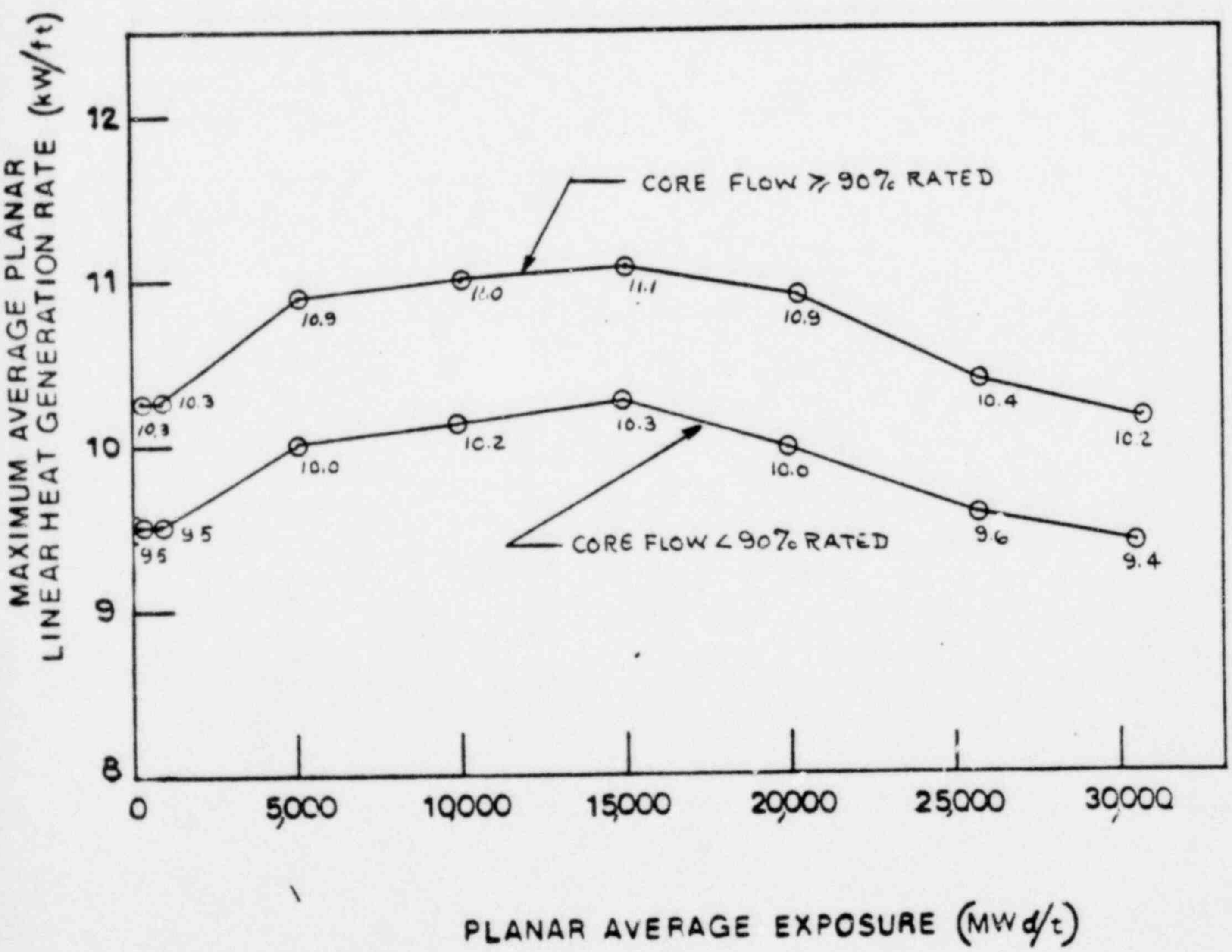
FIGURE 3.11-5

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE

VERSUS

PLANAR AVERAGE EXPOSURE

FUEL TYPE P8DRB282



Amendment No. 42

(DELETED)

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

A. The core shall consist of not more than 580 fuel assemblies of 8x8 (63 fuel rods) and P8x8R (62 fuel rods).

B. The reactor core shall contain 145 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

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.