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

THE YANKEE ATOMIC ELECTRIC COMPANY

DOCKET NO. 50-29

YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. DPR-3

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

   A. The application for amendment by the Yankee Atomic Electric Company (the licensee) dated March 26, 1981, as supplemented May 27, 1981 and July 8, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

   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 2.C(2) of Facility Operating License No. DPR-3 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 69, 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

Richard [illegible] for

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 22, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-3

DOCKET NO. 50-29

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by Amendment Number and contain vertical lines indicating the area of change.

DELETE	INSERT
IV	IV*
VI	VI
1-5	1-5**
1-6	1-6
2-1	2-1**
2-2	2-2
2-3	2-3
2-4	2-4**
2-5	2-5**
2-6	2-6
B2-3	B2-3**
B2-4	B2-4
B2-5	B2-5
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-29	3/4 1-29
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8**
3/4 3-1	3/4 3-1**
3/4 3-2	3/4 3-2
3/4 3-3	3/4 3-3
3/4 3-4	3/4 3-4**
3/4 3-7	3/4 3-7**
3/4 3-8	3/4 3-8
3/4 3-9	3/4 3-9
3/4 3-10	3/4 3-10**
3/4 3-12A	3/4 3-12A
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-15A	3/4 3-15A

* This is included to merely correct editorial errors.

** Overleaf page included for completeness of records.

DELETE	INSERT
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6
3/4 6-11	3/4 6-11**
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13**
3/4 6-14	3/4 6-14
--	3/4 6-14a (new)
3/4 7-5	3/4 7-5
--	3/4 7-5a (new)
3/4 7-9	3/4 7-9
B3/4 1-1	B3/4 1-1
B3/4 1-2	B3/4 1-2**
--	B3/4 2-4
--	B3/4 7-1**
B3/4 7-2	B3/4 7-2
B3/4 7-3	B3/4 7-3
5-1	5-1
5-2	5-2**
5-3	5-3**
5-4	5-4

** Overleaf page included for completeness of records.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

*With 3/4 4-5a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95% of the total non-iodine activity in the coolant.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 2\%$	$\geq 330^{\circ}F$
2. STARTUP	≥ 0.99	$\leq 2\%$	$\geq 330^{\circ}F$
3. HOT STANDBY	< 0.99	0	$\geq 330^{\circ}F$
4. HOT SHUTDOWN	< 0.96	0	$330^{\circ}F > T_{avg} > 200^{\circ}F$
5. COLD SHUTDOWN	< 0.96	0	$\leq 200^{\circ}F$
6. REFUELING**	≤ 0.95	0	$\leq 140^{\circ}F$

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, Main Coolant System pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for 4 and 3 loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop cold leg temperature and THERMAL POWER has exceeded (is above and to the right of) the appropriate Main Coolant System pressure line, be in HOT STANDBY within 1 hour.

MAIN COOLANT SYSTEM PRESSURE

2.1.2 The Main Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

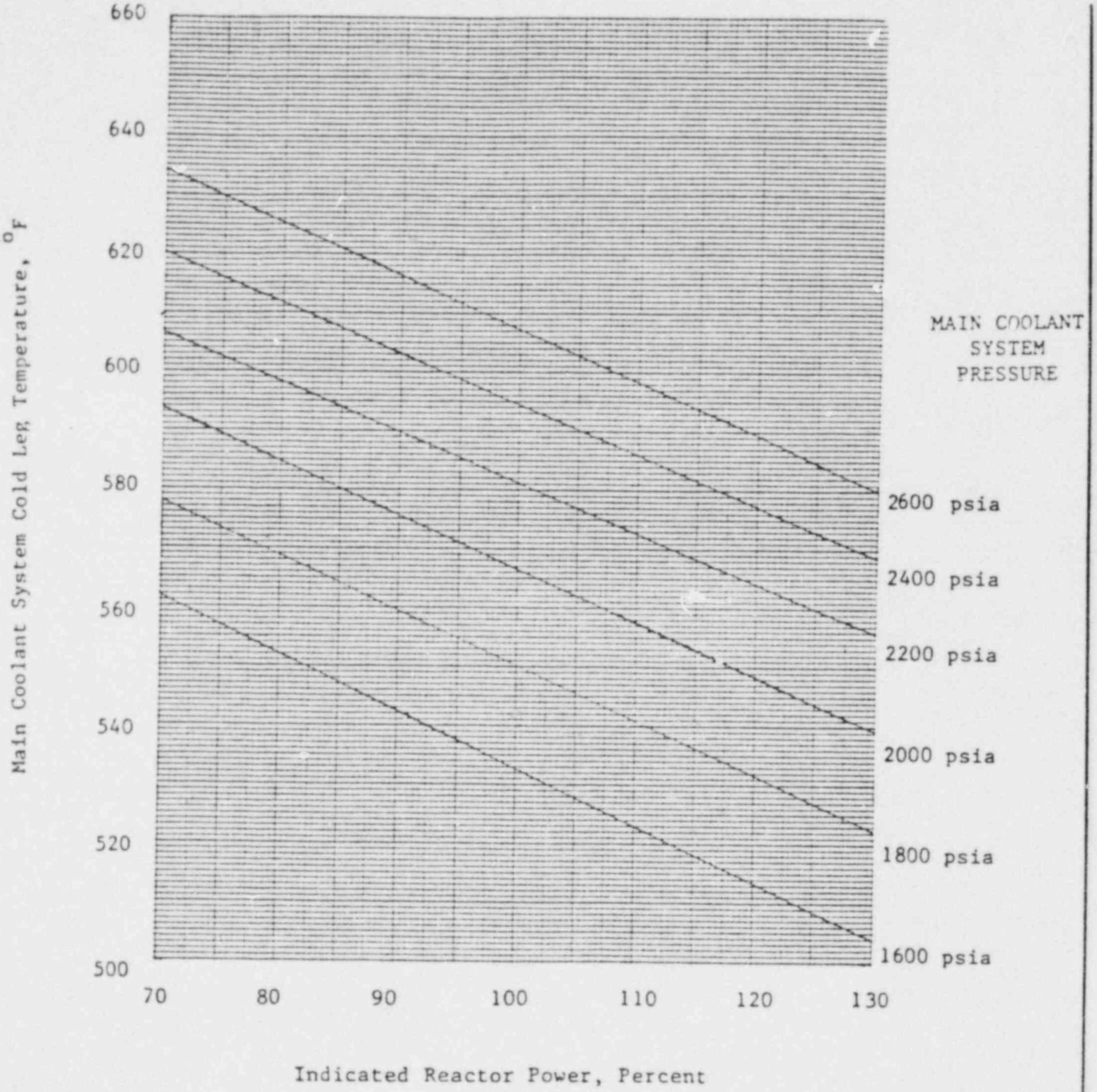
ACTION:

MODES 1 and 2

Whenever the Main Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Main Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

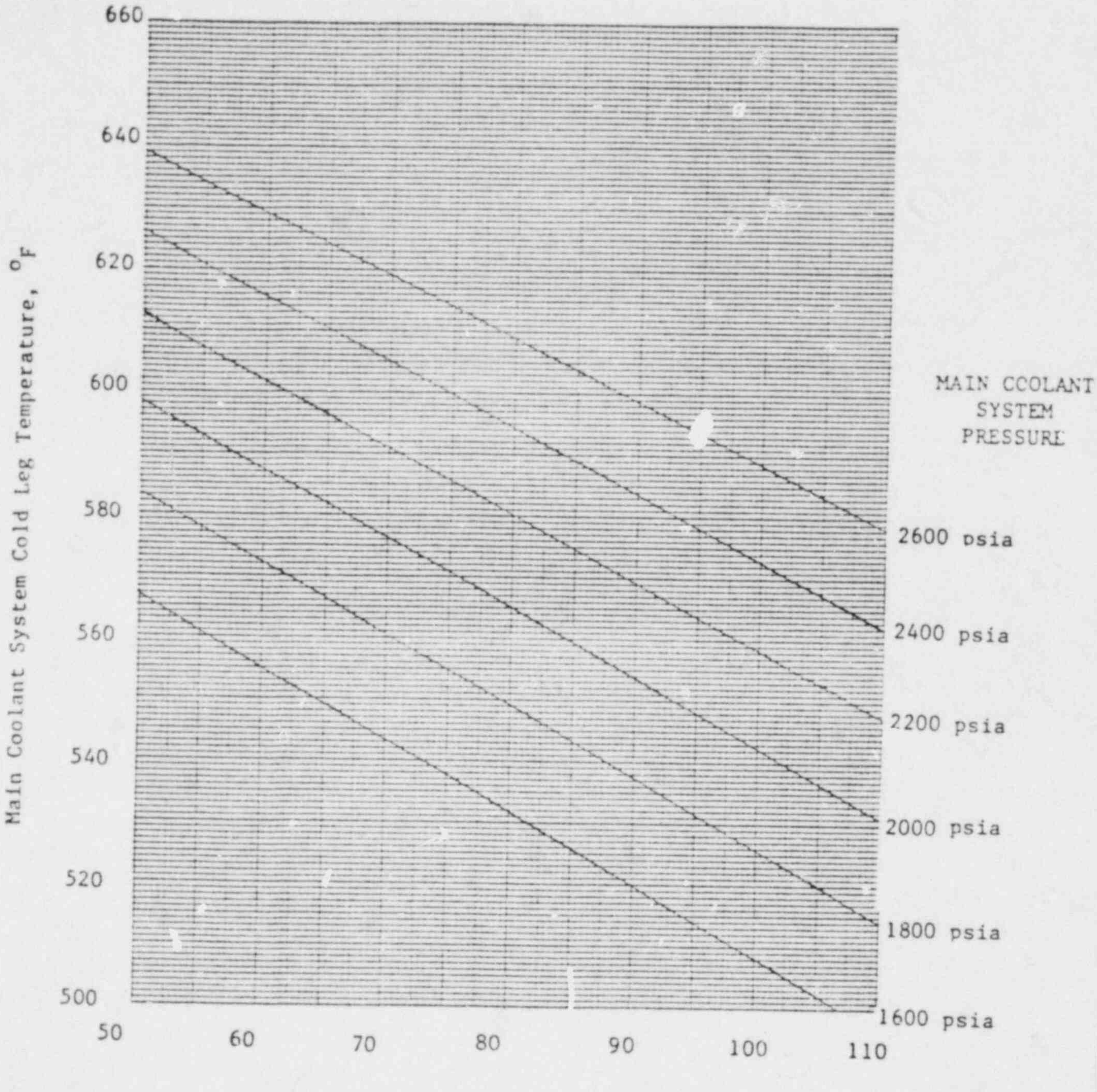
Whenever the Main Coolant System pressure has exceeded 2735 psig, reduce the Main Coolant System pressure to within its limit within 5 minutes.



REACTOR CORE SAFETY LIMIT - ALL LOOPS IN OPERATION

FIGURE 2.1-1

Amendment No. 43, 69



REACTOR CORE SAFETY LIMIT - 3 LOOPS IN OPERATION

FIGURE 2.1-2

Amendment No. 43, 69

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protective system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective system instrumentation trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. Manual Reactor Trip	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 35% of RATED THERMAL POWER High Setpoint - $\leq$ 108% of RATED THERMAL POWER with 4 main coolant pumps operating High Setpoint - $\leq$ 81% of RATED THERMAL POWER with 3 main coolant pumps operating
3. Intermediate Power Range, Neutron Flux	High Setpoint - $\leq$ 108 [illegible] of RATED THERMAL POWER with 4 main coolant pumps operating High Setpoint - $\leq$ 81% of RATED THERMAL POWER with 3 main coolant pumps operating
4. Intermediate Range, High Startup Rate	$\leq$ 5.2 decades/minute
5. Source Range, Neutron Flux	Not Applicable
6. Low Main Coolant Flow (steam generator ΔP)	$\geq$ 80% of Design Flow
7. Low Main Coolant Flow (main coolant pump current)	$\geq$ 240 Amperes, $\leq$ 960 Amperes

TABLE 2.2-1 (continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
8. High Main Coolant System Pressure	≤ 2300 psig
9. Low Main Coolant System Pressure	≥ 1800 psig
10. High Pressurizer Water Level	≤ 200 inches
11. Low Steam Generator Water Level	≥ - 13"*
12. Turbine Trip	Not Applicable
13. Generator Trip	Not Applicable
14. Main Steam Isolation Trip Logic	≥ 200 psig

*Where 0 inches corresponds to 10" above the feed ring centerline.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint limits specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and Main Coolant System are prevented from exceeding their safety limits.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range and Intermediate Power Range, Neutron Flux

The Power Range and Intermediate Power Range Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by pressurizer water level protective circuitry. The Power Range low set point provides additional protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed above 15 MWe and is manually reinstated at a power level below 15 MWe. The low setpoint trip is not assumed in the accident analysis.

The prescribed setpoint, with allowances for errors, is consistent with the trip point used in the accident analysis. The lower setting for three loop operation provides the protection at the reduced power level equivalent to that provided by the setting for four loop operation at full power.

Intermediate Range, Neutron Flux, High Startup Rate

The Intermediate Range High Startup Rate trip provides protection to limit the rate of power increase during low power conditions in the event of an uncontrolled rod withdrawal.

Low Main Coolant Flow (Steam Generator ΔP)

The Low Main Coolant Flow trips provide core protection in the event of a loss of one or more main coolant pumps.

Above a power of 15 MWE, with 4 main coolant pumps operating, an automatic reactor trip will occur if the flow in any two loops drops below 80% of nominal full loop flow and, with 3 main coolant pumps operating, automatic reactor trip will occur if the flow in any single operating loop drops below 80% of nominal full loop flow. The setpoints specified are consistent with the value assumed in the accident analysis.

Low Main Coolant Flow (Main Coolant Pump Current)

The Low Main Coolant Flow trips provide core protection in the event of a loss of one or more main coolant pumps.

Above a power of 15 MWE, with 4 main coolant pumps operating, an automatic trip will occur if the main coolant pump motor current is outside the limits on any two pumps, and with 3 main coolant pumps operating, automatic trip will occur if the main coolant pump motor current is outside the limits on any operating pump. The setpoints specified are consistent with the value assumed in the accident analysis.

Main Coolant System Low Pressure

The Main Coolant System Low Pressure trip is provided to prevent operation in the pressure range in which DNBR is less than 1.30 ensuring that the thermal and hydraulic limits assumed in the accident analysis are not exceeded. This Low Pressure trip provides protection by tripping the reactor in the event of a loss of main coolant pressure.

Pressurizer High Water Level

The Pressurizer High Water Level trip ensures protection against Main Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble, prevents water relief through the pressurizer safety valves, and provides core protection for an uncontrolled rod withdrawal incident or loss of load accident.

Steam Generator Water Level

The Low Steam Generator Water Level trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide 15 minutes, as assumed in the accident analysis, for starting delays of the emergency feedwater system.

Turbine and Generator Trip

A Turbine or Generator Trip causes a direct reactor trip when operating above 15 MWE. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability is required to enhance the overall reliability of the Reactor Protection System.

Main Steam Isolation Trip

A Main Steam Isolation Trip closes the main steam line non-return valves and causes a direct reactor trip. This trip reduces the severity of the cooldown and the ensuing transient effects resulting from a main steam line break. Its functional capability enhances the overall reliability of the Reactor Protection System.

Main Coolant System High Pressure

The Main Coolant System High Pressure trip is provided to ensure protection against main coolant system overpressurization caused by a loss of load incident. Its functional capability enhances the overall reliability of the Reactor Protection System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 5.5% $\Delta K/K$, for Main Coolant Core Average Temperatures $\geq$ 515°F.

The SHUTDOWN MARGIN shall be $\geq$ 4.72% $\Delta K/K$, for Main Coolant Core Average Temperatures < 485°F.

The SHUTDOWN MARGIN requirement is a linear function between 485°F and 515°F.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN less than required, immediately initiate and continue boration at $\geq$ 26 gpm of 2200 ppm boron concentration or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ that required:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod(s) is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 4 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality, by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

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* See Special Test Exception 3.10.1

\# With $K_{eff} \geq 1.0$

\##With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

1. Main Coolant System boron concentration,

2. Control rod position,

3. Main Coolant System average temperature,

4. Fuel burnup based on gross thermal energy generation,

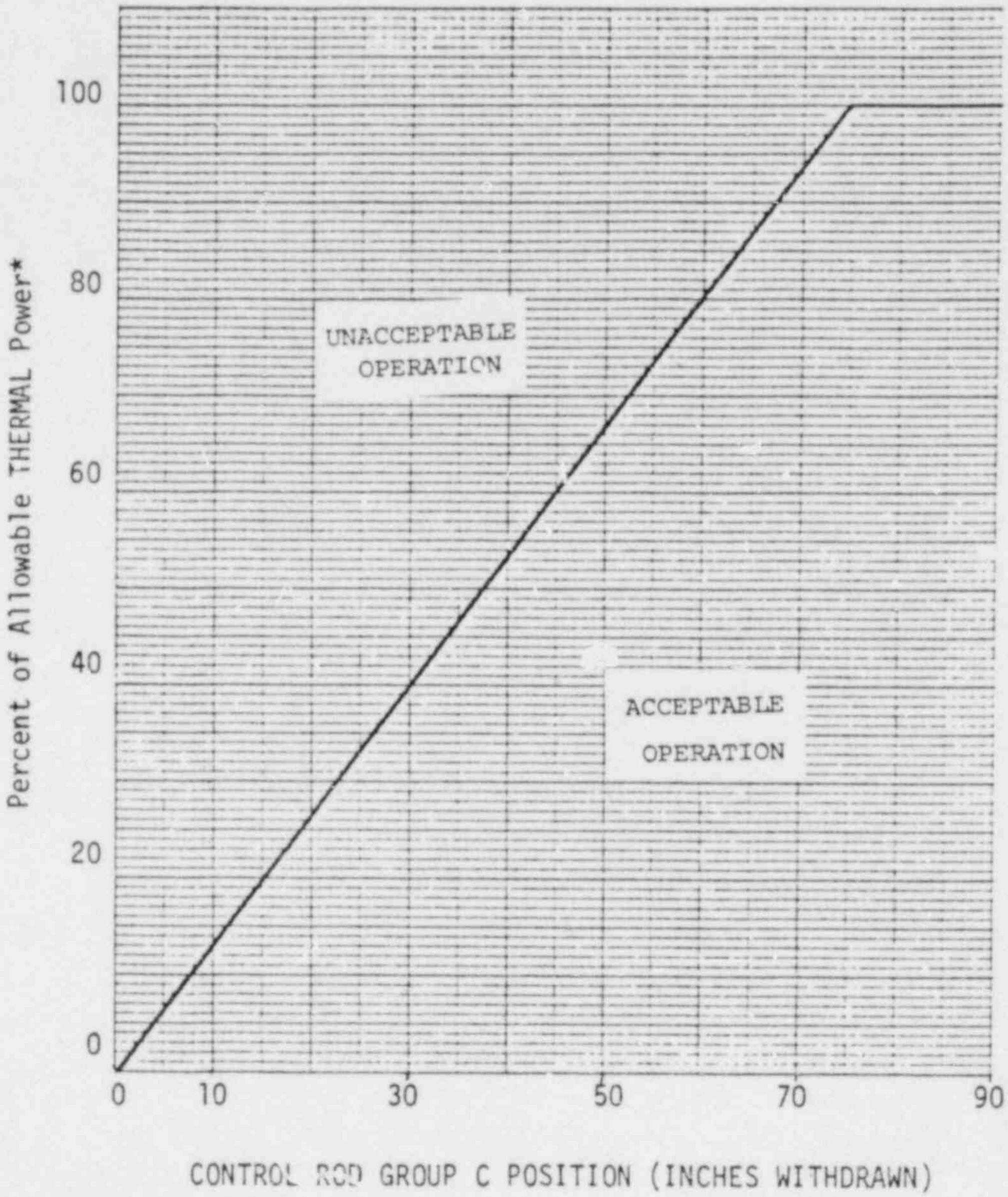
5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 0.8% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

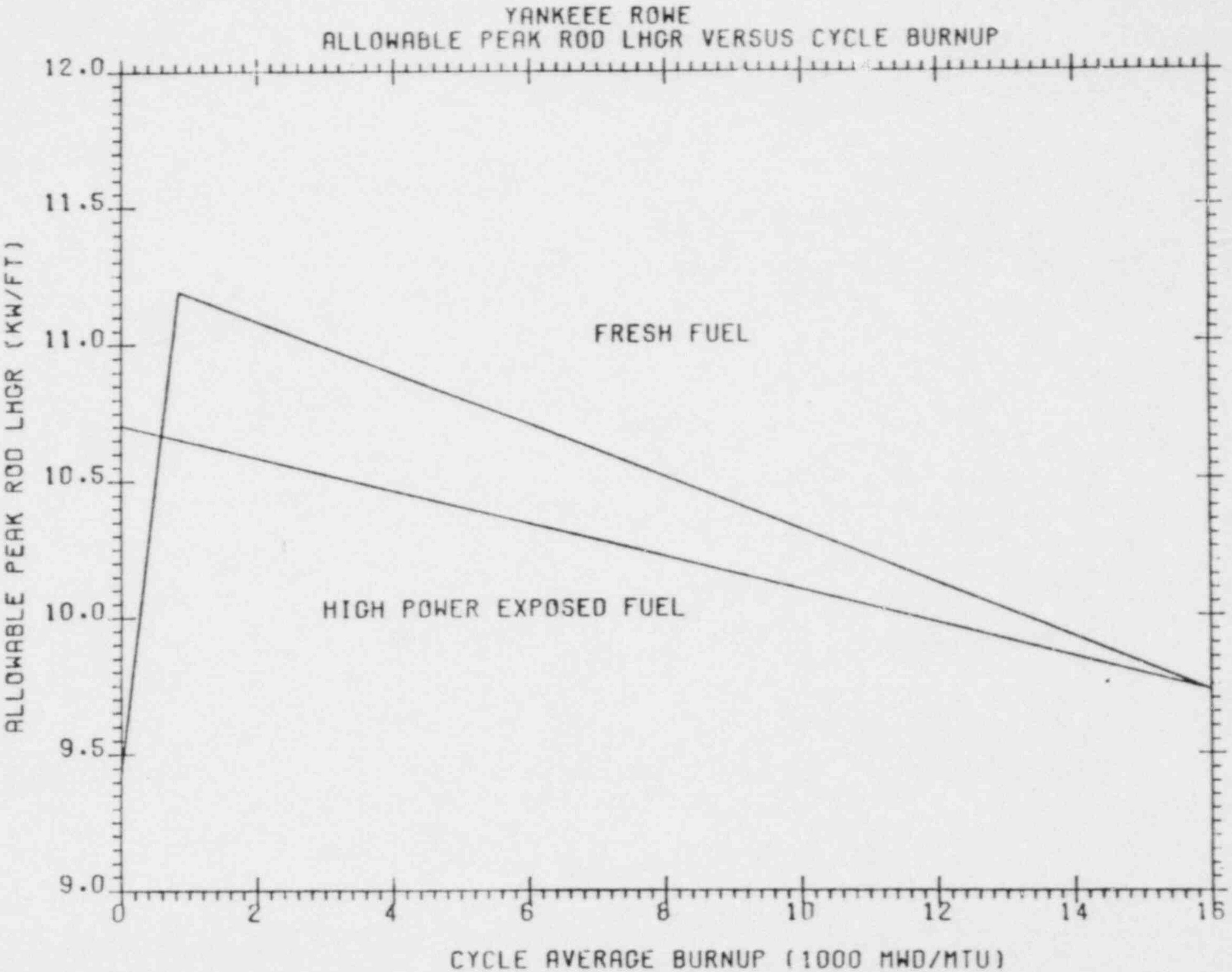
4.1.1.1.3 Whenever the reactor is shut down, before any operation which might result in a change of reactivity, a control rod group shall be withdrawn to a height sufficient to provide a reactivity worth of 1% for emergency shutdown capability. If for any reason this is not practical, the Main Coolant System shall be borated to provide 5% Δk/k SHUTDOWN MARGIN with all control rods inserted.

4.1.1.1.4 During a reactor startup in which core reactivity or control rod positions for criticality are not established, a plot of inverse multiplication rate (or count rate) versus rod position shall be made.



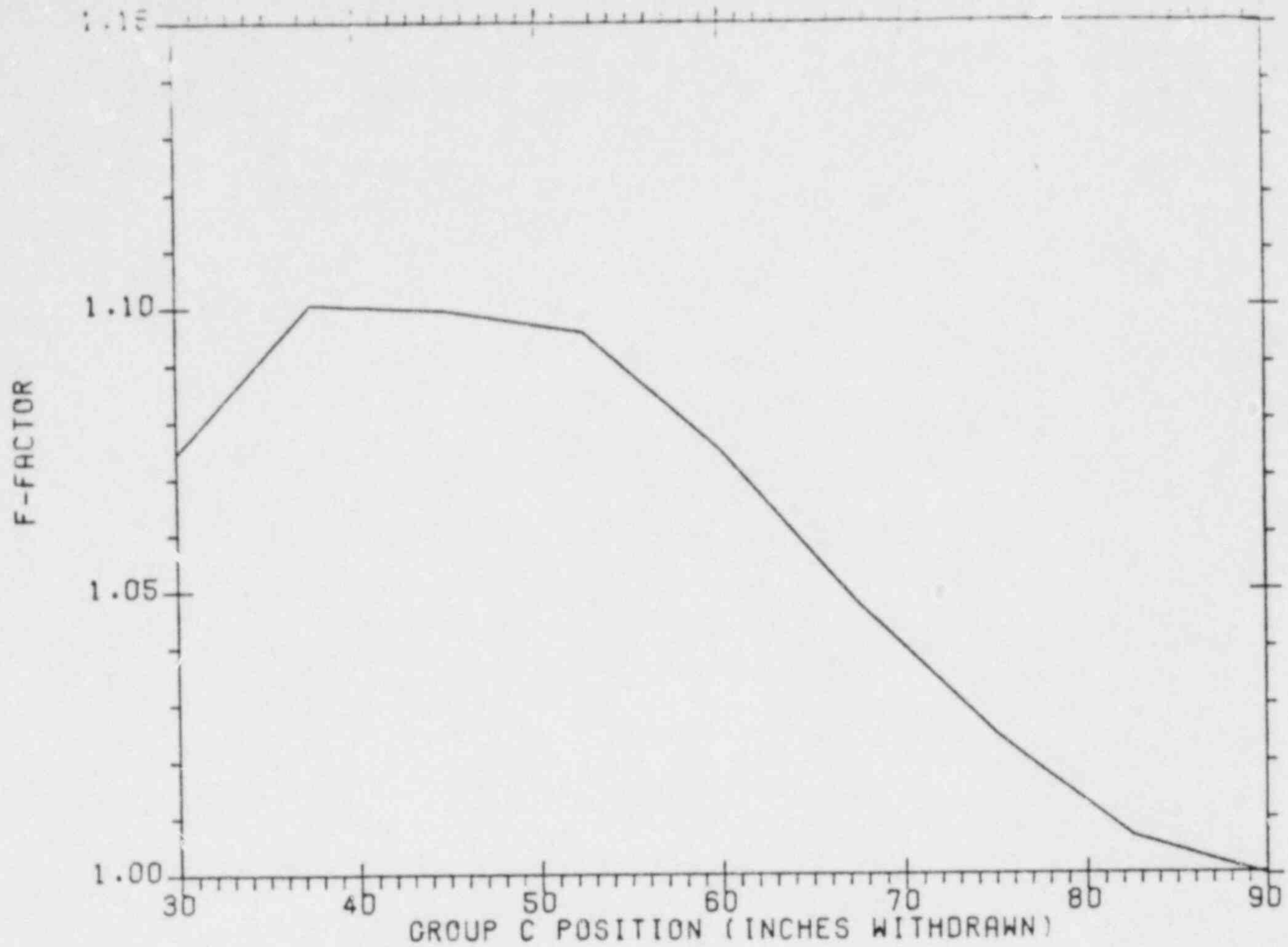
*Allowable THERMAL Power based on the main coolant pump combination in operation.

FIGURE 3.1-1



ALLOWABLE PEAK ROD LHGR
VERSUS CYCLE BURNUP

FIGURE 3.2-1



$$F_I = \frac{F\ @\ \text{Limit}}{F\ @\ \text{Measurement}}$$

FIGURE 3.2-2

Factor F as a Function of Rod Insertion

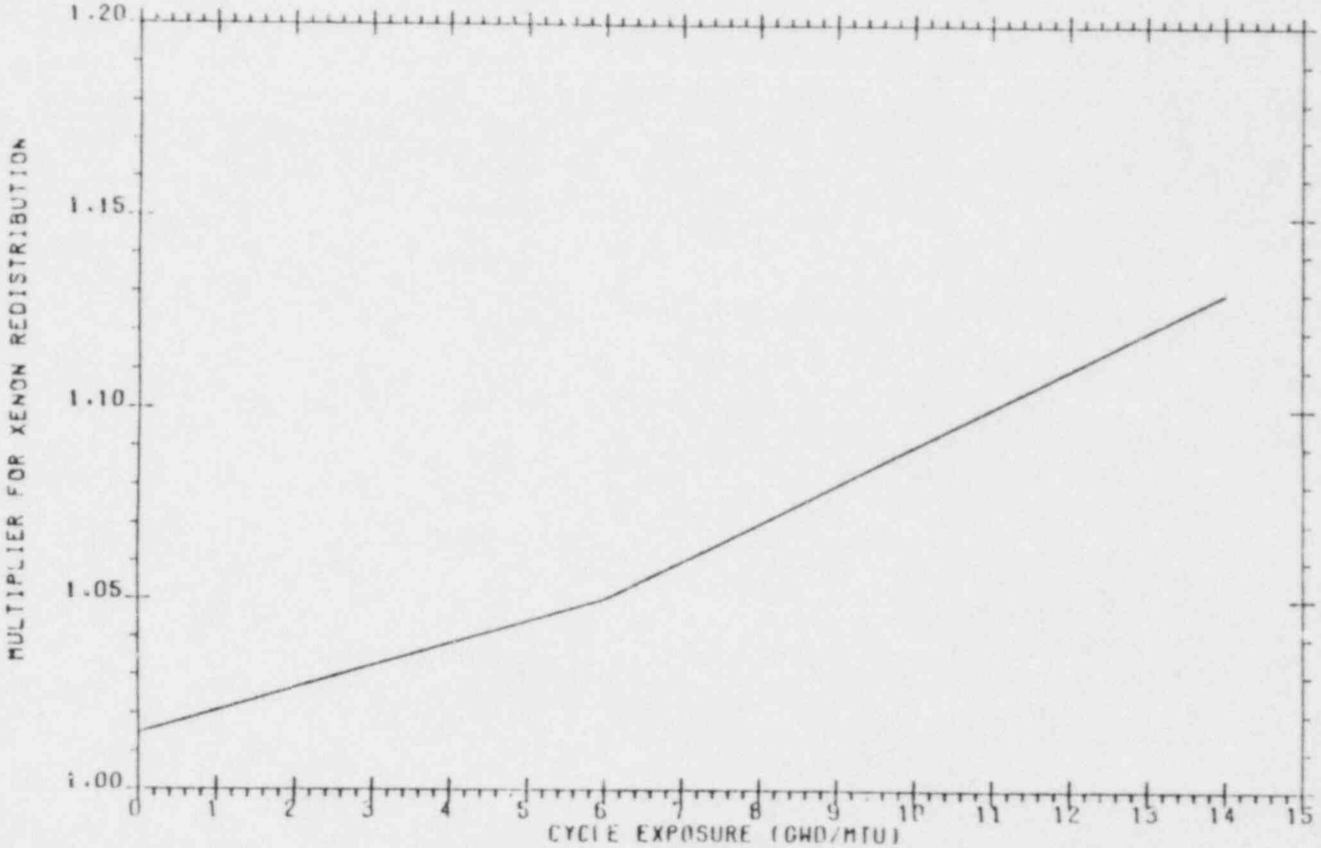


Figure 3.2-3

Multiplier for Xenon Redistribution as a Function of Exposure

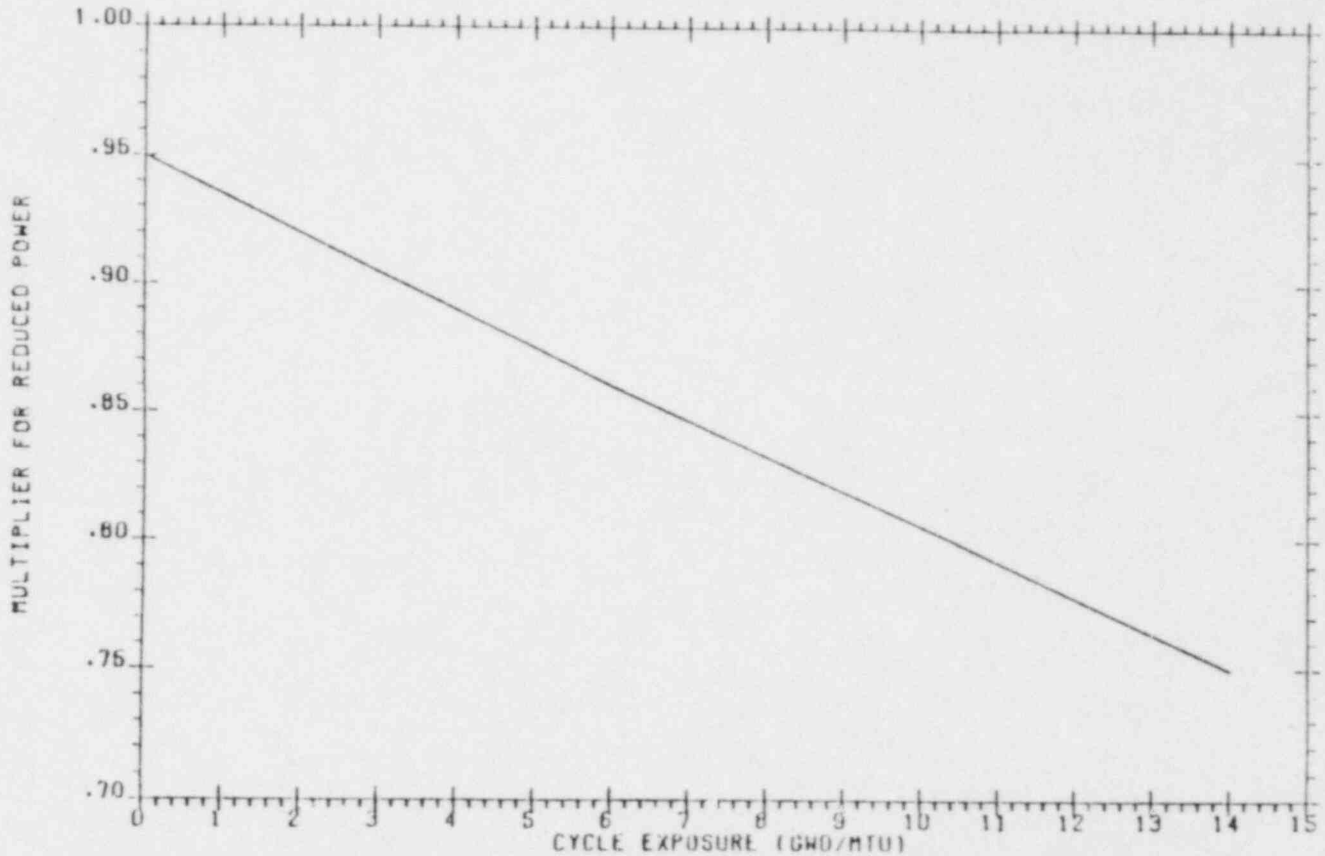


Figure 3.2-4

<u>Multiplier for Reduced Power as a Function of Exposure</u>

HEAT FLUX HOT CHANNEL FACTOR-F_q

LIMITING CONDITION FOR OPERATION

3.2.2 F_q shall be limited by the following relationships:

$$F_q \leq \frac{[2.76]}{P} \quad \text{for } P > 0.5$$

$$F_q \leq [5.52] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1

ACTION:

With F_q exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% F_q exceeds the limit within 15 minutes and similiarly reduce the Power Range and Intermediate Power Range Nuetron Flux-High Trip Setpoints within the next 4 hours.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided F_q is demonstrated through incore mapping to be within its limit.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective system instrumentation channels and reactor permissive functions of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the Reactor Permissive Circuit shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by permissive circuit operation. The total permissive function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by permissive circuit operation.

TABLE 3.3-1

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	3	1	3	1, 2 and *	1
2. Power Range, Neutron Flux and Intermediate Power Range, Neutron Flux	6	2	4	1, 2 and $*^{(1)}$	2**
3. Intermediate Range, Neutron Flux, High Startup Rate	2	1	2	$1^{(2)}$, 2 and *	3
4. Source Range, Neutron Flux					
a. Startup##	2	NA	2	$2^{\#}$ and $*^{(5)}$	4
b. Shutdown	2	NA	1	3, 4, $5^{(5)}$	5
5. Low Main Coolant Flow (SG P)	4	2	3	$1^{(3)}$	6**
6. Low Main Coolant Flow (MC Pump Current)					
a. System A	4	2	3	$1^{(3)}$	7**
b. System B	4	2	3	$1^{(3)}$	7**
7. High Main Coolant System Pressure	3	2	3	1, $2^{(4)}$	6**
8. Low Main Coolant System Pressure	3	2	3	1, $2^{(4)}$	6**
9. High Pressurizer Water Level	1	1	1	1, $2^{(4)}$	8
10. Low Steam Generator Water Level	4	2	3	$1^{(3)}$	6**

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Turbine Trip	1	1	1	$1^{(3)(6)}$	8
12. Generator Trip	1	1	1	$1^{(3)(7)}$	8
13. Reactor Trip Breaker	2	1	2	1, 2 and *	9
14. Automatic Trip Logic	2	1	2	1, 2 and *	9
15. Main Steam Isolation Trip Logic	2	1	2	1, $2^{(4)}$	6**

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

**The provisons of Specification 3.0.4 are not applicable.

#High voltage to detector is automatically de-energized above 5×10^{-9} Amperes on the Intermediate Range.

##Or when other activities might increase reactivity.

(1) Power Range, Neutron Flux, Low Setpoint Trip may be manually bypassed at $\geq$ 15 MWe. Bypass shall be manually removed at $\leq$ 15 MWe.

(2) Intermediate Range, Neutron Flux, High Startup Rate Trip is automatically bypassed $\geq$ 15 MWe. Bypass is automatically removed at $\leq$ 15 MWe.

(3) Trip may be manually bypassed $\leq$ 15 MWe. Bypass is automatically removed at $\geq$ 15 MWe.

(4) Trip may be manually bypassed when the reactor is not critical.

(5) Startup rate alarm setpoint $\leq$ 1.1 decade/minute.

(6) Turbine shall be protected by at least the following protective trips: rotor excessive axial movement, low bearing oil pressure; low condenser vacuum; and overspeed.

(7) Generator shall be protected by at least the following protective trips: overcurrent; differential; and loss of field.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 7 (Continued) -

b) The Minimum Channels OPERABLE requirement for each System is met; however, one additional channel in either system may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 8 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 9 - With the number of channels OPERABLE one less than required by the Minimum Channels Operable requirement, be in at least HOT STANDBY within 6 hours with reactor trip breakers open.

TABLE 4.3-1

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	NA	NA	S/U[1]	NA
2. Power Range, Neutron Flux and Intermediate Power Range, Neutron Flux	S	D[2], Q[5]	M	1, 2 and *
3. Intermediate Range, Neutron Flux, High Startup Rate	S	R[5]	M	1, 2 and *
4. Source Range, Neutron Flux	S	R[5]	S/U[1]	2, 3, 4, 5 and *
5. Low Main Coolant Flow (SGΔP)	S	R[4]	M[3]	1
6. Low Main Coolant Flow, Systems A and B (MC Pump Current)	S	R	M	1
7. High Main Coolant System Pressure	S	R[4]	M	1, 2
8. Low Main Coolant System Pressure	S	R[4]	M	1, 2
9. High Pressurizer Water Level	S	R[4]	M[3]	1, 2
10. Low Steam Generator Water Level	S	R[4]	M	1

TABLE 4.3-1 (continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11. Turbine Trip	NA	NA	S/U(1)	1
12. Generator Trip	NA	NA	S/U(1)	1
13. Reactor Trip Breaker	NA	NA	S/U(1)	1, 2 and *
14. Automatic Trip Logic	NA	NA	S/U(1)	1, 2 and *
15. Main Steam Isolation Trip Logic	NA	NA	Q	1, 2

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in the previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER, at least 3 times per week with a maximum time interval of 72 hours.

(3) - When shutdown longer than 24 hours, if not performed in the previous 31 days.

(4) - Known pressure applied to sensor.

(5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

TABLE 3.3-2 (Continued)

ENGINEERING SAFEGUARDS SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS AND SENSORS	CHANNELS AND SENSORS TO TRIP	MINIMUM CHANNELS AND SENSORS OPERABLE	APPLICABLE MODES	ACTION
2. CONTAINMENT ISOLATION (Continued)					
c. Actuation Channel B	1	1	1	1, 2, 3, 4, 5(1)	10
1) High Containment Pressure Sensor	1	1	1	1, 2, 3, 4, 5(1)	10
2) Safety Injection	(All Safety Injection Initiating Functions and Requirements)				
3. MAIN STEAM ISOLATION					
a. Low Steam Line Pressure	3/Steam Line	2/Steam Line	3/Steam Line	1, 2	6**
b. Automatic Trip Logic	2	1	2	1, 2(4)	6**
c. Manual Initiation	2	1	2	1, 2	6**
d. High Containment Pressure Trip Containment Isolation	2	1	2	1, 2	6**

TABLE 3.3-2 (continued)

TABLE NOTATION

** The provisions of Specification 3.0.4 are not applicable.

(1) Trip function may be bypassed in this MODE with main coolant pressure < 300 psig.

(2) Trip function may be bypassed in this MODE with main coolant pressure < 1800 psig.

(3) Automatic initiation of Actuation Channel #1 may be bypassed in this MODE during functional test of the Main Coolant System pressure channel.

(4) Trip may be manually bypassed when the reactor is not critical.

ACTION STATEMENTS

ACTION 10 - With the number of OPERABLE channels or sensors one less than the Total Number of Channels or sensors, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one safety injection channel high containment pressure sensor may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 1 hour.

2. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

TABLE 3.3-3

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. SAFETY INJECTION	
a. Actuation Channel #1	
1) RPS Low Main Coolant Pressure Channel	$\geq$ 1700 psig
2) High Containment Pressure Sensor	$\leq$ 5 psig
3) Manual Initiation	Not Applicable
b. Actuation Channel #2	
1) Low Main Coolant Pressure Sensor	$\geq$ 1700 psig
2) High Containment Pressure Sensor	$\leq$ 5 psig
3) Manual Initiation	Not Applicable
2. CONTAINMENT ISOLATION	
a. Manual Initiation	Not Applicable
b. Actuation Channel A	
1) High Containment Pressure Sensor	$\leq$ 5 psig
2) Safety Injection	(All Safety Injection Setpoints)
c. Actuation Channel B	
1) High Containment Pressure Sensor	$\leq$ 5 psig
2) Safety Injection	(All Safety Injection Setpoints)
3. MAIN STEAM ISOLATION	
a. Low Steam Line Pressure	$\geq$ 200 psig
b. Automatic Trip Logic	Not Applicable
c. Manual Initiation	Not Applicable
d. High Containment Pressure Trip-Containment Isolation	$\leq$ 5 psig

TABLE 4.1-2 (Continued)

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. CONTAINMENT ISOLATION (Continued)				
c. Actuation Channel B	S	N.A.	M(4)	1, 2, 3, 4, 5*
1) High Containment Pressure Sensor	S	R(3)	M(3)	1, 2, 3, 4, 5*
2) Safety Injection	(All Safety Injection Surveillance Requirements)			
3. MAIN STEAM ISOLATION				
a. Low Steam Line Pressure	S	R(3)	M(3)	1, 2
b. Automatic Trip Logic	N.A.	N.A.	Q	1, 2
c. Manual Initiation	N.A.	N.A.	R	1, 2
d. High Containment Pressure Trip	N.A.	N.A.	R	1, 2

SURVEILLANCE REQUIREMENTS (continued)

2. Verifying that the following valves are in their normally opened positions with power to the valve operators removed by removal of the circuit breaker from the motor control center:

Valve Number	Valve Function
a. SI-MOV-4	LPSI Pump Cross Over to HPSI Pump
b. SI-MOV-22	SI Header Isolation to Cold Leg
c. SI-MOV-23	SI Header Isolation to Cold Leg
d. SI-MOV-24	SI Header Isolation to Cold Leg
e. SI-MOV-25	SI Header Isolation to Cold Leg

3. Verifying that power to the valve operators is removed by disconnecting the power cables as they leave the motor starters:

Valve Number	Valve Function
a. CS-MOV-536	SI Header Isolation to Cold Leg
b. CS-MOV-537	SI Header Isolation to Cold Leg
c. CS-MOV-538	SI Header Isolation to Cold Leg
d. CS-MOV-539	SI Header Isolation to Cold Leg
e. MC-MOV-301	MCS Loop Isolation
f. MC-MOV-302*	MCS Loop Isolation
g. MC-MOV-309	MCS Loop Isolation
h. MC-MOV-310*	MCS Loop Isolation
i. MC-MOV-318*	MCS Loop Isolation
j. MC-MOV-319	MCS Loop Isolation
k. MC-MOV-325	MCS Loop Isolation
l. MC-MOV-326*	MCS Loop Isolation

In MODE 2, 3, 4*, 5*, power cables may be connected to the MCS loops isolation valves when required to close the valves for main coolant pump(s) starting. After the pump(s) has been started, the valve(s) shall be reopened and power cables disconnected.

4. Verifying that the following valves are in their normally closed position with power to the valve operator removed by disconnecting the power cables as they leave the motor starter:

Valve Number	Valve Function
a. CS-MOV-532	LPSI Recirculation Line
b. CS-MOV-534	LPSI Pump Header Isolation Valve Bypass

Note: CS-MOV-532 may be opened for ≤ 30 minutes once per week for safety injection tank mixing or low pressure safety injection pump testing after restoring power to the valve operator. Insure that power to the valve operator is properly removed after closing the valve.

5. Verifying that the following valves are in their normal position with power to the valve operator motors separated by dual contactors from the motor control center:

Valve Number	Valve Function	Normal Position
a. CS-MOV-533	LPSI Pump Header Isolation	Open
b. CS-MOV-535	LPSI Pump Header Isolation	Open
c. SI-MOV-518	LPSI Pump Header Suction Isolation	Open
d. SI-MOV-48	HPSI and LPSI Minimum Recirculation Line	Open
e. SI-MOV-49	HPSI and LPSI Minimum Recirculation Line	Open
f. SI-MOV-515	Hot Leg Injection Isolation	Closed
g. SI-MOV-514	Hot Leg Injection Isolation	Closed
h. SI-MOV-516	V.C. Sump Isolation	Closed
i. SI-MOV-517	V.C. Sump Isolation	Closed
j. SI-MOV-46	HPSI Flow Control	Open

6. Verifying that each ECCS safety injection subsystem is aligned to receive electrical power from an OPERABLE emergency bus.

7. Verifying that each pair of ECCS recirculation subsystem redundant valves is aligned to receive electrical power from separate OPERABLE busses.

8. Verifying that each pair of ECCS long-term hot leg injection subsystem redundant valves is aligned to receive electrical power from separate OPERABLE busses.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
A. AUTOMATIC ISOLATION VALVE			
TV-401A	No. 1 SG Blowdown	Yes	30
TV-401B	No. 2 SG Blowdown	Yes	30
TV-401C	No. 3 SG Blowdown	Yes	30
TV-401D	No. 4 SG Blowdown	Yes	30
TV-408	Containment Cooling Water Return	Yes	30
TV-409	Containment Heater Condensate Return	Yes	30
VD-SOV-301	Air Particulate Monitor-in	Yes	30
VD-SOV-302	Air Particulate Monitor-out	Yes	30
HV-SOV-1	Hydrogen Vent System	Yes	30
HV-SOV-2	Hydrogen Vent System	Yes	30
TV-202	Main Coolant Drain	Yes	30
TV-203	Main Coolant Vent	Yes	30
TV-204	Valve Stem Leakoff	Yes	30
TV-205	Component Cooling Return	No	30
TV-206	Main Coolant Sample	Yes	30
TV-207	Neutren Shield Tank Sample	Yes	30
TV-209	Containment Drain	Yes	30
TV-211	Containment Pressure Sensing	Yes	30
TV-212	Containment Pressure Sensing	Yes	30
TV-213	LP Sample	Yes	30

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
A. AUTOMATIC ISOLATION VALVE (Continued)			
TV-406*	Main Steam Drain to Condenser	No	30
TV-411*	Atmospheric Steam Dump	Yes	30
B. CHECK VALVES			
SI-V-14*	Safety Injection (HP)	NA	NA
CS-V-621*	Safety Injection (LP)	NA	NA
CH-V-611*	MC Feed to Loop #4	NA	NA
CC-V-667*	Component Cooling to MCP #1	NA	NA
CC-V-663*	Component Cooling to MCP #2	NA	NA
CC-V-671*	Component Cooling to MCP #3	NA	NA
CC-V-675*	Component Cooling to MCP #4	NA	NA
CC-V-649*	Component Cooling to Sample Cooler	NA	NA
CC-V-653*	Component Cooling to Neutron Shield Tank Coolers	NA	NA
CC-V-660*	Neutron Shield Tank Fill	NA	NA

*Not subject to Type C tests

TABLE 3.6-1 (Cont'd)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME (Seconds)
B. CHECK VALVES (Continued)			
SW-V-820*	Service Water to Containment Cooler #1	NA	NA
SW-V-821*	Service Water to Containment Coole. #2	NA	NA
SW-V-822*	Service Water to Containment Cooler #3	NA	NA
SW-V-823*	Service Water to Containment Cooler #4	NA	NA
HC-V-1199*	Steam Supply to Containment Heaters	NA	NA
C. Manual Valves			NA
SC-MOV-551+553*	Shutdown Cooling - In	No	NA
SC-MOV-552+554*	Shutdown Cooling - Out	No	NA
CH-MOV-522*	MC Feed to Loop Fill Header	NA	NA
CS-V-601	Shield Tank Cavity Fill	NA	NA
CA-V-746*	Containment Air Charge	NA	NA
HV-V-5*	Containment H2 Vent System	NA	NA
HV-V-6*	Containment H2 Vent System	NA	NA
CA-V-688	Containment H2 Vent System Air Supply	NA	NA
CS-MOV-500	Fuel Chute Lock Valve	No	NA

*Not subject to Type C tests

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
C. MANUAL VALVES (Cont'd)			
CS-CV-215	Fuel Chute Equalizing	NA	NA
CS-CV-216	Fuel Chute Dewatering Pump Discharge	NA	NA
VD-V-752*	Neutron Shield Tank-Outer Test	NA	NA
VD-V-754*	Neutron Shield Tank-Inner Test	NA	NA
BF-V-4-1	Air Purge Inlet	NA	NA
BF-V-4-2	Air Purge Outlet	NA	NA
HC-V-602	Air Purge Bypass	NA	NA
SI-MOV-516	ECCS Recirculation	No	NA
SI-MOV-517	ECCS Recirculation	No	NA
BF-CV-1000*	SG#1 Feedwater Regulator	No	30
BF-CV-1100*	SG#2 Feedwater Regulator	No	30
BF-CV-1200*	SG#3 Feedwater Regulator	No	30
BF-CV-1300*	SG#4 Feedwater Regulator	No	30
NRV-405A*	Main Steam Non-Return Valve	No	5
NRV-405B*	Main Steam Non-Return Valve	No	5
NRV-405C*	Main Steam Non-Return Valve	No	5
NRV-405D*	Main Steam Non-Return Valve	No	5

*Not subject to Type C tests.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	TESTABLE DURING PLANT OPERATION (Yes or No)	ISOLATION TIME Seconds
C. MANUAL VALVES (Cont'd)			
PR-V-610	Main Coolant Heise Pressure Gauge	NA	NA
PU-V-543	Purification System Containment Sump Suction	NA	NA
PU-V-544	Purification System Containment Sump Suction	NA	NA
EBF-MOV-557*	Alternate S.G. Feed	NA	NA
MS-V-627***	Main Steam Bypass	NA	NA
MS-V-628***	Main Steam Bypass	NA	NA
MS-V-629***	Main Steam Bypass	NA	NA
MS-V-630***	Main Steam Bypass	NA	NA
AS-V-719*	Emergency Feed Pump Steam Supply	NA	NA
AS-V-720*	Steam Drain	NA	NA

* Not subject to Type C tests.

*** Valve may be open for a 4 hour period during secondary plant heat-up and pressure equalization in Modes 2 and 3. Not subject to type C tests.

LIMITING CON[illegible]ITION FOR OPERATION

3.7.1.2 At least two independent emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One feedwater pump capable of being powered from an emergency bus, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one emergency feedwater pump inoperable, restore at least two emergency feedwater pumps (one capable of being powered from an emergency bus, and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting the pump.

2. Verifying that the motor driven pump develops a discharge pressure of $\geq$ 950 psig while recirculating back to the supply tanks.

3. Verifying that, on recirculation flow, the steam turbine driven pump develops a discharge pressure of $\geq$ 950 psig when the secondary steam pressure is greater than 100 psig.

4. Verifying that the pump operates for at least 15 minutes.

5. Cycling each testable manual and power operated valve in the flow path through at least one complete cycle of full travel.

6. Verifying that each valve in the flow path that could interrupt all emergency feedwater flow is locked open and the remaining valves are verified to be in the correct position.

b. At least once per 18 months during shutdown by:

1. Cycling each manual valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that the steam turbine drive pump develops a discharge pressure of $\geq$ 950 psig at a flow of $\geq$ 80 gpm while feeding a steam generator and that the motor driven pumps each develop a discharge pressure of $\geq$ 950 psig with a flow of at least 80 gpm while feeding a steam generator.

3. Cycling each main feed control valve manually through at least one complete cycle of full travel.

c. Prior to startup from COLD SHUTDOWN by:

1. Verifying that each valve in the flow path from the emergency feedwater sources to the main feedwater and the steam generator blowdown lines is properly aligned to provide an uninterrupted flow path to the steam generators from the emergency feedwater system, and

2. Performing a flow test from the emergency feedwater sources to the steam generators to verify the normal flow path.

PLANT SYSTEMS

M_N STEAM NON-RETURN VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam non-return valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

MODES 1 - With one main steam non-return valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam non-return valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided the inoperable valve is maintained closed; otherwise, be in at least HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam non-return valve that is open shall be demonstrated OPERABLE by:

a. Cycling each valve through at least 10% of full travel at least once per 92 days, and

b. Verifying full closure within 5 seconds on any closure actuation signal whenever shutdown longer than 24 hours, if not performed in the previous 92 days.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, Main Coolant System boron concentration, and Main Coolant System T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled Main Coolant System cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 4.72% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with accident analysis assumptions. The value of 5.5% $\Delta k/k$ is incorporated to provide added SHUTDOWN MARGIN and reflects the actual excess of shutdown margin available at the plant. With $T_{avg} \leq 330^{o}F$, the reactivity transients resulting from a postulated steam line break cooldown are minimal. 5% $\Delta k/k$ SHUTDOWN MARGIN (with all rods inserted) provides adequate protection to preclude criticality for all postulated accidents with the reactor vessel head in place.

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. Normally, when full power is reached after each refueling, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted steady state curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction and the reactivity status of the core can be continuously evaluated, and any deviation would be thoroughly investigated and evaluated.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 950 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Main Coolant System. A flow rate of at least 950 GPM will circulate an equivalent Main Coolant System volume of 2,940 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control. Restriction of boron dilution with Main Coolant System temperature < 250°F prevents inadvertent criticality due to excess dilution below the temperature limit for criticality.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in Main Coolant System boron concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

A 34% DNBR credit is needed to offset the full-closure rod bow penalty in Yankee Rowe. The full-closure penalty was previously approved (D. Ross and D. Eisenhut memorandum of December 12, 1976) for Yankee Rowe since a gap closure model was not available. Generic credits (D. Edwards letter to NRC dated February 9, 1977) equivalent to 13.2% DNBR margin were approved for Yankee Rowe. The limiting transient for Yankee Rowe with respect to DNB is the 2 of 4 pump loss of flow. Based on design conditions, this event results in a minimum DNBR in excess of 2.05. Thus, 36.6% margin to a DNBR of 1.3 exists for this limiting event, which is applied to the remaining 20.8% margin required by the rod-bow penalty.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1035 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Code, 1956 Edition. The total relieving capacity for all valves on all of the steam lines is 3.1×10^6 lbs/hr which is 129 percent of the total secondary steam flow of 2.4×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Tables 3.7-1 and 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (108)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (81)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam generator

U = Maximum number of inoperable safety valves per operating steam generator

(108) = Power Range and Intermediate Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

(81) = Maximum percent of RATED THERMAL POWER permissible for 3 loop operation

X = Total relieving capacity of all safety valves per steam generator in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater system ensures that the Main Coolant System can be cooled down to less than 330°F from normal operating conditions in the event of a total loss of off-site power.

Each emergency feedwater pump is capable of delivering a total feedwater flow of 80 gpm at a pressure of 950 psig. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Main Coolant System temperature to less than 330°F when the Shutdown Cooling System may be placed into operation.

The monthly testing interval of the steam generator emergency feedwater pumps verifies their operability by recirculating water to the supply tank. Proper functioning of the emergency feedwater pumps will be made by direct visual observation.

3/4.7.1.3 PRIMARY AND DEMINERALIZED WATER STORAGE TANK

The OPERABILITY of the primary and demineralized water storage tanks with the minimum combined water volume ensures that sufficient water is available to maintain the Main Coolant System at HOT STANDBY in excess of 24 hours with steam discharge to the atmosphere concurrent with total loss of off-site power.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

The steam break accident is based upon a postulated release of the entire contents of the secondary system to the atmosphere using a site boundary dose limit of 1.31 rem for thyroid dose.

The limiting dose for this accident results from iodine in the secondary coolant. The reactor distribution of iodine isotopes with 1% failed fuel was used for this calculation. I-131 is the dominant isotope because of its low MPC in air and because the other iodine isotopes have shorter half-lives and therefore cannot build up to significant concentrations in the secondary coolant, given the limitations on primary system leak rate and activity. The entire secondary system contains approximately $132m^3$ of water at standard conditions. One-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets.

3/4.7.1.5 MAIN STEAM NON-RETURN VALVES

The OPERABILITY of the main steam non-return valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. Main steam non-return valve auto-closure minimizes the Main Coolant System cooldown associated with the blowdown. This feature enhances plant performance by:

1) Minimizing the reactivity transient.

2) Minimizing the Main Coolant and Secondary System thermal transient.

3) Providing additional backup to normal non-return action as a check valve to limit the containment transient resulting from a main steam line rupture inside the containment.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel spherical shell having the following design features:

a. Nominal inside diameter = 125 feet.

b. Minimum thickness of steel shell = 7/8 inches.

c. Net free volume = 860,000 cubic feet.

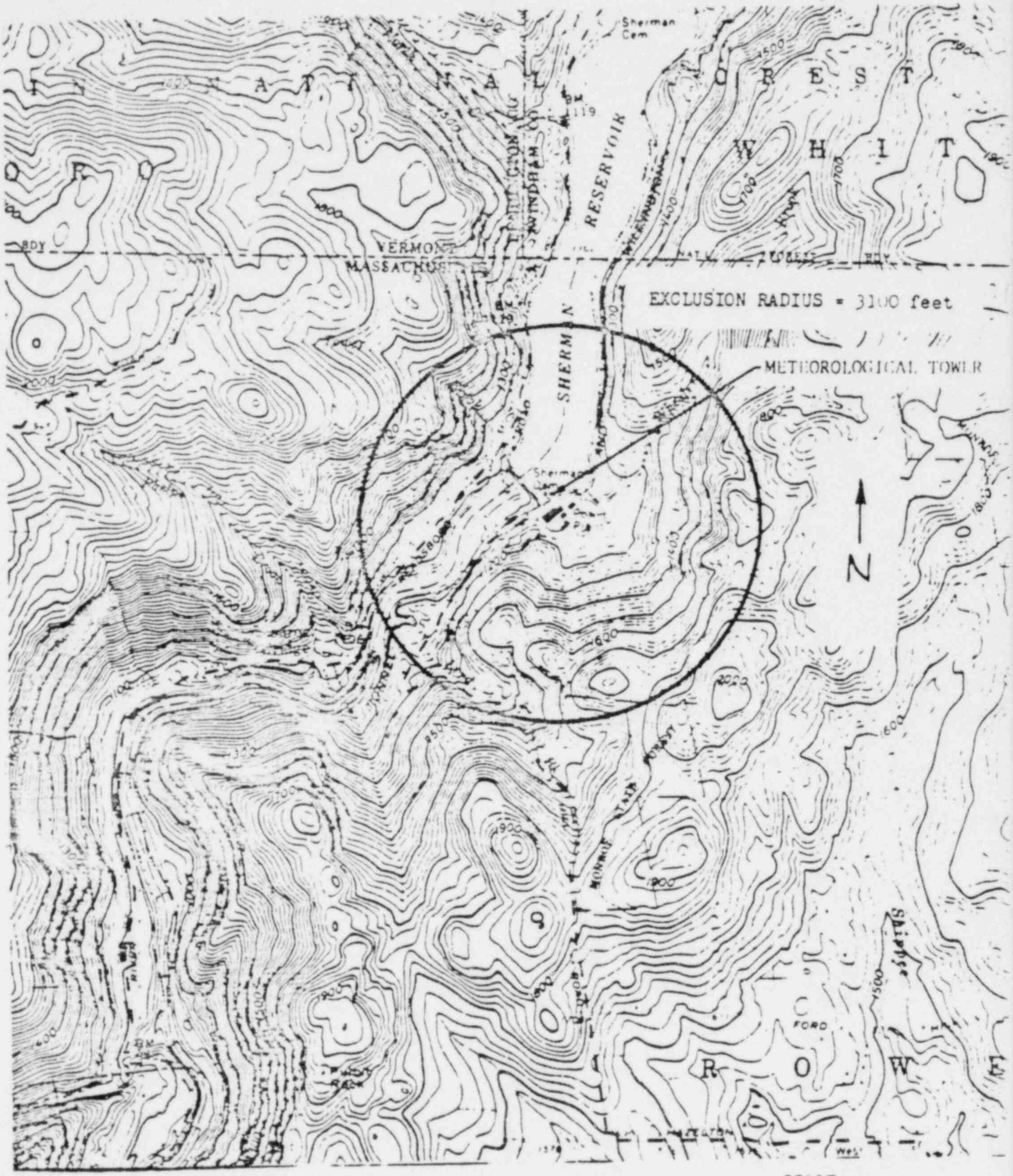
DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 34.5 psig and a temperature of 249^{o}F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 76 fuel assemblies with each fuel assembly containing 230 or 231 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 91 inches. Each fuel assembly shall contain a maximum total weight of 234 kilograms uranium. Reload fuel is similar in physical design to the Core XII EXXON fuel and shall have a maximum enrichment of 3.5 weight percent U-235.



SCALE
1 inch = 2000 feet

EXCLUSION AREA

FIGURE 5.1-1

LOW POPULATION ZONE

FIGURE 5.1-2

CONTROL ROD

5.3.2 The reactor core shall contain 24 control rods. The control rods shall contain a nominal 90 inches of absorber material. The nominal values of this absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. The silver-indium-cadmium control rods shall be clad with Inconel.

5.4 MAIN COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Main Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in ASME Boiler and Pressure Vessel Code, Section VIII, including all addenda through 1956, and the ANSI (formerly ASI) Standards, Power Piping Code, B31.1, 1955 Edition, and B16.5, 1957 Edition, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 668°F.

VOLUME

5.4.2 The total water and steam volume of the Main Coolant System is 2940 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

5.1.1 The meteorological tower shall be located as shown in Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to -0.95 with the new or spent fuel storage areas flooded with unborated water. The k_{eff} of -0.95 includes a conservative allowance of 3% k/k for uncertainties.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-3

YANKEE ATOMIC ELECTRIC COMPANY

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

DOCKET NO. 50-29

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