SSES UNIT 2 TECHNICAL SPECIFICATIONS CHANGES

2. Accordingly, for the Facility Operating License No. NPF-22, paragraph 2.C.(1) is hereby amended to read as follows:

   (1) Maximum Power Level

   3441

   Pennsylvania Power & Light Company (PP&) is authorized to operate the facility at reactor core power levels not in excess of 3293 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: JUN 27 1984

FROM SSES UNIT 2 OPERATING LICENSE NO. NPF-22

DEFINITIONS

RATED THERMAL POWER

1.33 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3293~~ MWT. 3441

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.35 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.36 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows, resilient material seals, or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1a.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than ~~10% of rated flow.~~ 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than ~~10% of rated flow~~, 10 million lbm/hr. be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than ~~10% of rated flow.~~ 10 million lbm/hr.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06* and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than ~~10% of rated flow~~ 10 million lbm/hr, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

*See Specification 3.4.1.1.2.a for single loop operation requirement.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	NO CHANGE ≤ 15% of RATED THERMAL POWER	NO CHANGE ≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.58 W+59%#, with a maximum of	≤ 0.58 W+62%#, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ ~~1037~~ 1087 psig	≤ ~~1067~~ 1093 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero*	≥ 11.5 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10% closed	≤ 11% closed
6. Main Steam Line Radiation - High	≤ 7.0 x full power background	≤ 8.4 x full power background
7. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 88 gallons	≤ 88 gallons
b. Float Switch	≤ 88 gallons	≤ 88 gallons
9. Turbine Stop Valve - Closure	≤ 5.5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 460 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.
#See Specification 3.4.1.1.2.a for single loop operation requirements.

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2 for SNP fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. XN-NF-524(A) Revision 1).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the XN-3 correlation is valid for critical power calculations at pressure greater than 580 psig and bundle mass fluxes greater than 0.25×10^6 lbs./hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity Safety Limit is established by a limiting condition on core THERMAL POWER with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to assure a minimum bundle flow for all fuel assemblies which have a relatively high power and potentially can approach a critical heat flux condition. For the SNP 9 x 9 fuel design, the minimum bundle flow is greater than 30,000 lbs/hr. For the SNP 9 x 9 design, the coolant minimum flow and maximum flow area is such that the mass flux is always greater than 0.25×10^6 lbs/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lbs/hr-ft^2 is 3.35 Mwt or greater. At 25% thermal power a bundle power of 3.35 Mwt corresponds to a bundle radial peaking factor of ~~greater than~~ 3.0 which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressures below 785 psig is conservative.

approximately

See inserts on next page for revisions to BASES 2.2.1.9 and 2.2.1.10.

ATTACHMENT 2 TO PLA-4055

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

9. Turbine Stop Valve-Closure

Insert Paragraph "A"

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 5.5% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient assuming the turbine bypass valves operate.

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

Insert Paragraph "B"

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a faster closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section 15.2 of the Final Safety Analysis Report.

11. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

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

INSERT TO BASIS 2.2.1.9

(PARAGRAPH "A")

This function is not required when THERMAL POWER is below 30% of RATED THERMAL POWER. The Turbine Bypass System is sufficient at this low power to accommodate a turbine stop valve closure without the necessity of shutting down the reactor. This function is automatically bypassed at turbine first stage pressures less than the analytical limit of 147.7 psig, equivalent to THERMAL POWER of about 30% RATED THERMAL POWER. Turbine first stage pressure of 147.7 psig is equivalent to 22% of rated turbine load.

INSERT TO BASIS 2.2.1.10

(PARAGRAPH "B")

This function is not required when THERMAL POWER is below 30% of RATED THERMAL POWER. The Turbine Bypass System is sufficient at this low power to accommodate a turbine control valve closure without the necessity of shutting down the reactor. This function is automatically bypassed at turbine first stage pressures less than the analytical limit of 147.7 psig, equivalent to THERMAL POWER of about 30% RATED THERMAL POWER. Turbine first stage pressure of 147.7 psig is equivalent to 22% of rated turbine load.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Verifying the continuity of the explosive charge.
2. Determining that the available weight of sodium pentaborate is greater than or equal to 5500 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-2 by chemical analysis.*
3. Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

1224 c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to ~~1190~~ psig is met.

d. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.
2. **Demonstrating that all heat traced piping is unblocked by pumping from the storage tank to the test tank and then draining and flushing the discharge piping and test tank with demineralized water.
3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized.

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

NO CHANGE

NO CHANGE

TRIP SETPOINT #	ALLOWABLE VALUE #
$S \leq (0.58W + 59\%)\ T$ $S_{RB} \leq (0.58W + 50\%)\ T$	$S \leq (0.58W + 62\%)\ T$ $S_{RB} \leq (0.58W + 53\%)\ T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. The FRACTION OF LIMITING POWER DENSITY (FLPD) for SNP fuel is the actual LHGR divided by the LINEAR HEAT GENERATION RATE for APRM Setpoints limit specified in the CORE OPERATING LIMITS REPORT.

T is always less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as determined above, initiate corrective action within 15 minutes and adjust S and/ or S_{RB} to be consistent with the Trip Setpoint value* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

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* With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

\# See Specification 3.4.1.1.2.a for single loop operation requirements.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - ~~Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure until the function is automatically bypassed within 2 hours.~~

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

Initiate a reduction in THERMAL POWER within 15 minutes, and reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within 2 hours.

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon determination that a trip setpoint cannot be restored to within its specified value during performance of the CHANNEL CALIBRATION, the appropriate ACTION, 3.3.1a or 3.3.1b, shall be followed.

(b) This function is automatically bypassed when the reactor mode switch is in the Run position.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(g) This function is automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn.*

(j) ~~This function shall be automatically bypassed when turbine first stage pressure is less than 108 psig or 17% of the value of first stage pressure in psia at valves wide open (V.W.O.) steam flow, equivalent to THERMAL POWER of about 24% of RATED THERMAL POWER.~~

(k) Also actuates the EOC-RPT system.

This function shall not be automatically bypassed when turbine first stage pressure is greater than an allowable value of 136 psig.

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* Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

ATTACHMENT 2 TO PLA-4055

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥13.0 inches*	≥ 11.5 inches
2) Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
3) Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Manual Initiation	NA	NA
d. SGTS Exhaust Radiation - High	≤ 23.0 mR/hr	≤ 31.0 mR/hr
e. Main Steam Line Radiation - High	≤ 7.0 X full power background	≤ 8.4 X full power background
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Refuel Floor High Exhaust Duct Radiation - High	≤ 2.5 mR/hr	≤ 4.0 mR/hr
d. Railroad Access Shaft Exhaust Duct Radiation - High	≤ 2.5 mR/hr	≤ 4.0 mR/hr
e. Refuel Floor Wall Exhaust Duct Radiation - High	≤ 2.5 mR/hr	≤ 4.0 mR/hr
f. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Main Steam Line Radiation - High	≤ 7.0 X full power background	≤ 8.4 X full power background
c. Main Steam Line Pressure - Low	≥ 861 psig	≥ 841 psig
d. Main Steam Line Flow - High	≤ 107 psid 113 psid**	≤ 110 psid 121 psid**

TABLE 3.3.2-2 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
<u>MAIN STEAM LINE ISOLATION</u> (Continued)		
e. Condenser Vacuum - Low	≥ 9.0 inches Hg vacuum	≥ 8.8 inches Hg vacuum
f. Reactor Building Main Steam Line Tunnel Temperature - High	≤ 177°F	≤ 184°F
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	≤ 99°F	≤ 108°F*
h. Manual Initiation	NA	NA
i. Turbine Building Main Steam Line Tunnel Temperature - High	≤ 197°F	≤ 200°F
4. <u>REACTOR WATER CLEANUP SYSTEM ISOLATION</u>		
a. RWCU Δ Flow - High	≤ 60 gpm	≤ 80 gpm
b. RWCU Area Temperature - High	≤ 147° F or 131°F#	≤ 154°F or 137°F#
c. RWCU/Area Ventilation Δ Temperature - High	≤ 69°F or 40.8°F#	≤ 72°F or 43.8°F#*
d. SLCS Initiation	NA	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
f1. RWCU Flow - High	≤ ~~420~~ gpm 462	≤ ~~436~~ gpm 472
f2. Non-Regenerative Heat Exchanger Discharge Temperature - High	≤ 144°F	≤ 150°F
g. Manual Initiation	NA	NA
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>	138	143
a. RCIC Steam Line Δ Pressure - High	≤ ~~186~~" H_2O	≤ ~~196~~" H_2O
b. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig

* These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

ATTACHMENT 2 TO PLA-4055

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)		
d. RCIC Equipment Room Temperature - High	< 167°F	< 174°F
e. RCIC Equipment Room Δ Temperature - High	≤ 89°F	≤ 98°F*
f. RCIC Pipe Routing Area Temperature - High	< 167°F##	≤ 174°F##
g. RCIC Pipe Routing Area Δ Temperature - High	< 89°F##	≤ 98°F##*
h. RCIC Emergency Area Cooler Temperature - High	< 147°F	< 154°F
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	< 1.72 psig	< 1.88 psig
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	< ~~275~~ 387 inches H_2O	< ~~292~~ 399 inches H_2O
b. HPCI Steam Supply Pressure - Low	> 104 psig	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	< 10 psig	≤ 20 psig
d. HPCI Equipment Room Temperature - High	< 167°F	≤ 174°F
e. HPCI Equipment Room Δ Temperature - High	< 89°F	< 98°F
f. HPCI Emergency Area Cooler Temperature - High	< 147°F	< 154°F
g. HPCI Pipe Routing Area Temperature - High	< 167°F##	< 174°F##
h. HPCI Pipe Routing Area Δ Temperature - High	< 89°F##	< 98°F##*
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	< 1.72 psig	< 1.88 psig

*These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

ATTACHMENT 2 TO PLA-4055

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches*	≥ 11.5 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 98 psig	≤ 108 psig
c. RHR Flow - High	≤ 25,000 gpm	≤ 26,000 gpm
d. Manual Initiation	NA	NA
e. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

* See Bases Figure B 3/4 3-1.
\# Lower setpoints for TSH-G33-2N600 E, F and TDSH-G33-2N602 E, F.
\## 15 minute time delay.

** Initial value. Final value to be determined based on Power Uprate startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.

INCLUDED FOR REFERENCE. SEE FOOTNOTE "**" ON THE FOLLOWING PAGE.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	NA	M	Q	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	NA	M	Q	1, 2**, 3**
f. Reactor Building Main Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	NA	M	Q	1, 2, 3*
h. Manual Initiation	NA	R	NA	1, 2, 3
i. Turbine Building Main Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	S	M	R	1, 2, 3
b. RWCU Area Temperature - High	NA	M	Q	1, 2, 3
c. RWCU Area Ventilation Δ Temperature - High	NA	M	Q	1, 2, 3*
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
f.1. RWCU Flow - High#	S	M	R	1, 2, 3
f.2. Non-Regenerative Heat Exchanger Discharge Temperature - High#	S	M	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

* These trip functions need not be OPERABLE from October 16, 1989 to January 1, 1990.

\# For Unit 2 Cycle 3 operation, the Non-Regenerative Heat Exchanger Discharge Temperature - High Channel shall be OPERABLE in place of RWCU Flow - High Channel '81, [illegible]

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
d. HPCI Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3
f. HPCI Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
h. HPCI Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3****
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	Q	1, 2, 3
c. RHR Flow - High	S	M	R	1, 2, 3
d. Manual Initiation	NA	R	NA	1, 2, 3
e. Drywell Pressure - High	NA	M	R	1, 2, 3

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When ~~reactor steam dome pressure > 1043 psig and/or~~ any turbine stop valve is open.

*** When VENTING or PURGING the drywell per Specification 3.11.2.8.

**** This trip function need not be OPERABLE from October 19, 1989 to January 19, 1990.

ATTACHMENT 2 TO PLA-4055

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)
1. Turbine Stop Valve - Closure	2(b)
2. Turbine Control Valve-Fast Closure	2(b)

(a)A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

(b)~~This function shall be automatically bypassed when turbine first stage pressure is less than 100 psig or 17% of the value of first stage pressure in psia at valves wide open (V.W.O) steam flow. This value is equivalent to THERMAL POWER of about 24% of RATED THERMAL POWER.~~

This function shall not be automatically bypassed when turbine first stage pressure is greater than an allowable value of 136 psig.

See insert on next page for revisions to trip functions 1.a and 2.a

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
~~a. Upscale##~~	~~≤ 0.66 W + 42%~~	~~≤ 0.66 W + 45%~~
b. Inoperative	NA	NA
c. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 of divisions full scale
2. APRM		
~~a. Flow Biased Neutron Flux - Upscale##~~	~~≤ 0.58 W + 50%*~~	~~≤ 0.58 W + 53%*~~
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
	No CHANGE	No CHANGE
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< 2 x 10^5 cps	< 4 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 0.7 cps**	≥ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	≤ 44 gallons	≤ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW	114	117
a. Upscale	≤ ~~108~~/125 divisions of full scale	≤ ~~111~~/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.
**Provided signal-to-noise ratio is ≥ 2. Otherwise, 3 cps as trip setpoint and 2.8 cps for allowable value.
##See Specification 3.4.1.1.2.a for single loop operation requirements.

INSERT TO TABLE 3.3.6-2

1. ROD BLOCK MONITOR

a. Upscale ##	≤ 0.63 W + 41%	≤ 0.63 W + 43%

2. APRM

a. Flow Biased Neutron Flux Upscale##		
1) Flow Biased	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
2) High Flow Clamped	≤ 108% of RATED THERMAL POWER	≤ 111% of RATED THERMAL POWER

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

ACTION: (Continued)

2. If Region II of Figure 3.4.1.1.1-1 is entered and greater than or equal to 50% of the required LPRM upscale alarms OPERABLE, immediately exit the region by:

   a) inserting a predetermined set of high worth control rods, or

   b) increasing core flow.

3. With less than 50% of the required LPRM upscale alarms OPERABLE, follow ACTION a.1.d upon entry into Region II of Figure 3.4.1.1.1-1.

b. In OPERATIONAL CONDITION 2 with no reactor coolant system recirculation loops in operation, return at least one reactor coolant system recirculation loop to operation, or be in HOT SHUTDOWN within the next 6 hours.

c. With any pump discharge valve not OPERABLE remove the associated loop from operation, close the valve and comply with the requirements of Specification 3.4.1.1.2.

d. With any pump discharge bypass valve not OPERABLE close the valve and verify closed at least once per 31 days.

4.4.1.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

a core flow of 109.5 million lbm/hr

4.4.1.1.1.2 Each pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to ~~102.5~~ and ~~105%,~~ respectively, ~~of rated core flow,~~ at least once per 18 months.

4.4.1.1.1.3 At least 50% of the required LPRM upscale alarms shall be determined OPERABLE by performance of the following on each LPRM upscale alarm:

1) CHANNEL FUNCTIONAL TEST at least once per 92 days, and

2) CHANNEL CALIBRATION at least once per 184 days.

110.5 million lbm/hr

**If not performed within the previous 31 days.

REPLACE WITH New Figure
(ATTACHED)

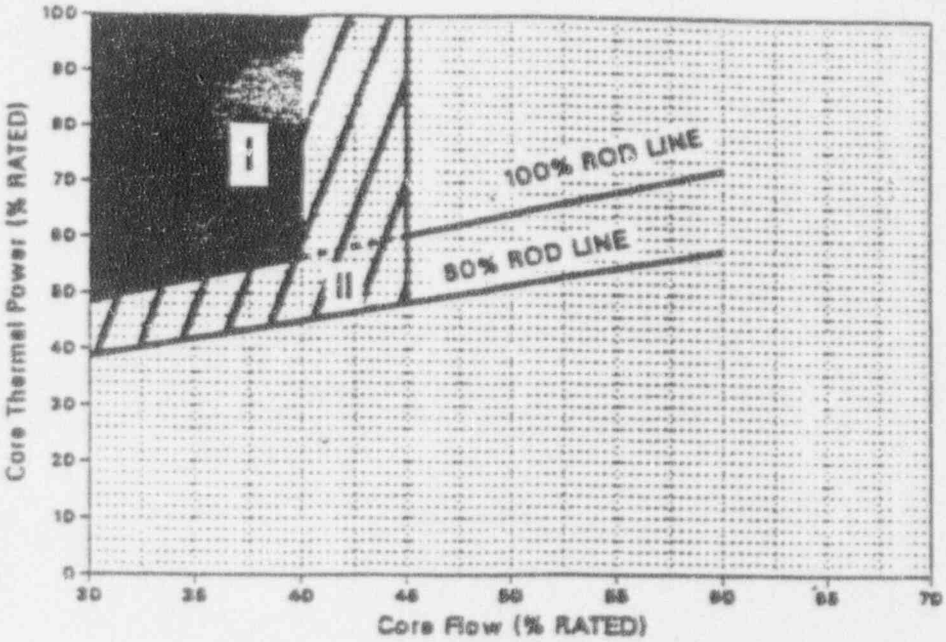


Figure 3.4.1.1.1-1
THERMAL POWER RESTRICTIONS

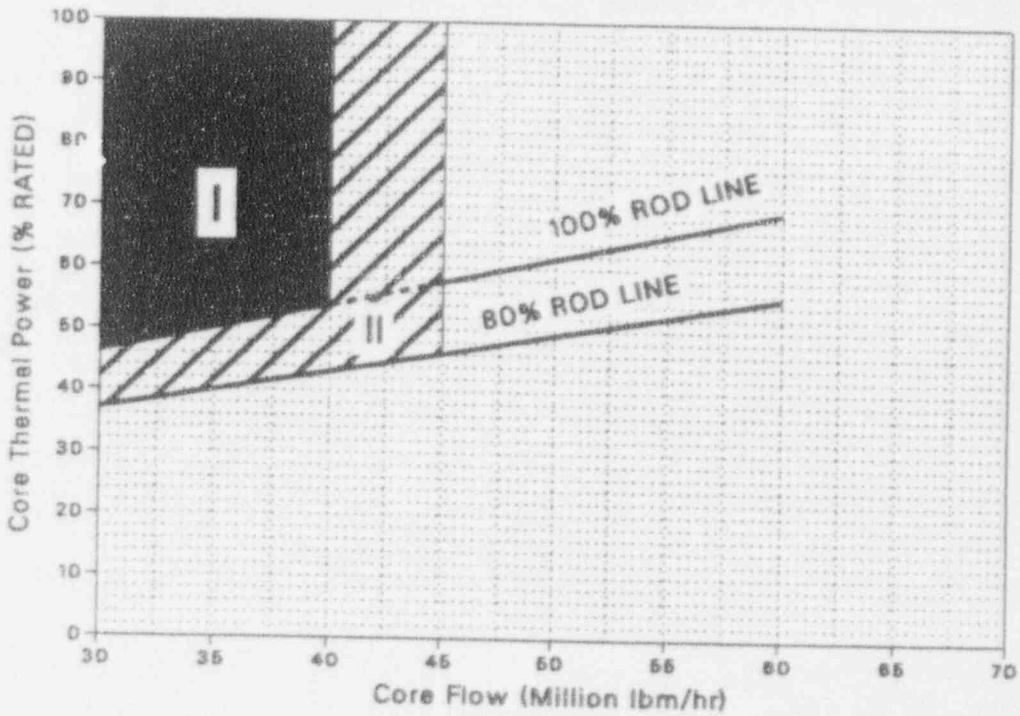


Figure 3.4.1.1.1-1
THERMAL POWER STABILITY RESTRICTIONS

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed ≤ 80% of the rated pump speed and the reactor at a THERMAL POWER/core flow condition outside of Regions I and II of Figure 3.4.1.1.1-1, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.
2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

NO CHANGE

Trip Setpoint	Allowable Value
≤ 0.58W + 54%	≤ 0.58W + 57%

3. Specification 3.2.2: the APRM Setpoints shall be as follows:

NO CHANGE

Trip Setpoint	Allowable Value
$S \le (0.58W + 54\%)\ T$ $S_{RB} \le (0.58W + 45\%)\ T$	$S \le (0.58W + 57\%)\ T$ $S_{RB} \le (0.58W + 48\%)\ T$

4. Specification 3.2.3: The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the applicable Single Loop Operation MCPR limit as specified in the CORE OPERATING LIMITS REPORT.

5. Specification 3.2.4: The LINEAR HEAT GENERATION RATE (LHGR) shall be less than or equal to the applicable Single Loop Operation LHGR limit as specified in the CORE OPERATING LIMITS REPORT.

~~5.~~ 6. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

	Trip Setpoint	Allowable Value
a. RBM - Upscale	≤ ~~0.66W + 36%~~ .63W+35%	≤ ~~0.66W + 39%~~ .63W+37%
	Trip Setpoint	**Allowable Value**
b. APRM - Flow Biased	≤ 0.58W + 45%	≤ 0.58W + 48%

NO CHANGE

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*+, except during two loop operation.#

ACTION:

a. In OPERATIONAL CONDITION 1:
   1. With
      a) no reactor coolant system recirculation loops in operation, or
      b) Region I of Figure 3.4.1.1.1-1 entered, or
      c) Region II of Figure 3.4.1.1.1-1 entered and core thermal hydraulic instability occurring as evidenced by:

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

f. With any pump discharge bypass valve not OPERABLE close the valve and verify closed at least once per 31 days.

SURVEILLANCE REQUIREMENTS

4.4.1.1.2.1 Upon entering single loop operation and at least once per 24 hours thereafter, verify that the pump speed in the operating loop is ≤ 80% of the rated pump speed.

4.4.1.1.2.2 At least 50% of the required LPRM upscale alarms shall be determined OPERABLE by performance of the following on each LPRM upscale alarm.

1) CHANNEL FUNCTIONAL TEST at least once per 92 days, and

2) CHANNEL CALIBRATION at least once per 184 days.

4.4.1.1.2.3 Within 15 minutes prior to either THERMAL POWER increase resulting from a control rod withdrawal or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30%**** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50%**** of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b.## ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c.## ≤ 50°F between the reactor coolant within the loop not in operation and operating loop.

4.4.1.1.2.4 The pump discharge valve and bypass valve in both loops shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

a core flow of 109.5 million lbm/hr

4.4.1.1.2.5 The pump MG set scoop tube electrical and mechanical stops shall be demonstrated OPERABLE with overspeed setpoints less than or equal to ~~102.5%~~ and ~~105%~~, respectively, ~~of rated core flow,~~ at least once per 18 months.

110.5 million lbm/hr

4.4.1.1.2.6 During single recirculation loop operation, all jet pumps, including those in the inoperable loop, shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:###

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established single recirculation pump speed-loop flow characteristics.

SURVEILLANCE REQUIREMENTS (Continued)

b. The indicated total core flow differs by more than 10% from the established total core flow value from single recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patterns by more than 10%.

4.4.1.1.2.7 The SURVEILLANCE REQUIREMENTS associated with the specifications referenced in 3.4.1.1.2a shall be followed.

* See Special Test Exception 3.10.4.

** If not performed within the previous 31 days.

Power Uprate

**** Initial value. Final value to be determined based on startup testing. Any required change to this value shall be submitted to the Commission within 90 days of ~~test~~ completion.

See Specification 3.4.1.1.1 for two loop operation requirements.

This requirement does not apply when the loop not in operation is isolated from the reactor pressure vessel.

At least once per 18 months (555 days), data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the performance of required surveillances.

+ The LPRM upscale alarms are not required to be OPERABLE to meet this specification in OPERATIONAL CONDITION 2.

Power Uprate startup test program

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed mismatch shall be maintained within:

a. 5% of each other with core flow greater than or equal to ~~75% of rated core flow.~~ 75 million lbm/hr.

b. 10% of each other with core flow less than ~~75% of rated core flow.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* when both recirculation loops are in operation.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION required by Specification 3.4.1.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation pump speed mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 10 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:* ** 12

~~2 safety-relief valves @ 1146 psig ±1%~~

2 → 4 safety-relief valves @ 1175 psig ±1%

~~4 safety-relief valves @ 1185 psig ±1%~~

6 → 2 safety-relief valves @ 1195 psig ±1%

8 → 3 safety-relief valves @ 1205 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool water temperature is 105°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.25 of the full open noise level# by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a

b. Calibration in accordance with procedures prepared in conjunction with its manufacturer's recommendations at least once per 18 months.##

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

**Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

#Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

##The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 1 gpm leakage at a reactor coolant system pressure of ~~1000~~ 1035 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b. and/or c., above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm pressure at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than ~~1040~~ psig. 1050

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

1050

With the reactor steam dome pressure exceeding ~~1040~~ psig, reduce the pressure to less than ~~1040~~ psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

1050

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than ~~1040~~ psig at least once per 12 hours.

1050

*Not applicable during anticipated transients.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the CSS, the LPCI system, and the HPCI system:

a) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water by:

1. Venting at the high point vents

2. Performing a CHANNEL FUNCTIONAL TEST of the condensate transfer pump discharge low pressure alarm instrumentation.

b) Verifing that each valve, manual, power-operated, or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct** position.

2. For the CSS, performance of a CHANNEL FUNCTIONAL TEST of the core spray header ΔP instrumentation.

3. For the LPCI system, verifying that at least one LPCI system subsystem cross-tie valve is closed with power removed from the valve operator.

4. For the HPCI system, verifying that the pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a total flow of at least 6350 gpm against a test line pressure of $\geq$ 282 psig, corresponding to a reactor vessel steam dome pressure of $\geq$ 105 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 12,200 gpm against a test line pressure of $\geq$ 222 psig, corresponding to a reactor vessel to primary containment differential pressure $\geq$ 20 psid.

3. The HPCI pump develops a flow of at least 5000 gpm against a test line pressure of $\geq$ ~~1260~~ 1140 psig when steam is being supplied to the turbine at 920, +140, - 20 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

**Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

See insert on next page for Revision to Bases 3/4.3.4.

INSTRUMENTATION

BASES

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Insert Paragraph "A"

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 175 ms.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

INSERT TO BASIS 3.4.3.4

(PARAGRAPH "A")

This function is not required when THERMAL POWER is below 30% of RATED THERMAL POWER. The Turbine Bypass System is sufficient at this low power to accommodate a turbine stop valve or control valve closure without the necessity of tripping the reactor recirculation pumps. This function is automatically bypassed at turbine first stage pressures less than the analytical limit of 147.7 psig, equivalent to THERMAL POWER of about 30% RATED THERMAL POWER. Turbine first stage pressure of 147.7 psig is equivalent to 22% of rated turbine load.

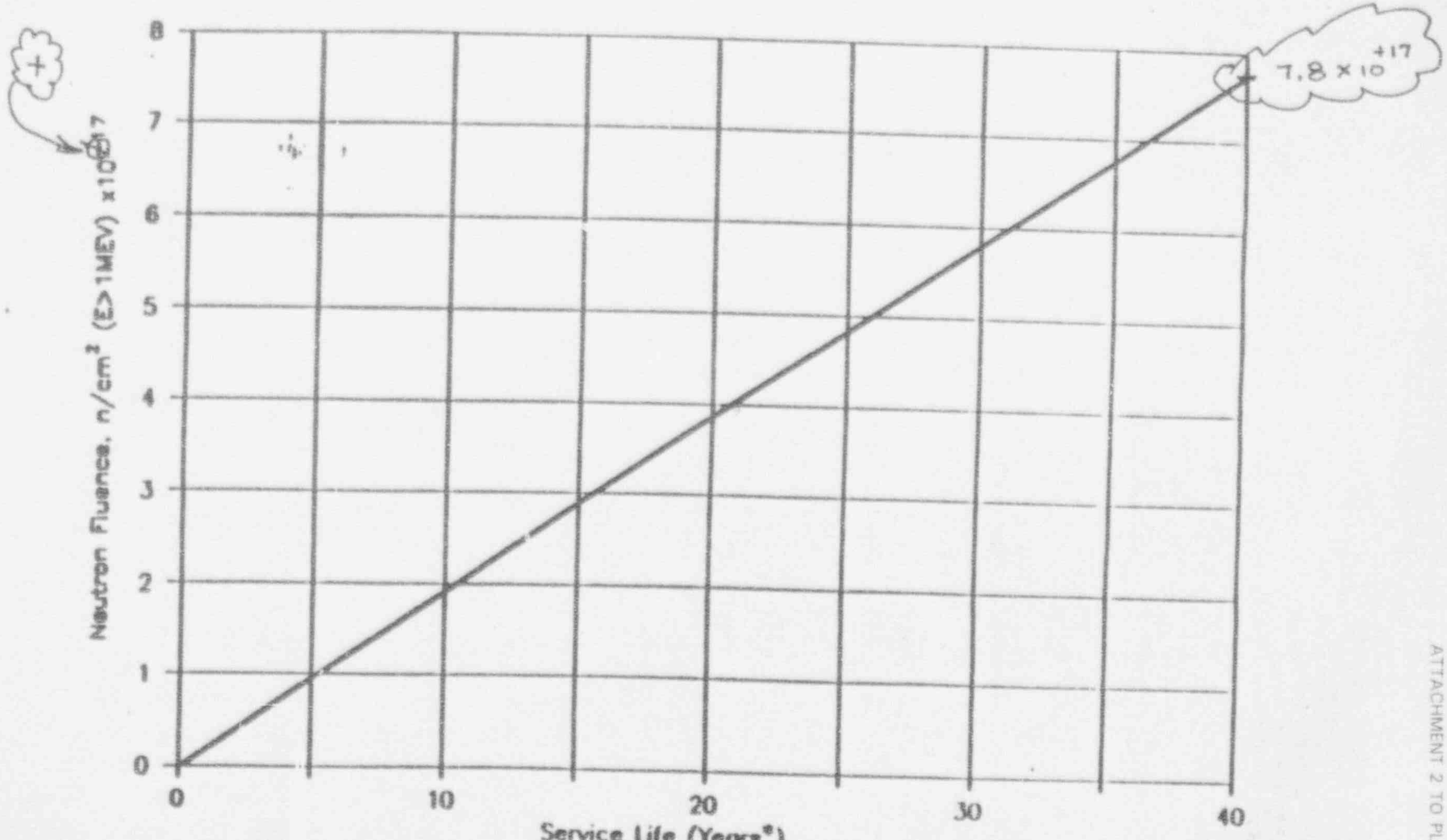
BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

Limiting Beltline Component	Weld Seam I.D. or Mat'l. Type	Heat/Slab or Heat/Lot	CU(%)	Ni(%)	Starting RT_{NDT} (°F)	ΔRT_{NDT} (°F)*	Min. Upper Shelf (Ft-Lbs)	Max. RT_{NDT} (°F)
Plate	SA-533 GR B CL.1	~~6C1053/1~~ C2421-3	~~0.10~~ 0.13	~~0.58~~ 0.68	~~+10~~ -10	~~40~~ 56.7	N/A	~~+50~~ 46.7
Weld	N/A	624263/ E204A27A	0.06	0.89	-20	50	N/A	+30

NOTE: * These values are given only for the benefit of calculating the 32 EFPY RT_{NDT} Per R.G. 1.99 Rev. 2.

NON-BELTLINE COMPONENT	MATERIAL TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring #5	SA-533 GR B Cl.1	All	+10
Bottom Head Dome	"	C0462	+20
Bottom Head Torus	"	C0472	+10
Top Head Side Plates	"	C0473-1	+10
Top Head Flange	SA-508, Cl.2	125H446	+10
Vessel Flange	"	2L2393	+10
Feedwater Nozzle	"	Q2Q62W	-10
Steam Outlet Nozzle	"	Q2Q64W	+30
Weld	Bottom Head Flanges to Shell Top Head Other Non-Beltline	All All All	-20 -20 0
Closure Studs	SA-540 GR B24	All	Meet requirements of 45 ft-lbs and 25 mils lateral expansion at +10°F



Fast Neutron Fluence (E>1 Mev) at I.D. Surface as a Function of Service Life*

Bases Figure B 3/4.4.6-1

*At 90% of RATED THERMAL POWER and 90% availability

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS) is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI mode of the RHR system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the automatic depressurization system (ADS).

The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CSS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Two subsystems, each with two pumps, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HCPI system continues to operate until reactor vessel pressure is below the pressure at which CS system operation or LPCI mode of the RHR system operation maintains core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5000 gpm at reactor pressures between 1157 and 150 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

1187

CONTAINMENT SYSTEMS

BASES

3/4 6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 53 psig during primary system blowdown from full operating pressure.

1053 psia

The suppression chamber water provides the heat sink for the reactor coola system energy release following a postulated rupture of the system. The suppre sion chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~~1055 psig~~. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 53 psig, the suppression chamber maximum pressure. The design volum of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 45.0 psi which is below the design pressure of 53 psig. Maximum water volume of 133,540 ft^3 results in a downcomer submergence of 12 feet and the minimum volum of 122,410 ft^3 results in a submergence approximately 24 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blow-down tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppressior chamber water temperature of 90°F results in a water temperature of approximately 128°F immediately following blowdown which is below the 170°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure durinc the accident injection phase. If both RHR loops are used for containment coolir there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of non-enriched or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of Zirconium alloy filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or analyses to comply with all fuel safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in non-limiting core regions. Each fuel rod shall have a nominal active fuel length of 150 inches. Reload fuel shall have a maximum average enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C), and/or Hafnium metal. The control rod shall have a nominal axial absorber length of 143 inches. Control rod assemblies shall be limited to those control rod designs approved by the NRC for use in BWRs.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 528°F.

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TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

551°F

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 546°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 hydrostatic pressure and leak tests	Pressurized to $\geq$ 930 psig and $\leq$ 1250 psig.

CORE OPERATING LIMITS REPORT (Continued)

14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.

15. ~~XN-NF-80-19(A), Volumes 2, 2A, 2B, and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, Inc., September 1982.~~

16. ~~XN-NF-CC-33(A), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Exxon Nuclear Company, Inc., November 1975.~~

17. ~~XN-NF-82-07(A), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., November 1982.~~

18. ~~XN-NF-84-117(P), "Generic LOCA Break Spectrum Analysis: BWR 3 and 4 with Modified Low Pressure Coolant Injection Logic," Exxon Nuclear Company, Inc., December 1984.~~

19. ~~XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9x9 Fuel," Exxon Nuclear Company, Inc., May 1986.~~

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

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

NRC SER ON PP&L POWER UPRATE LTR (later)