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May 26, 1978

Docket No. 50-335

Florida Power and Light Company
Advanced Systems & Technology
ATTN: Mr. Robert E. Uhrig
Vice President
P. O. Box 529100
Miami, Florida 33152

Gentlemen:

The Commission has issued the enclosed Amendment No. 27 to Facility Operating License No. DPR-67 for St. Lucie Plant, Unit No. 1. The amendment consists of changes to the license and its appended Technical Specifications in accordance with your requests dated March 3 and 22, April 4, 5, 12 and 28, and May 1, 1978. Your applications were supplemented by information dated April 17 and 21, and May 11, 19, 22, and 23, 1978.

The enclosed amendment consists of:

(1) Technical Specification (TS) changes resulting from the analyses of Cycle 2 reload fuel;

(2) TS changes to include consideration of a new water hole peaking factor;

(3) Approval to operate with sleeved Control Element Assembly (CEA) guide tubes;

(4) Deletion of certain license requirements that have been completed;

(5) TS changes authorizing the removal of all part length control element assemblies; ~~and~~

(6) Resistance Temperature Detector testing requirements; and

(7) Extension of time to install neutron shielding.

GD

OFFICE ➤						
SURNAME ➤						
DATE ➤						

NRC FORM 318 (9-76) NRCM 0240

☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 27 to DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

No legal objection

	C-PSB:DOR	C-RS:DOR	C-EB:DOR	EEB:DOR
	WButler	RBaer	LShao	LBarrett
	5/25/78	5/25/78	5/28/78	5/26/78

OFFICE➤	ORB#4:DOR	ORB#4:DOR	OELD	C-ORB#4:DOR	STSS	AD-E&P:DOR
SURNAME➤	RIngram:dn	PErickson	Tourtellotte	RWReid	JMcGough	BGrimes
DATE➤	5/25/78	5/25/78	5/ /78	5/25/78	5/ /78	5/26/78

Florida Power & Light Company

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ccw/enclosures and incoming:
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as suppl. 4/17&21, 5/11,19,22,&23/78
Bureau of Intergovernmental
Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. DPR-67

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

   A. The applications for amendment by Florida Power and Light Company (the licensee) dated March 3 and 22, April 4, 5, 12 and 28, and May 1, 1978, as supplemented, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, 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 by the following additional changes to Facility Operating License No. DPR-67:

   A. Revise paragraph 2.C.(2) in its entirety to read as follows:

      (2) Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 27 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   B. Revise condition D of Enclosure 1 appended to the license in its entirety to read as follows:

      Additional neutron shielding as described in letter from Florida Power and Light Company dated November 29, 1976, as revised by letter dated August 3, 1977, shall be installed during the next scheduled reactor shutdown of sufficient duration for the installation work. The shield shall be installed no later than the next refueling shutdown, however.

   C. Delete in their entirety conditions I., M., N., O., and P. of Enclosure 1 appended to the license.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 26, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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B 2-1
B 2-4
B 2-6
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

BASES

BASES

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals.

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHAL POWER TILT} = \max\left[\frac{\text{Power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (µCi/gram) which alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad\qquad Y_I = AY_E + B$$

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.24 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

SHIELD BUILDING INTEGRITY

1.25 SHIELD BUILDING INTEGRITY shall exist when:

1.25.1 Each door is closed except when the access opening is being used for normal transit entry and exit, and

1.25.2 The shield building ventilation system is OPERABLE.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.29 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective 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 instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 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 INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1)		
Four Reactor Coolant Pumps Operating	$\leq$ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.	$\leq$ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	$\leq$ 2400 psia	$\leq$ 2400 psia
5. Containment Pressure - High	$\leq$ 3.9 psig	$\leq$ 3.9 psig
6. Steam Generator Pressure - Low (2)	$\geq$ 485 psig	$\geq$ 485 psig
7. Steam Generator Water Level -Low	$\geq$ 37.0% Water Level - each steam generator	$\geq$ 37.0% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is 370,000 gpm.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 800 psig	$\geq$ 800 psig
11. Rate of Change of Power - High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 585 psig; bypass shall be automatically removed at or above 585 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

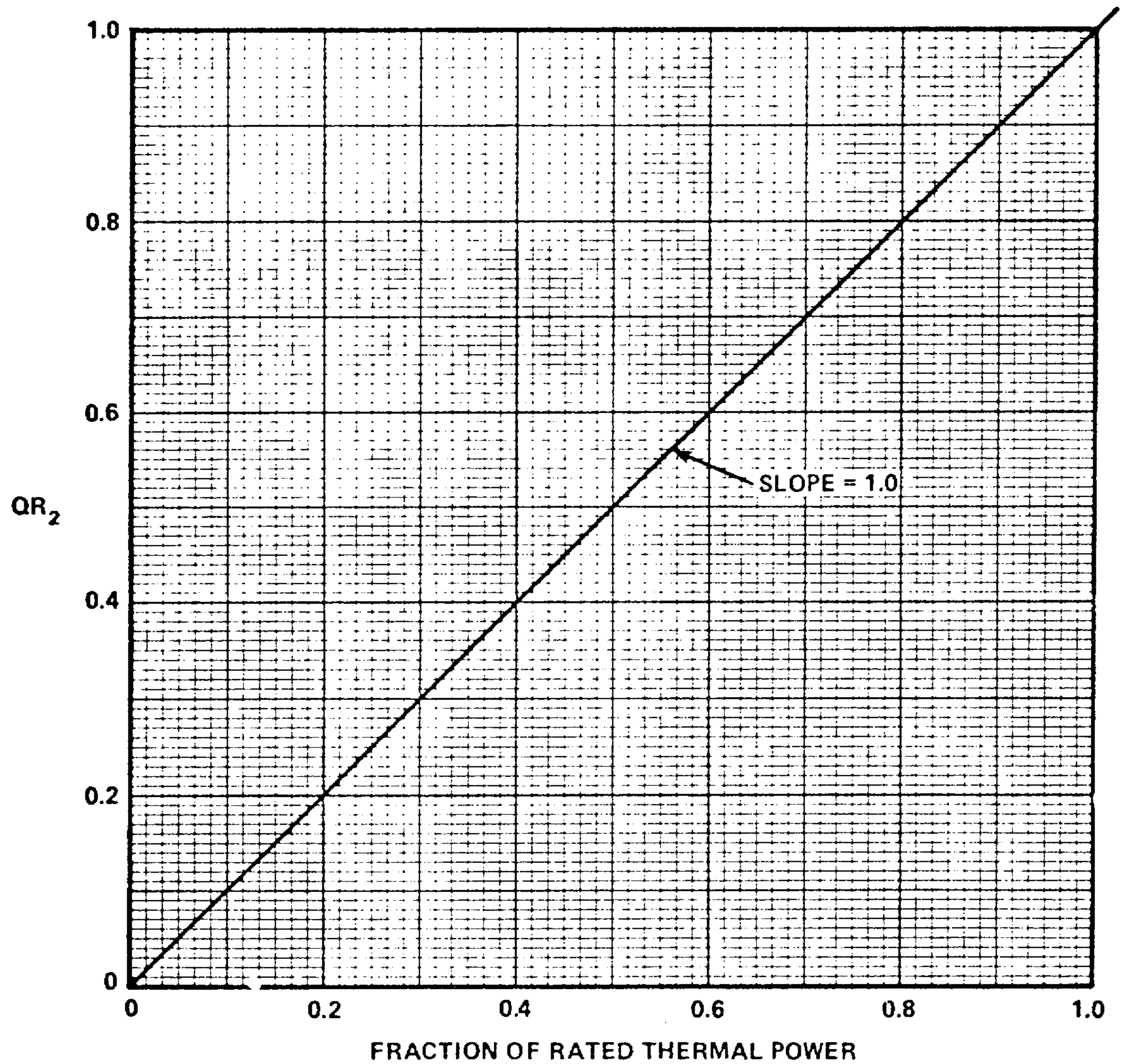

FIGURE 2.2-1

Local Power Density – High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

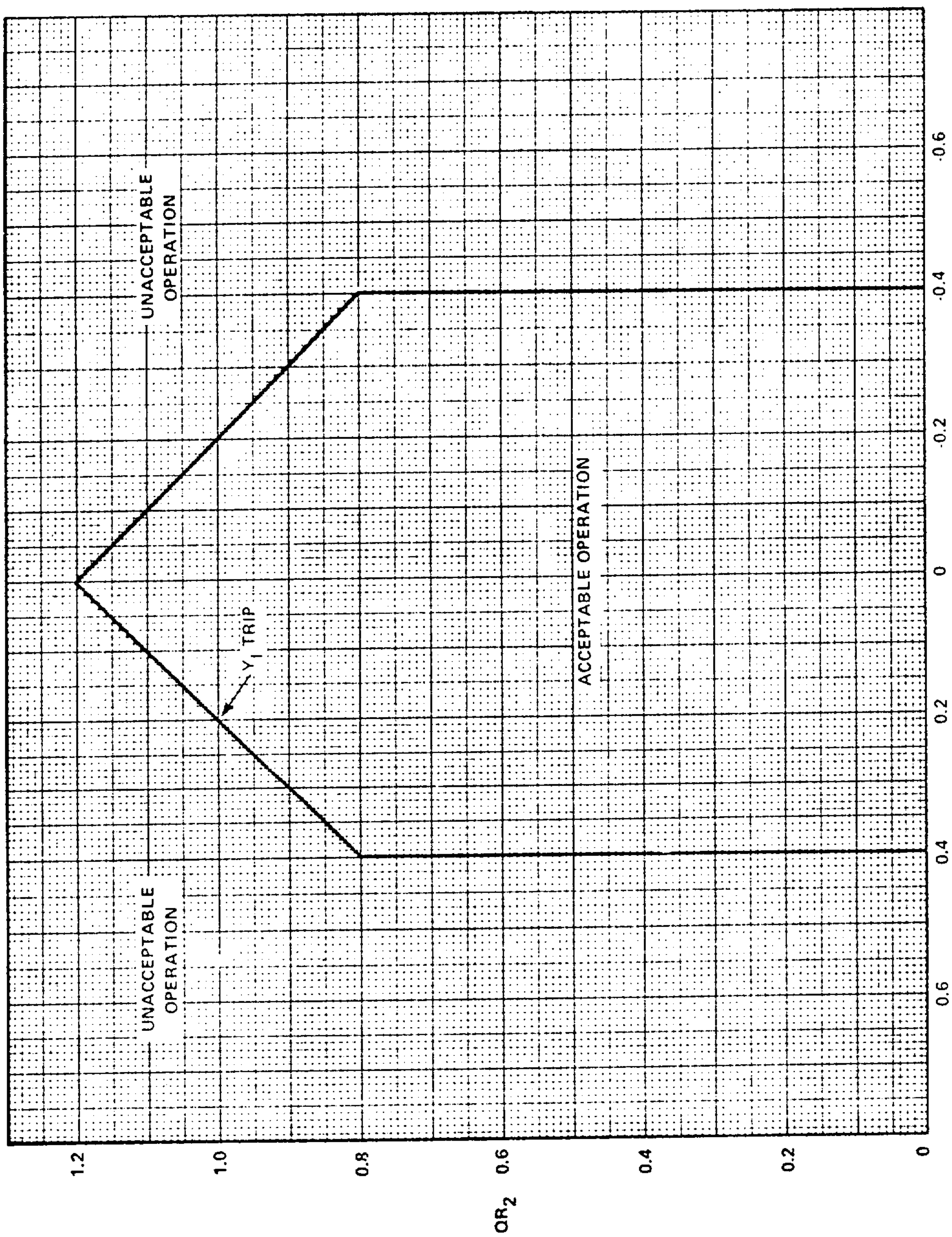

Figure 2.2-2 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2 (QR_2 VERSUS Y_I)

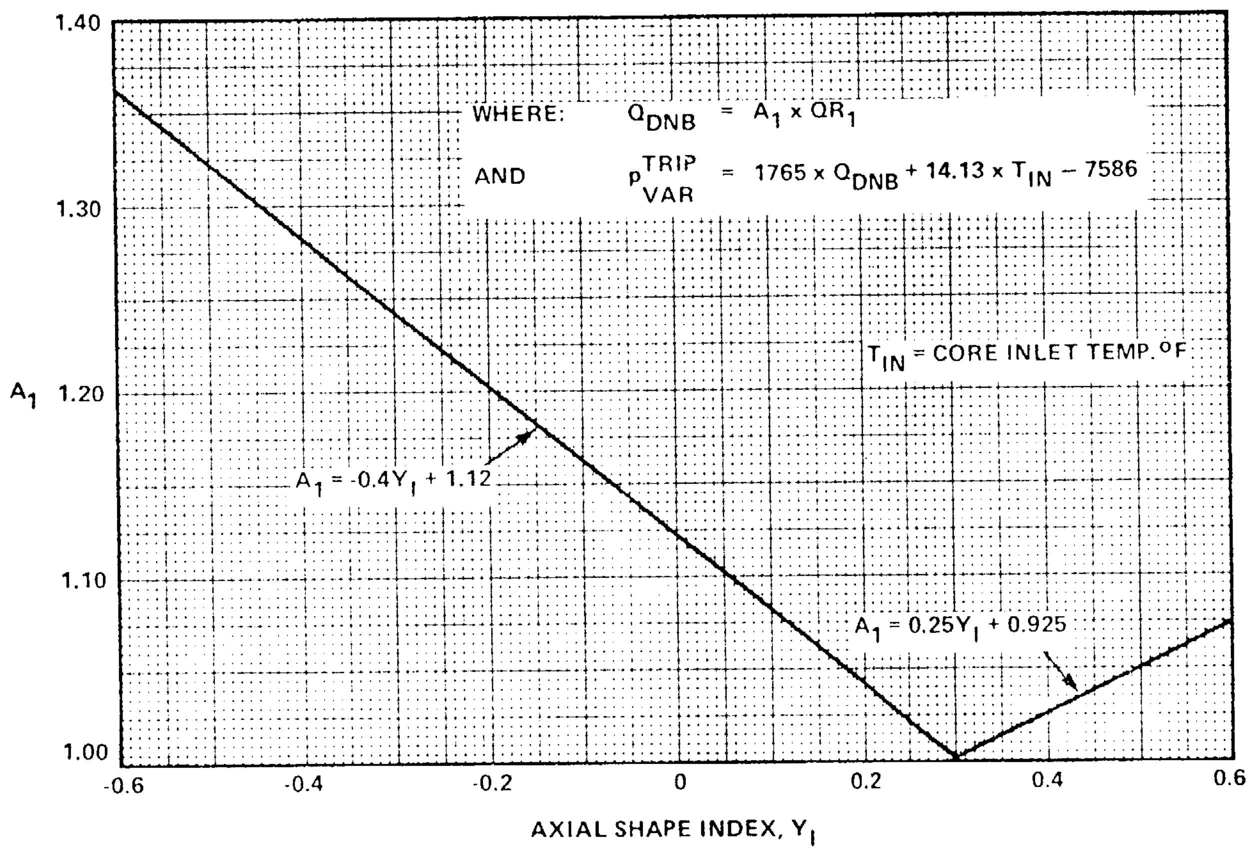

FIGURE 2.2-3
Thermal Margin/Low Pressure Trip Setpoint
Part 1 (Y_I Versus A_1)

WHERE: $A_1 \times QR_1 = Q_{DNB}$

AND $P_{VAR}^{TRIP} = 1765 \times Q_{DNB} + 14.13\,T_{IN} - 7586$

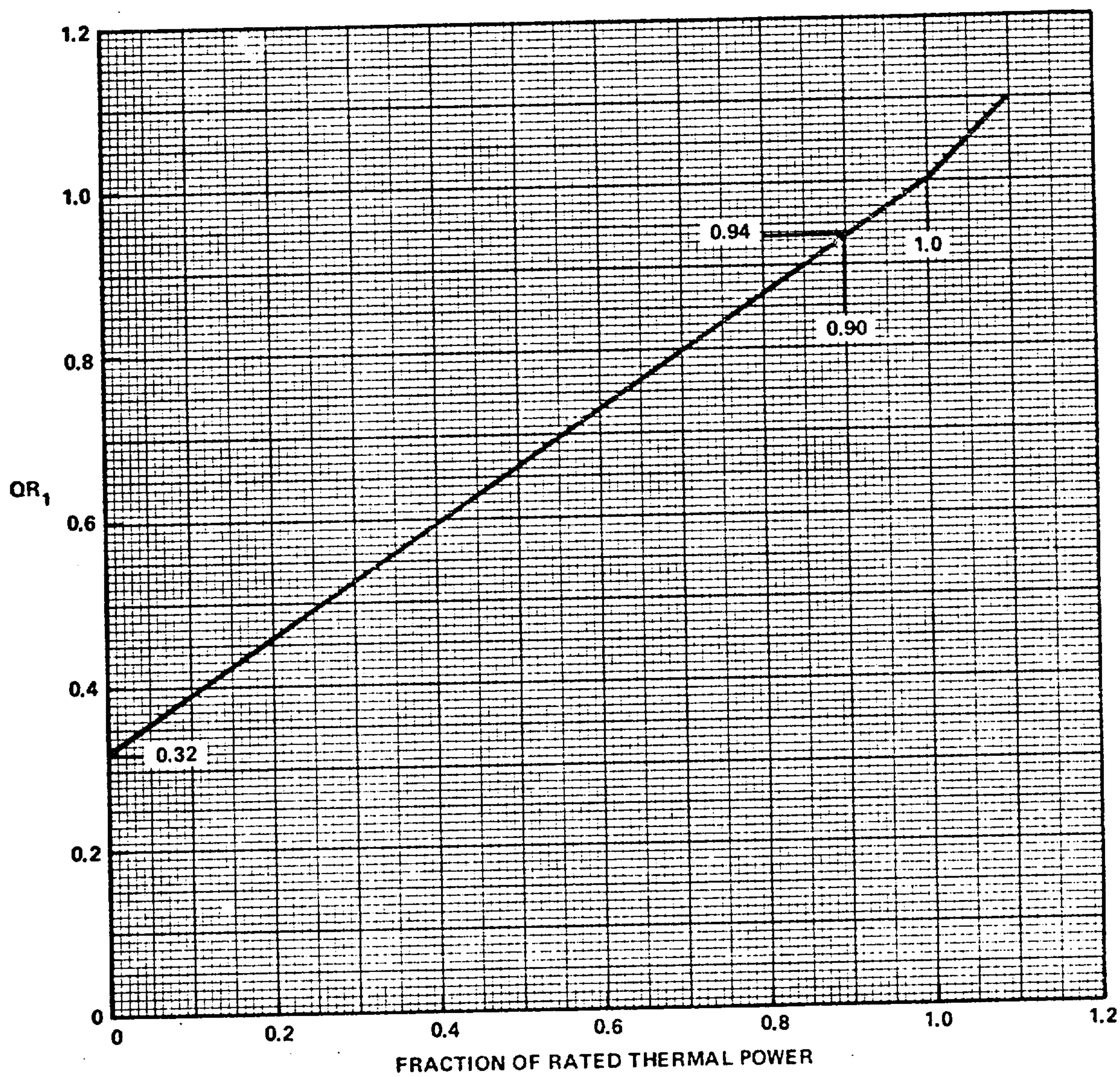

FIGURE 2.2-4

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER Versus QR_1)

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.30 for the family of axial shapes and corresponding radial peaks shown in Figure B 2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.1-1. The area of safe operation is below and to the left of these lines.

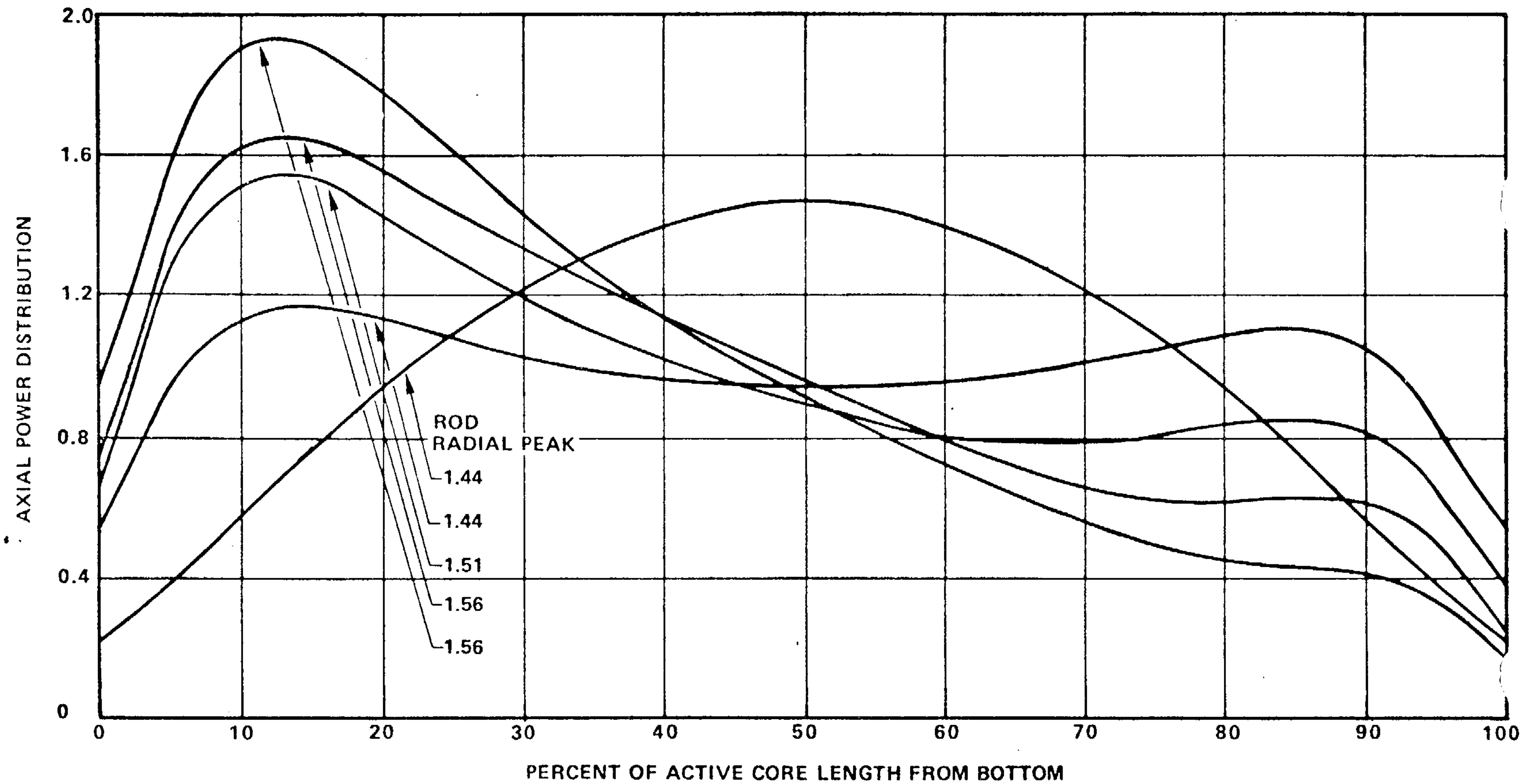

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.30 and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

BASES

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. 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.

Manual Reactor Trip

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

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit operation of the reactor at reduced power if one or two

Reactor Coolant Flow-Low (Continued)

reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.30 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip in initiated concurrently with a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 485 psig is sufficiently below the full-load operating point of 800 psig so as not

Steam Generator Pressure-Low (Continued)

to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded due to loss of steam generator heat sink. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.30.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty; and a further allowance of 74 psia to compensate for pressure measurement error and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 74 psia allowance is made up of a 22 psia pressure measurement allowance and a 52 psia time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuring transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 3.3% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 3.3% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.3% $\Delta k/k$:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA 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 CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor 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 $\pm$ 1.0% $\Delta k/k$ 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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.2×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.2×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each refueling.

b. At any THERMAL POWER, within 7 EFPD after initially reaching a RATED THERMAL POWER equilibrium boron concentration.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 At least the above required boric acid pump(s) shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 75 psig, and

c. Verifying pump operation for at least 15 minutes.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and one associated heat tracing circuit with a minimum contained volume of 1660 gallons of 8 weight percent boron.

b. The refueling water tank with:

   1. A minimum contained volume of 125,000 gallons,

   2. A minimum boron concentration of 1720 ppm, and

   3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration of the water,

   2. Verifying the water level of the tank, and

   3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the site ambient air temperature is < 40°F.

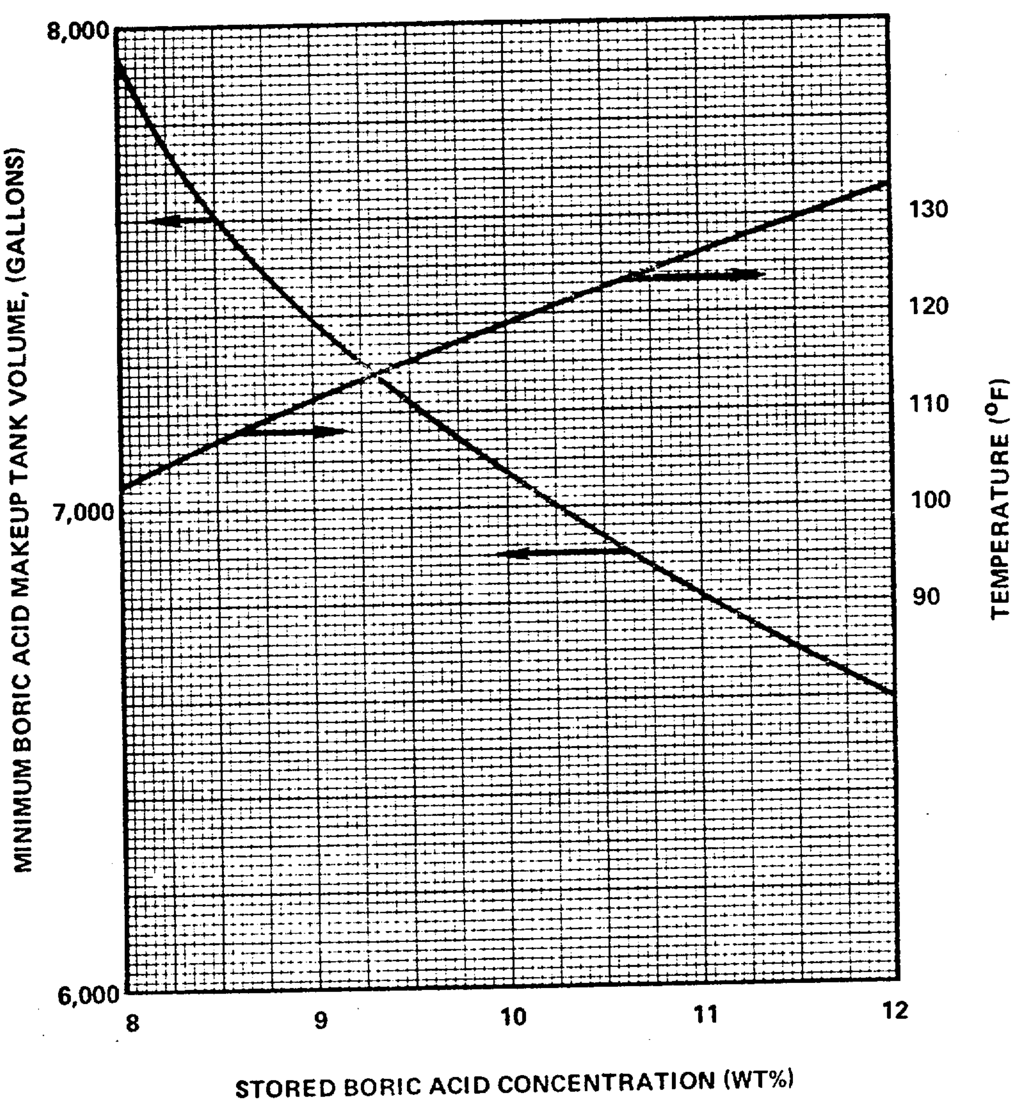

Figure 3 1-1 Minimum Boric Acid Makeup Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following three borated water sources shall be OPERABLE:

a. Two boric acid makeup tanks and one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A minimum contained volume of 371,800 gallons of water,
2. A minimum boron concentration of 1720 ppm,
3. A maximum solution temperature of 100°F,
4. A minimum solution temperature of 55°F when in MODES 1 and 2, and
5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least one per 7 days by:

1. Verifying the boron concentration in each water source,

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the water level in each water source, and

3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

FULL LENGTH CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Block Circuit and all full length (shutdown and regulating) CEAs shall be OPERABLE with each CEA of a given group positioned within 7.5 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, be in HOT STANDBY within 6 hours.

b. With the CEA Block Circuit inoperable, within 6 hours either:

   1. Restore the CEA Block Circuit to OPERABLE status, or

   2. Fully withdraw all CEAs in groups 3, 4, 5 and 6 and withdraw the CEAs in group 7 to less than 5% insertion and place and maintain the CEA drive system mode switch in either the "Manual" or "Off" position, or

   3. Be in at least HOT STANDBY.

c. With one full length CEA inoperable (unless immovable as a result of excessive friction or mechanical interference or known to be untrippable) but within its above specified alignment requirements, operation in MODES 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year.

d. With one or more full length CEAs misaligned from any other CEAs in its group by more than 7.5 inches but less than 15 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

*See Special Test Exceptions 3.10.2 and 3.10.5.

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LIMITING CONDITION FOR OPERATION

3.1.3.3 All shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within $\pm$ 2.25 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Deleted.

b. With a maximum of one reed switch position indicator channel per group or one (except as permitted by ACTION item d. below) pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, within 6 hours either:

   1. Restore the inoperable position indicator channel to OPERABLE status, or

   2. Be in HOT STANDBY, or

   3. Reduce THERMAL POWER to $\leq$ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

      a) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; or

REGULATING CEA INSERTION LIMITS (Continued)

LIMITING CONDITION FOR OPERATION

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

   2. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Power Dependent Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits shall be determined at least once per 24 hours.

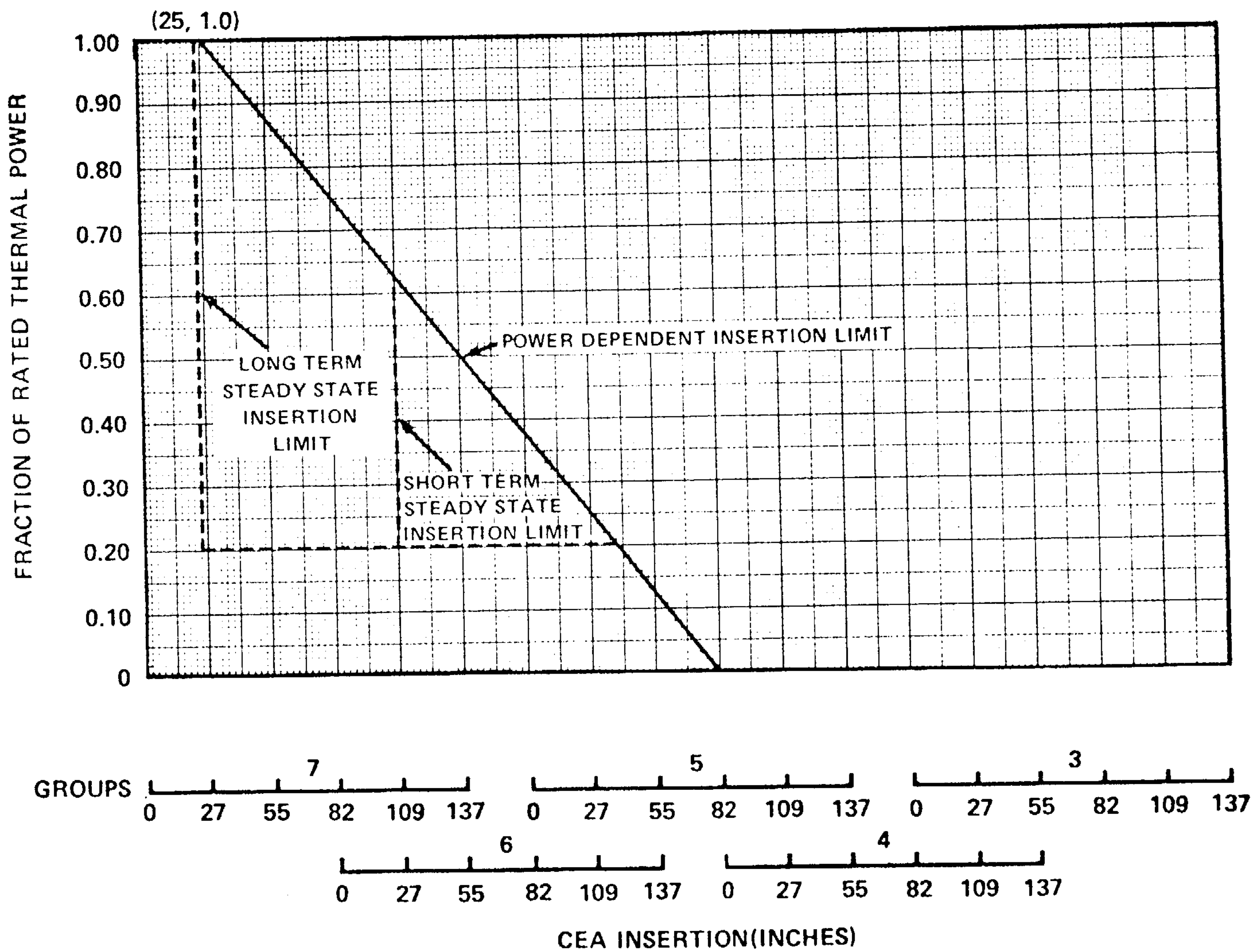

Figure 3.1-2 CEA Insertion Limits vs THERMAL POWER with 4 Reactor Coolant Pumps Operating

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

b. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

SURVEILLANCE REQUIREMENTS (Continued)

M x N

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by Figure 3.2-3.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

   1. Flux peaking augmentation factors as shown in Figure 4.2-1,

   2. A measurement-calculational uncertainty factor of 1.058,

   3. An engineering uncertainty factor of 1.03,

   4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

   5. A THERMAL POWER measurement uncertainty factor of 1.02.

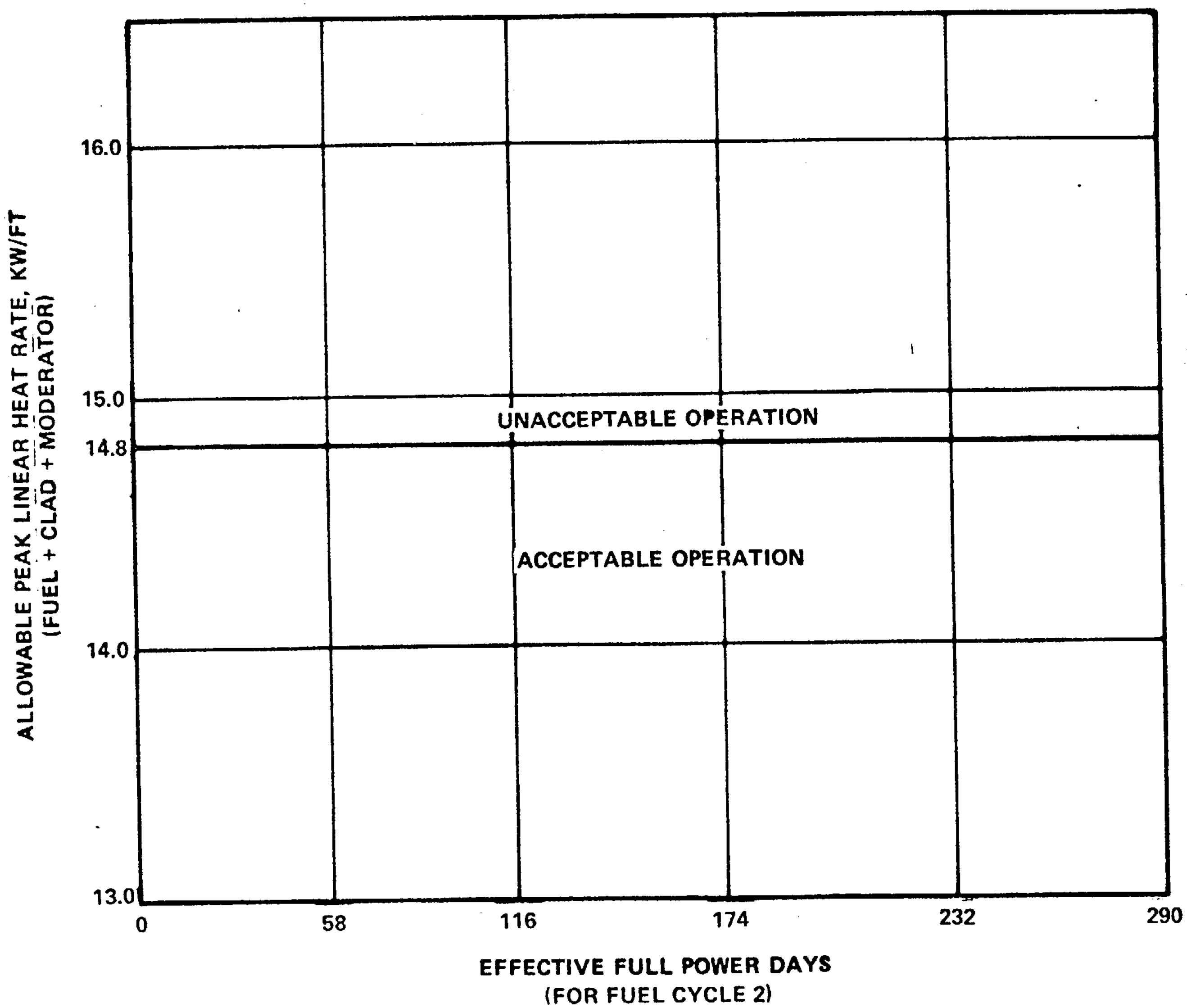

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

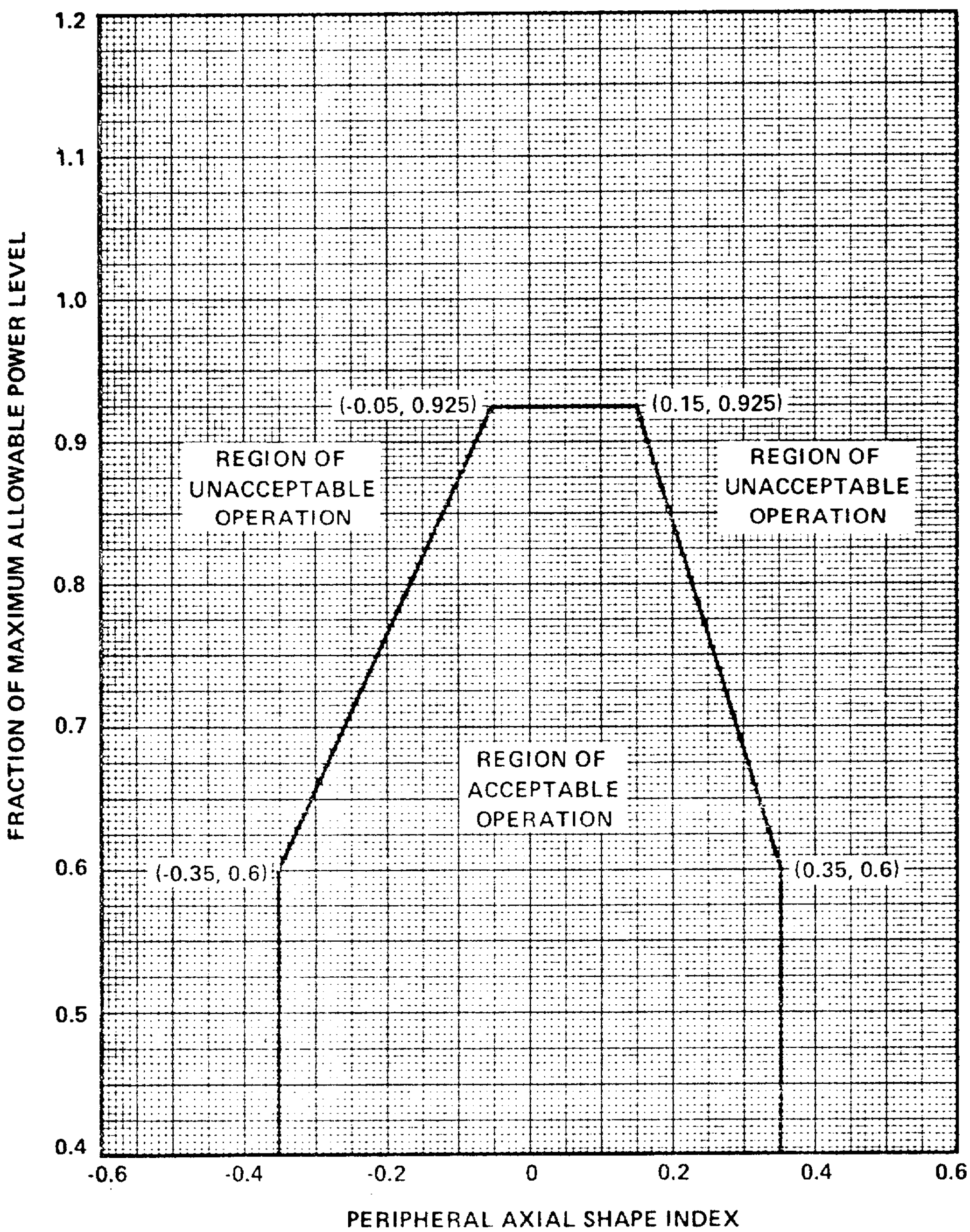

Figure 3.2-2
AXIAL SHAPE INDEX vs Fraction of Maximum Allowable
Power Level per Specification 4.2.1.3

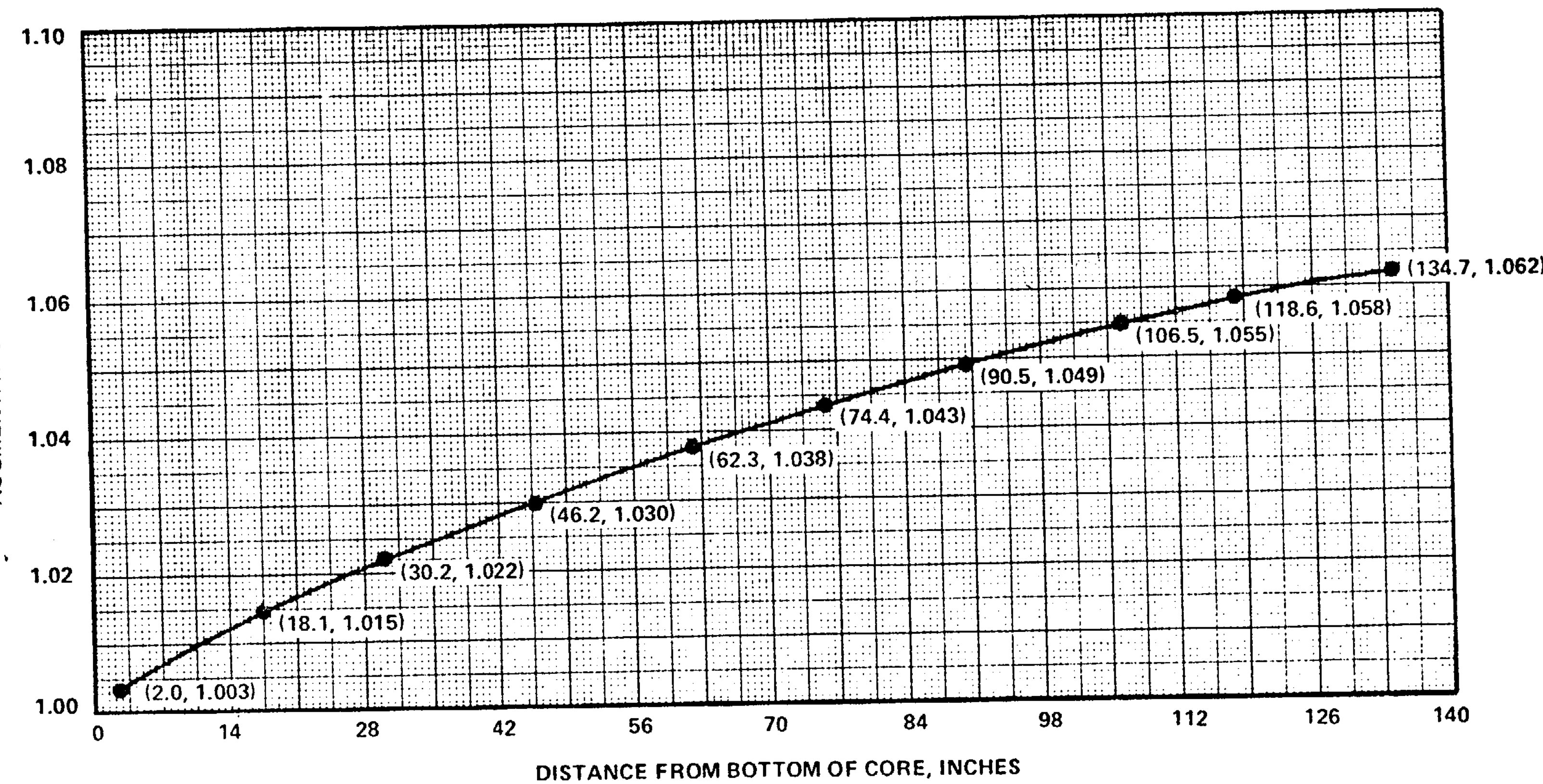

Figure 4.2-1 Augmentation Factors vs Distance from Bottom of Core

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.589.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.589, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ and F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.02.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height and shall exclude regions influenced by grid effects.

4.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

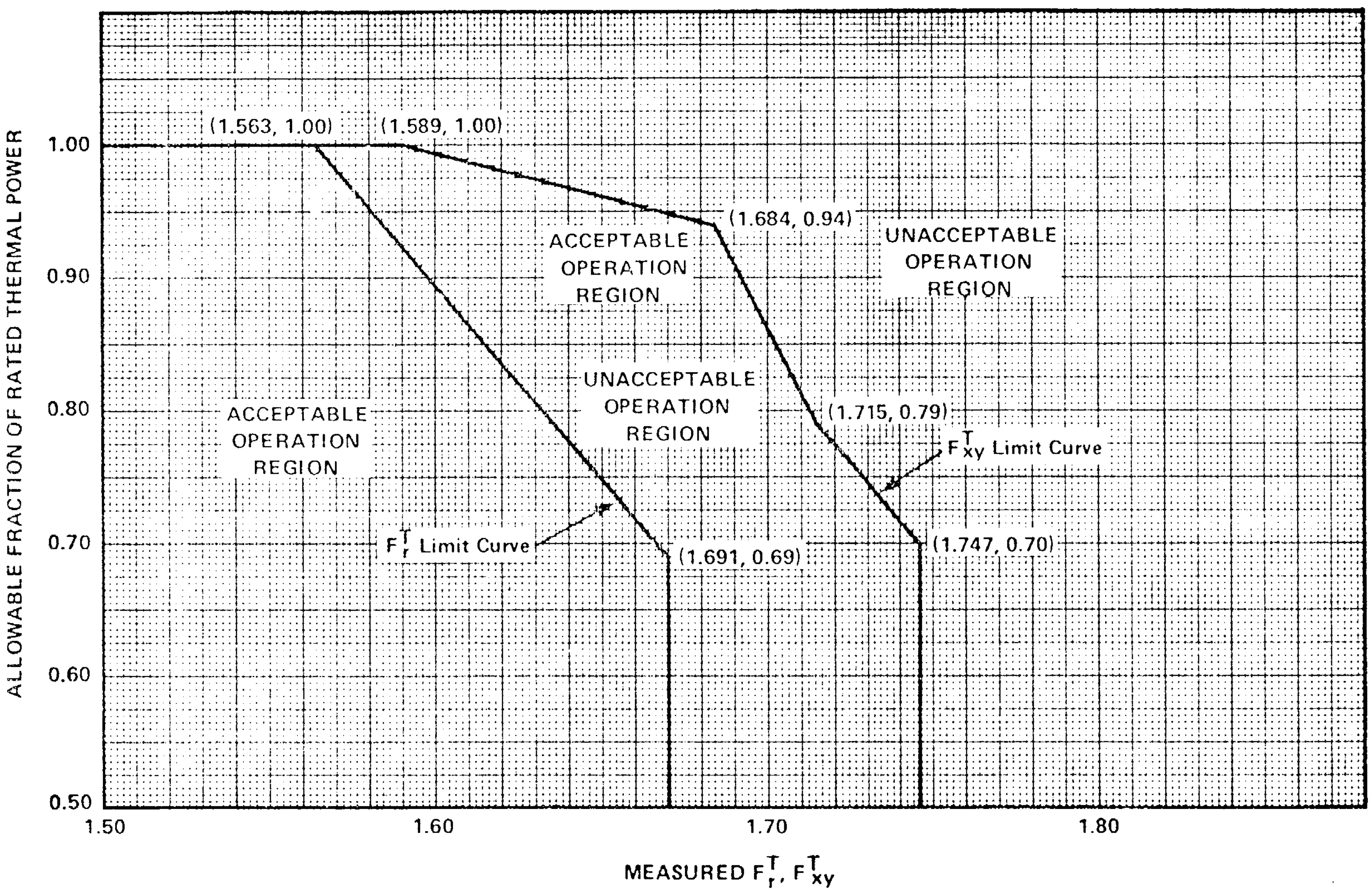

FIGURE 3.2-3
Allowable Combinations of THERMAL POWER and F_r^T and F_{xy}^T

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$, shall be limited to $\leq$ 1.563.

APPLICABILITY: MODE 1*.

ACTION:

With $F_r^T > 1.563$, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r\ (1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL TILT (T_q) is > 0.020.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.02.

APPLICABILITY: MODE 1*

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.020 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days when the Subchannel Deviation Alarm is OPERABLE,

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Calculating the tilt at least once per 12 hours when the Subchannel Deviation Alarm is inoperable, and

c. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to ≤ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits by instrument readout at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB MARGIN

LIMITS

Parameter	Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 542°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ 370,000 gpm
AXIAL SHAPE INDEX	Figure 3.2-4

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

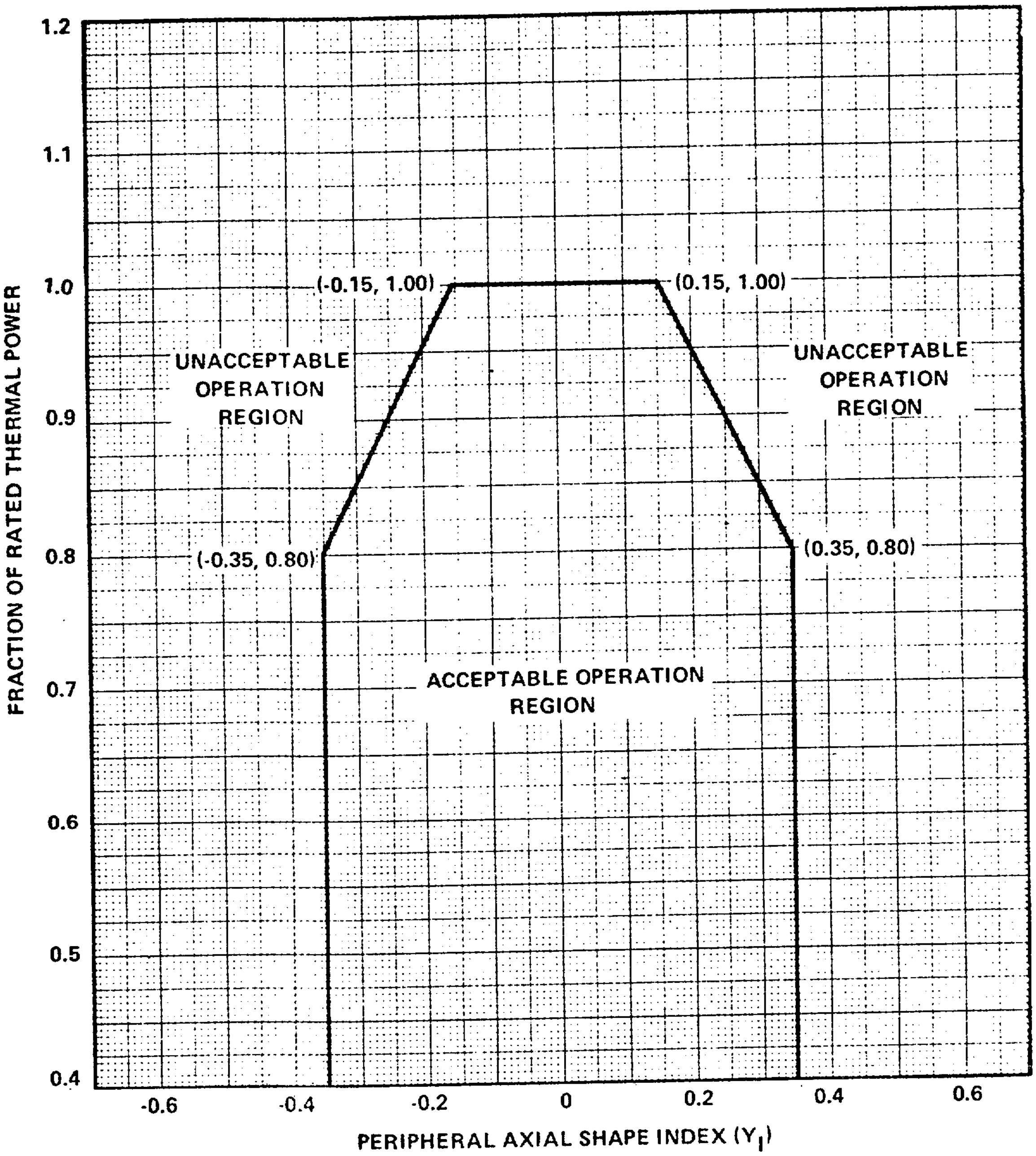

FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits with 4 Reactor Coolant Pumps Operating

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating-- Rate of Change of Power - High	4	2(d)	3	1, 2 and *	2#
b. Shutdown	4	0	2	3, 4, 5	3
12. Reactor Protection System Logic	4	2	4	1, 2*	4
13. Reactor Trip Breakers	4	2	4	1, 2*	4

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 585 psig; bypass shall be automatically removed at or above 585 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL power is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

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 HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on than channel provided the other inoperable channel is placed in the tripped condition.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	$\leq$ 0.40 seconds*# and $\leq$ 8.0 seconds##
3. Reactor Coolant Flow - Low	$\leq$ 0.65 seconds
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Containment Pressure - High	$\leq$ 1.40 seconds
6. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
7. Steam Generator Water Level - Low	$\leq$ 0.90 seconds
8. Local Power Density - High	$\leq$ 0.40 seconds*# and $\leq$ 8.0 seconds##
9. Thermal Margin/Low Pressure	$\leq$ 0.90 seconds*# and $\leq$ 8.0 seconds##
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	Not Applicable
11. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

#Response time does not include contribution of RTDs.

##RTD response time only. This value is equivalent to the time interval required for the RTDs output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Level - High				
a. Nuclear Power	S	D(2), M(3), Q(5)	M	1, 2
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow - Low	S	R	M	1, 2
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Containment Pressure - High	S	R	M	1, 2
6. Steam Generator Pressure - Low	S	R	M	1, 2
7. Steam Generator Water Level - Low	S	R	M	1, 2
8. Local Power Density - High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	N.A.	S/U(1)	N.A.
11. Wide Range Logarithmic Neutron Flux Monitor	S	N.A.	S/U(1)	1, 2, 3, 4, 5 and *
12. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

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

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr - ΔT Pwr." During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATER THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

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

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Four reactor coolant pumps shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.

ACTION:

MODES 1 and 2:

With less than four reactor coolant pumps in operation, be in at least HOT STANDBY within 6 hours.

MODES 3, 4 and 5:

Operation may proceed provided (1) at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump and (2) the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is increased to and maintained at $\geq$ 4.1% $\Delta k/k$ during operation in MODE 3 when less than four reactor coolant pumps are in operation. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1 The Flow Dependent Selector Switch shall be determined to be in the 4 pump position within 15 minutes prior to making the reactor critical and at least once per 12 hours thereafter.

All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1. Two independent shield building ventilation systems shall be OPERABLE.

APPLICABLILITY: MODES 1, 2, 3 and 4.

ACTION:

With one shield building ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

   2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested accordance with ANSI N510-1975 (130°C, 95% R.G.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.); or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510-1975 (130°C, 95% R.H.) and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

SURVEILLANCE REQUIREMENTS (Continued)

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%, and

b) Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 6.15 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

4. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.

5. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

6. Verifying that the main heaters dissipate 30 $\pm$ 3 kw and the auxiliary heaters dissipate 1.5 $\pm$ 0.25 kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the filtration system at a flow rate of 6000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filtration system at a flow rate of 6000 cfm $\pm$ 10%.

SHIELD BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.2 SHIELD BUILDING INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated at least once per 31 days by verifying that the door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or 3.2.4 are suspended.

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 This specification deleted.

SURVEILLANCE REQUIREMENTS

4.10.3 This specification deleted.

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 This specification deleted.

SURVEILLANCE REQUIREMENTS

4.10.4 This specification deleted.

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, RCS boron concentration, and RCS 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 RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values assumed for the MTC used in the accident and transient analyses were $+ 0.5 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $<$ 70% of RATED THERMAL POWER, $+ 0.2 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $>$ 70% of RATED THERMAL and $- 2.2 \times 10^{-4}$ $\Delta k/k/°F$ at RATED THERMAL POWER. Therefore, these limiting values are included in this specification. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

BASES

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 7,925 gallons of 8.0% boric acid solution from the boric acid tanks or 13,700 gallons of 1720 ppm borated water from the refueling water tank.

The requirements for a minimum contained volume of 371,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron addition capability after the plant has been placed in MODES 5 and 6 requires either 1660 gallons of 8% boric acid solution from the boric acid tanks or 1630 gallons of 1720 ppm borated water from the refueling water tank to makeup for contraction of the primary coolant that could occur if the temperature is lowered from 200°F to 140°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment (≥ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time of 3.0 seconds used in the safety analyses. Measurement with $T_{avg} \geq 515°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Long Term Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Power Dependent Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.058, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy}^T, F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_{xy}^T = F_{xy}(1 + T_q)$ and $F_r^T = F_r(1+T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

RADIATION MONITORING INSTRUMENTATION (Continued)

by the individual channels and 2) an alarm is initiated when the radiation level alarm setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capbility is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs", February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 This specification deleted

3/4.10.4 This specification deleted

3/4.10.5 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

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 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,100 ± 180 cubic feet at a nominal T_{avg} of 567°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new fuel storage racks are designed and shall be maintained with a center-to-center distance of not less than 21 inches between fuel assemblies placed in the storage racks. The spent fuel storage racks are designed and shall be maintained with a center-to-center distance of not

CRITICALITY (Continued)

less than 12.53 inches between fuel assemblies placed in the storage racks. These spacings ensure a K_{eff} equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The K_{eff} of $\leq$ 0.95 includes the conservative assumptions as described in Section 9.1 of the FSAR. In addition, fuel in the storage pool shall have a U-235 loading of $\leq$ 41.45 grams of U-235 per axial centimeter of fuel assembly.

DRAINAGE

5.6.2 The fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 728 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

5.9 COMPONENT CYCLE OR TRANSIENT LIMITS

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO 27 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

1.0 Introduction

By application dated March 3 and 22, 1978, April 4, 5, 12, and 28, 1978, and May 1, 1978; and supplemental information dated April 17 and 21, 1978 and May 11, 19, 22 and 23, 1978, Florida Power and Light Company (the licensee) requested amendment of Facility Operating License No. DPR-67. This amendment, in response to applications, consists of:

(1) Technical Specification (TS) changes resulting from the analyses of Cycle 2 reload fuel;

(2) TS changes to include consideration of a new water hole peaking factor;

(3) Approval to operate with sleeved Control Element Assembly (CEA) guide tubes;

(4) Deletion of certain license requirements that have been completed;

(5) TS changes authorizing the removal of all part length control element assemblies;

(6) Resistance Temperature Detector Testing Requirements; and

(7) Extension of time to install neutron shielding.

The major proposed changes to the Technical Specifications are listed in Table 1.

Table 1

Proposed Change

1. Revision to Shutdown Margin Requirements
2. Revision to Moderator Temperature Coefficient (MTC), and MTC Surveillance Requirements
3. Revisions to Boration System Requirements
4. Revisions to Thermal Margin/Low Pressure Trip Values
5. Revision to RTD Response Time Value
6. Deletion of Loss of Turbine and Rate of Power Change Trips
7. Change to Power Dependent Insertion Limit Requirements
8. Change in the Allowable Length of Cycles Operation Based on Cladding Collapse Calculations

2.0 Fuel Design

St. Lucie Unit 1 will replace 60 Batch A fuel assemblies with 60 Batch D fuel assembles for Cycle 2 operation. The licensee states that

> "The mechanical design of the reload fuel assemblies, Batch D, is identical to that introduced with the Calvert Cliffs-1 Batch D fuel with the exception of a change in pellet dish depth and nominal fill pressure." (Reference 1)

The licensee has performec calculations (Reference 1) to show that no collapse of the cladding will occur during the design lifetime of the fuel in the core if axial gaps were formed due to fuel densification. Based on these calculations, the licensee has proposed eliminating requirements of the Technical Specifications which limit the length of a cycle due to cladding collapse considerations. Because the licensee has shown that no collapse will occur during the design lifetime of the fuel, the deletion of this Technical Specification requirement is acceptable. We have reviewed the licensee's calculations and have determined that the basic assumptions are appropriate and the methods used for calculations are adequately conservative. Therefore, we find that there is adequate assurance that significant clad collapse will not occur during cycle 2 of St. Lucie Unit No. 1

3.0 Nuclear Design

The licensee reported making the following changes to the nuclear design methodology for Cycle 2 from the methodology used previously for St. Lucie Unit No. 1.

3.1 Extended Pointwise Doppler Feedback Technique

This technique considers the effect of temperature on reactivity feedback in calculating the power distribution on a local basis rather than on a batch basis which was the previous practice. This method was reported in Reference 5 and was approved by the staff in Reference 6.

This method produces more accurate determinations of power distribution under asymmetric CEA insertions than previous methods.

3.2 Uncertainties in Nuclear Power Peaking Factors and Water Hole Peaking

3.2.1 Uncertainty in Nuclear Power Peaking Factors in Cycle 1

FPL had assumed the following uncertainties in nuclear power peaking factors:

DNBR LSS	F_r uncertainty = 10%
DNBR LCO	F_r uncertainty = 8%
KW/FT LSSS	F_q uncertainty = 10%
KW/FT LCO	F_q uncertainty = 8%

CE proposed that an uncertainty in F_r of 5.1% and an uncertainty in F_q of 5.8% would be more appropriate (Reference 12)(In Reference 12 5.2% is stated instead of 5.1%. To one more figure the number is 5.13%) Based on the data on hand the staff judges these lower uncertainties to be nonconservative. However, a number of excess conservatisms still exist in the CE methodology. The staff judges these excess conservatisms to more than offset the potential nonconservatism in the 5.1% and 5.8%. Specifically the parts of the CE methodology which contain excess conservatism are:

(1) The W-3 DNBR correlation;

(2) The thermal-hydraulic methods in COSMO-INTHERMIC;

(3) Specific margin purposely built into the ASI trip tent;

(4) The ECCS analysis;

(5) Conservative assumption that all fuel has the same KW/FT limit as the oldest fuel in the reactor.

(6) Certain uncertainties that could logically be combined by root-sum-square are combined multiplivatively;

(7) Pseudo-Hot-Pin synthesis; and

(8) Unrealistically severe axial flux shapes predicted by QUIX.

The staff judges the package of the probably nonconservative 5.1% and 5.8% and the excess conservatisms just listed to produce a safety analysis that is conservative and therefore acceptable.

3.2.2 Improved Prediction of Power Peaking in Fuel Pins Adjacent to Water Holes

In recent benchmarking of their PDQ (computer code) design model against critical experiments, Combustion Engineering (CE) discovered that their design model underpredicted by 4.6% power peaking in pins near water holes which are the hot pins in the assemblies (Reference 3).

FPL recommended that the design model be modified by adding the 4.6% to the previously assumed uncertainty in the core "power peaking" factors.

FPL also suggested that credit be given for certain conservatisms in their analysis to partially or wholly offset this 4.6% penalty. These suggested credits were not approved by the NRC and consequently were not used in the development of the Technical Specifications.

Thus, the licensee is taking the full 4.6% penalty and is assuming a total uncertainty for F_r^T of 5.1% + 4.6% = 9.7% and a total uncertainty for F_q^T of 5.8% + 4.6% = 10.4%. The application of these uncertainties to the TS are described in Section 12.1. We therefore conclude that the licensee has appropriately incorporated water hole peaking in his analysis.

3.2.3 Statistical Combination of Uncertainties

In Cycle 1 FPL had conbimed certain uncertainties multiplicatively. FPL proposed that a root-sum-square combination of these uncertainties was more appropriate than the multiplicative combination (Reference 3) and the staff concurred with this proposal (Reference 4). In their reanalysis FPL used the root-sum-square combination of uncertainties.

3.3 Improved Correlation Between Fuel Temperatures and Local Power Density

The FATES Computer Code is used to calculate the fuel temperature used in turn to calculate Doppler broadening of nuclear cross sections.

The FATES Code had been previously approved by the NRC for calculating fuel temperatures for input to LOCA calculations (Reference 11). For cross section calculations, the previous CE method used an empirical method which did not account for fuel burnup effects on fuel temperature.

Since this new method, using FATES to calculate nuclear cross-sections, will result in more accurately calculated cross section, we find this method acceptable.

3.4 Nuclear Core Characteristics

Several changes to the core nuclear characteristics required by the Technical Specifications have been made from Cycle 1 to Cycle 2.

(1) The moderator temperature coefficient for Cycle 2 at rated thermal power is slightly less negative than the Cycle 1 value.

(2) The licensee has recalculated his power dependent insertion limits for Cycle 2. For Cycle 2 the maximum steady state insertion of the first regulating bank is limited to 25". This number was greater than 65" for Cycle 1.

The effects of these two changes are included in the Safety Analysis of the postulated Anticipated Operational Occurrences (AOO's) or accidents. (Section 5 below) The licensee has either reanalyzed these events for Cycle 2 or stated that the previous analyses are still bounding. Therefore, these changes should be made to make the safety analysis and Technical Specifications consistent.

4.0 Thermal Hydraulics

The licensee stated that the same design codes as described in the FSAR were used for the Cycle 2 analyses. A change was made to the methodology used in these calculations for Cycle 2. This was the removal of the local engineering heat flux uncertainty factor from the F_r^T value used to calculate the DNBR. This factor was combined with the nuclear uncertainty on the F_r^T and fuel rod bowing augmentation factor. The combined statistical factor was then used to reduce the PFDM curves (which protect the core from fuel failure due to exceeding the critical heat flux) in order to account for uncertainties. The staff concludes that the licensee has adequately considered these uncertainties.

5.0 Postulated Anticipated Operational Occurrences and Accidents

The licensee has reviewed the key input parameters to all Postulated Anticipated Operational Occurrences (AOO's) and accidents to assure that the Cycle 2 input parameters (reactivity coefficients, rod worths, etc.) are bounding. The licensee has stated this to be true with the following exceptions:

(1) Dropped CEA Worth - Full Length

(2) Post CEA Ejection Radial Peaks and Ejected CEA Worths

(3) Inverse Boron Worths and Critical Boron Concentration

(4) Radial Peaking Factors, F_{xy}^T, F_r^T

(5) CEA Differential Worth

(6) Moderator Cooldown Feedback during a Steam Line Break

(7) Three Pump Plenum Factor

Where one or more of these parameters was no longer bounding in the analysis of an AOO or accident, the event was reanalyzed.

Table 2 (from Reference 1) gives a list of the AOO's and accidents in the St. Lucie Unit 1 FSAR. The FSAR analysis is still valid for those events which were not reanalyzed. In all cases of reanalysis, the licensee has shown that the applicable criteria are satisfied. The DNBR limit and the fuel centerline melting limit are two such criteria. The reanalyses are therefore acceptable.

TABLE 2

ST. LUCIE UNIT 1, CYCLE 2
INCIDENTS CONSIDERED IN TRANSIENT AND ACCIDENT ANALYSIS

	Analysis Status
Anticipated Operational Occurrences for which the PPS* Assures no Violation of SAFDLs:**	
Control Element Assembly Withdrawal	Reanalyzed
Boron Dilution	Reanalyzed
Startup of an Inactive Reactor Coolant Pump	Not Reanalyzed
Excess Load	Not Reanalyzed
Loss of Load	Not Reanalyzed
Loss of Feedwater Flow	Not Reanalyzed
Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
Reactor Coolant System Depressurization	Not Reanalyzed
Loss of Coolant Flow[1]	Reanalyzed
Loss of AC Power	Not Reanalyzed
Anticipated Operational Occurrences which are Dependent on Initial Overpower Margin for Protection Against Violation of SAFDLs:	
Loss of Coolant Flow	Reanalyzed
Loss of AC Power	Not Reanalyzed
Full Length CEA Drop	Reanalyzed
Part Length CEA Drop	Not Reanalyzed
Part Length CEA Malpositioning	Not Reanalyzed
Transients Resulting from Malfunction of One Steam Generator	Not Reanalyzed
Postulated Accidents:	
CEA Ejection	Reanalyzed
Steam Line Rupture	Reanalyzed
Steam Generator Tube Rupture	Not Reanalyzed
Seized Rotor	Reanalyzed

[1]Requires Low Flow Trip.

*RPS: Reactor Protection System

**SAFDL: Specified Acceptable Fuel Design Limits

In analyzing the CEA withdrawal transient, the licensee took credit for a decreased coil holding time. The reduction was from a value of 0.5 second for Cycle 1 to 0.4 second for Cycle 2. This reduction is based on rod drop tests run during Cycle 1. Based on our review, we accept this reduction and the subsequent credit in DNBR. The licensee has agreed to additional tests during the startup for Cycle 2 to demonstrate that the lower value of coil holding time is still valid. The results of these measurements will be reported in the physics startup test report (see Section 11).

All other AOOs and accidents were reanalyzed according to the methods of the FSAR and are therefore acceptable with the following two exceptions. The Steam Line Break Analysis was partially reanalyzed. Since there were no changes to the system between Cycle 1 and Cycle 2 which would affect the cooldown curve, cooldown analysis was not redone. The Cycle 1 cooldown curve was reused, correcting for the fact that there was an energy addition to the coolant for Cycle 1 and not for Cycle 2. Changes in reactivity were recalculated based on the nuclear characteristics of the Cycle 2 core.

The loss of coolant flow was analyzed with a margin in DNB of 2% included to accommodate an increase in the required overpower margin due to the use of a analysis method previously used for the Calvert Cliffs Unit 1 Cycle 2 reload safety evaluation. This is the use of the CEDNBR code rather than the COSMO/INTHERMIC code. The staff review of CEDNBR has not yet been completed. However, at this stage in the review, no safety related problems have been identified, and our review has progressed to the point that we are confident that St. Lucie Unit 1 can be safety operated in Cycle 2.

6.0 Loss-of-Coolant Accident

The licensee has reanalyzed the most limiting break from Cycle 1 and has determined that the required ECCS criteria are met. The high density Batch D fuel was calculated to be most limiting for Cycle 2 in terms of peak cladding temperature. Table 3 gives the peak cladding temperatures and cladding oxidation thicknesses for the low density Batch B fuel and the high density Batch D fuel.

The assumed peak linear heat generation rate for Cycle 2 was assumed to be the same as that for Cycle 1, 14.8 kw/ft. The calculations were performed with a model in full compliance with 10 CFR 50 Appendix K. The results of the calculations are shown in Table 3. These results satisfy the criteria of paragraph 50.46 of 10 CFR 50 and, therefore, are acceptable.

TABLE 3

RESULTS OF ECCS ANALYSIS

Fuel Type	Density(%TD*)	Peak Cladding Temperature(°F**)	Cladding Oxidation Thickness(%)
Batch B	93	1972	11.8
Batch D	95	2035	12.0

*TD = theoretical density

**For a 0.8 DES/PD break at 14.8 $\frac{kw}{ft}$

7.0 CEA Guide Tube Integrity

Indications of significant wear in the CEA guide tubes of fuel assemblies were found during the fuel inspection program following Cycle 1 operation of Millstone Unit 2. The guide tube wear has been observed at the location of the control rod tips in the "full up" position. Subsequent inspections of discharged fuel at St. Lucie Unit 1 and other operating reactors supplied by Combustion Engineering have produced similar indications. The guide tubes serve in a dual capacity as the primary structural members of the fuel assembly and as guiding channels for the control rods during insertion.

Considering these findings, FPL instituted an eddy current testing (ECT) program to quantify the extent of wear experienced during Cycle 1. This program was developed to assess the thermal hydraulic performance and structural integrity of fuel assemblies with worn guide tubes for service in Cycle 2. The observations were incorporated in analyses to demonstrate the ability of the core to maintain its coolable geometry and the ability of the CEAs to scram, as required by the safety analyses. The licensee has concluded that fuel assemblies with worn guide tubes can be operated safely. However, the licensee has decided to modify 87 of 217 fuel assemblies in the core by an addition of a stainless steel sleeve to restore lost structural margins.

Combustion Engineering has developed a method of reinforcing worn guide tubes by using thin stainless steel sleeves. The sleeves are inserted within the guide tubes, bridging the worn cross-sections, thus providing a significant increase of strength and stiffness.

The sleeves are made of type 304 stainless steel, slightly cold-worked to provide a yield strength of over 60,000 psi. They are chromium plated on the ID and on the upper part of the OD to improve wear resistance. The sleeves extend from the top of the guide tube to several inches below the area where the wear has occurred. The sleeves are securely fastened in place by mechanically "bulging" both the sleeve and the guide tube at the lower end of the sleeve. This "bulge" extends for approximately one inch axially, and results in diametral expansions of the guide tubes of a few hundredths of an inch on new (unirradiated) guide tubes, and slightly less on used (both worn and unworn) irradiated tubes.

In addition to this guide tube expansion, the lower portion of the sleeves is expanded diametrally toward the guide tubes, so that the annular gap between the guide tube and the sleeve is approximately zero at room temperature. At operating temperature contact stresses develop from differential thermal expansion between the Zircaloy and the stainless steel. The gap in the upper portion of the assembly permits axial and radial differential thermal expansion of the sleeve without imposing significant loads on the assembly.

A series of slots and holes is provided in the sleeves to permit water flow in the annulus between the sleeve and the guide tube to minimize the possibility of "steaming" caused by poor heat transfer between the sleeve and the guide tube.

The sleeving modification serves as an interim solution to mitigate the effects of guide tube wear but does not eliminate the source of the wear. The wear is believed to be caused by a flow induced vibration of the Inconel CEA rubbing against the Zircaloy guide tube. Investigations are continuing through out-of-pile flow visualization tests in an effort to understand the mechanisms producing the vibrations.

All guide tubes in fuel assemblies under CEAs will be unworn and sleeved with the exception of one slightly worn A batch assembly that will not be sleeved. The unsleeved A batch assembly will be located at the center of the core, a location that has exhibited low wear in the previous cycle. The total wear at the end of cycle 2 is expected to result in relatively little loss of tube cross section. Therefore, operation in cycle 2 with this assembly in the center of the core is acceptable.

7.1 Structural-Mechanical

The stainless steel sleeve provides reinforcement by adding strength and stiffness in the worn region. It is free to expand axially under heatup or cooldown. Consequently, because of its manner of installation, the sleeve does not provide axial support. However, it does significantly limit lateral deflection of the guide tube arising from both external moments and moments generated by the asymmetrical wear and thus reduces guide tube stresses.

The licensee has provided a stress analysis for the normal operating and accident loading conditions for the limiting conditions of wear in both sleeved and unsleeved assemblies to be loaded in Cycle 2. The various mechanical loads to which the fuel assemblies are or may be subject include: fuel assembly holddown loads, fuel assembly handling loads, CEA scram deceleration loads, seismic loads and loss-of-coolant accident (LOCA) loads. The capability of the guide tubes to sustain these loads is determined by demonstrating that the lateral deflection of the guide tubes and the associated mechanical friction during scram are not sufficient to prevent CEA insertion and that a coolable geometry is maintained by limiting permanent deformation of the fuel assembly.

The licensee has provided an analysis of the mechanical integrity of the core for a postulated LOCA and has concluded that the fuel remains in a coolable array. However, a review of the response of the core to this loading condition has been deferred pending resolution of the generic Category A-2 task action plan, "Asymmetric Blowdown Loads on PWR Reactor Vessels." The targeted completion date of this program that includes a revised LOCA analysis and any required plant modifications is January 1980. The continued operation in the interim period of time is justified in view of the low probability of a large pipe break.

A seismic analysis was completed for the effects of a postulated safe shutdown earthquake using the St. Lucie 1 reactor vessel flange acceleration time history. The staff concludes that this input realistically defines seismic excitation of the core. The seismic analysis accounts for the interaction effects of adjacent fuel assemblies and the core shroud through the use of appropriate gap and impact elements. Therefore, we find the licensee's seismic analysis methods to be acceptable.

The licensee's analyses show that the stress during expected and postulated loading conditions in all guide tubes, whether sleeved or unsleeved, remains below the unirradiated yield strength of the Zircaloy-4 material. In addition, the stainless steel sleeve stress intensity was calculated for the corresponding portion of the load that it carries and the stress was shown to be less than the material yield strength as given in Table 1-2.2, Appendix 1, Section III of the ASME pressure vessel code.

Interaction between the sleeve and guide tube creates substantial secondary stresses in addition to the before-mentioned primary stresses. Differential thermal expansion, differential irradiation induced growth, and creep have been considered and the resulting stresses have been determined.

Scram tests of a sleeved fuel assembly were also conducted to measure the 90% CEA insertion time. The tests were performed at operating temperatures and maximum flow conditions. The measured insertion times fell within the limits specified in the plant Technical Specifications, and did not vary from times measured with unsleeved assemblies and, therefore, the CEA scram are acceptable.

We have concluded that the licensee's calculated stress intensities are low enough to assure an adequate margin of safety. Furthermore, we have concluded that the licensee has demonstrated scramability and coolability as required by the General Design Criteria.

The consequences of a fuel handling accident were found to be bounded by the Final Safety Analysis Report (FSAR) analysis of a 176 rod activity dose where the corresponding exclusion boundary dose was found to be only a small fraction of the 10 CFR 100 allowable limits.

7.2 Control of Sleeving Procedure

The sleeving method used was covered by a written procedure (licensee's) that includes qualification of the tooling before each operation, and replacement of those parts of the tooling subject to wear or deterioration before any deleterious effects on the process could occur. After sleeving, the following checks are made to ensure that the process was performed correctly.

(1) A pull test of 50 lbs. was performed on each sleeve.

(2) A visual inspection was performed to ensure that the sleeve is properly seated and that no debris is left in the area.

(3) Two separate gaging operations, using a single thimble gage, and a five-finger gage, were performed to ensure that there will be no interference with CEA operation.

(4) To make sure that the "bulging" process did not produce cracks in the irradiated guide tubes, six assemblies were examined by borescope, including one of the most highly irradiated assemblies. No evidence of cracking was found.

7.3 Testing of Sleeved Guide Tubes

Combustion Engineering has performed a number of tests on sleeved guide tubes to verify the mechanical strength of the assembly, effect of sleeves on scram time, wear performance, and possible enhanced corrosion in the annulus between the sleeve and tube.

CE determined that the force necessary to pull out a sleeve from the guide tube is on the order of 800 pounds, and after 15 thermal cycles between room temperature and 625 degrees Fahrenheit to simulate relaxation that would occur in service, the pull-out force was still greater than 400 pounds.

They also ran a loop test on a sleeved assembly with a CEA inserted at the nominal full-out position to simulate the condition causing the guide tube wear. The chromium plated sleeves showed no measurable wear after 464 hours, just a slight polishing or burnishing. The mating CEA finger tips also showed no wear, just a slight polish. Sleeved tubes were cut open and examined metallographically. No evidence of accelerated corrosion in the crevices (annuli) was found.

Scram tests were also run on sleeved assemblies to determine if the presence of the sleeves, or the reduction in clearance (reduced by about a factor of 2) between the CEA fingers and the inside of the tube would affect scram time. The results of these tests showed negligible effects on scram time.

7.4 Conclusion on Sleeving

We have evaluated the information submitted by the licensee and have concluded that the sleeved guide tubes will perform their function of reducing guide tube stresses to acceptably low values, and that the mechanical design of the sleeved assembly is satisfactory for at least one fuel cycle. Any long term effects of relaxation of the mechanical "bulge" joint, including the possibility of radiation-enhanced relaxation, will have to be evaluated on selected assemblies at the next refueling outage.

FP&L has agreed to provide a CEA guide tube evaluation program at least 90 days prior to St. Lucie Unit No. 1 shutdown for Cycle 3 reload outage.

Some details of our evaluation are provided below.

7.4.1 Wear Resistance

Chromium plating of stainless steel and other similar alloys is commonly used in reactors, and has performed well. Chromium plate is extremely hard and wear resistant, often orders of magnitude better than materials like Zircaloy and stainless steel. Further, the desirable frictional and anti-galling properties of chromium plate tend to reduce wear on mating softer materials. We conclude that chromium plated sleeves are not likely to be worn significantly during at least one fuel cycle.

7.4.2 Mechanical Properties

The mechanical joint between the sleeve and the guide tube is designed to be several inches below the area of excessive wear. The diametral expansion of the lower portion of the sleeve also is intended to be below the lowest wear area to prevent stressing of the worn region of the guide tube through thermal contact stresses between the sleeve and guide tube. There should be no prior cracks, notches, or severe hydriding where the stresses in the guide tube occur. The mechanical properties of the irradiated Zircaloy guide tube will be more than adequate to sustain the stresses involved.

7.4.3 Crevice Corrosion and Hydriding

The installation of a sleeve in a guide tube creates an annulus between the guide tube ID and the stainless steel sleeve OD which reduces to a crevice at the expanded region. In response to our questions, CE considered the possibility of enhanced corrosion and hydriding of the guide tubes in the crevice areas. They have stated that the crevice in the "bulge" area will be too small (and after short exposure will be further closed up by corrosion product) to provide an entrance for the necessary water to cause extensive corrosion. They also argued that in the sleeve expansion region, this crevice will be closed at operating temperatures by the differential thermal expansion between the Zircaloy and the stainless steel, and the water will be squeezed out of the crevice, also limiting possible corrosion.

The crevice above the expanded region will be water filled. Holes and slots in the sleeve will allow some water circulation, minimizing the corrosion problems from stagnant water of acceleration of corrosion rate by the presence of steam phase.

We, too, have evaluated the possibility of detrimental enhanced corrosion and hydriding in the sleeve-to-tube crevice. Factors considered by the NRC staff included:

(1) Similar crevices between stainless steel and Zircaloy are present in Westinghouse low parasitic fuel assemblies and operate successfully.

(2) The next fuel cycle for St. Lucie Unit No. 1 will be relatively short duration (about 7000 EFPH), thus limiting the time before post-irradiation inspections will be performed on sleeved assemblies.

(3) The 464-hour tests described above showed that there were no short-term problems under out-of-pile conditions.

(4) In sleeved assemblies, the portion of the guide tubes subjected to high loads, such as the "bulge" area, will not have wear induced cracks or sharp notches. Under these conditions, some enhanced hydriding could be tolerated.

(5) In reviewing reactor experiences with crevices, no enhanced corrosion or hydriding has been noticed except in those cases where concentration of nonvilatile impurities such as lithium hydrozide has occurred. Since the lithium hydroxide concentration could be increased in the sleeve/tube crevice by boiling (even if intermittent), there is some possibility of accelerated corrosion, enhanced hydrogen pickup, or both. The long-range aspects of the problem, including study of the possibility of hydrogen migration to the bulge region, are still under active review by the NRC staff.

We have concluded that there is a likelihood of some enhanced corrosion but it should not be severe enough to compromise the mechanical integrity of the sleeved design. Operation with sleeved guide tubes is acceptable for Cycle 2.

8.0 License Conditions

The operating license for St. Lucie (DPR-67) specified a number of requirements that must be completed prior to reactor startup following the first refueling shutdown. Some of these initial license conditions have been completed and deleted from license No. DPR-67. All of the remaining license conditions that were to be completed prior to reactor startup on Cycle 2 have been satisfactorily completed with the exception of license condition D for which FPL has asked for an extension of time to complete and license condition Q which requires the submittal of certain surveillance specifications for the NaOH containment spray additive system.

8.1 License Condition D

Condition D requires the installation of additional neutron shielding for the reactor cavity by the end of the first refueling shutdown. The shielding modification proposed by FPL has been reviewed and found acceptable by the NRC. By letter dated May 19, 1978, FPL requested that the NRC authorize a postponement in the installation of the shield because of dimensional interferences found during the initial fitup of the structural steel used for supporting the shielding. FPL further stated that substantial redesign of the structure would be necessary prior to shield installation and, therefore, requested that the NRC authorize an extension of time to complete the shield installation. FPL requested a delay until the end of the next refueling outage. FPL provided a discussion of health physics procedures that are to maintain radiation exposures to "as low as reasonably achievable" (ALARA).

We have reviewed the proposed extension of time for the installation of this shield the potential exposures, and the radiation exposure control procedures to be used by FPL and have determined that the shield should be installed during the next scheduled reactor shutdown (October 1978) if sufficient time (about one week) is available. We also determined that the shield should be installed no later than during the next refueling shutdown. Condition D of license has accordingly been modified. We have determined that exposures under the above license conditions will be ALARA and find the proposed amendment as modified acceptable.

8.2 License Condition I-2

License Condition I-2 requires the installation of auxiliary heaters in the shield building ventilation system for humidity control. FPL has now completed the installation of these heaters in accordance with their proposal which was reviewed and found acceptable by the NRC (SER Supplement #2, March 1, 1976). Surveillance requirements have been established in the Technical Specifications to assure continued operability of this humidity control system. With this system installed as required condition I-2 may be deleted from the license.

8.3 License Condition I-3

License condition I-3 required the installation of redundant and independent valve position indication for the mini-flow bypass valves (V-3659 and V-3660). The position indication system is now installed and this license condition may be deleted.

8.4 License Condition I-4

License condition I-4 requires installation of a permanent tornado-protected source of makup water to the reactor coolant system to accommodate moderator shrinkage during plant shutdown. FPL has provided this source with the installation of an interconnection between Safety Injection Tank and Volume Control tank. We have determined that the safety injection tanks, which are tornado protected, will provide adequate redundancy for makeup water in the event that the normal source is lost during reactor shutdown. With this system completed, license condition I-4 may be deleted.

8.5 License Condition I-5

License Condition I-5 required the installation of the necessary hardware to permit future interties to Unit No. 2. This equipment is now installed and I-5 may be deleted.

8.5 License Condition M

License Condition M required certain additional monitoring in mode one if excore monitoring is used with less than 10,000 MWD/MTU burnup. Since St. Lucie Unit No. 1 now has more than 10,000 MWD/MTU burnup, this requirement (condition M) is no longer applicable and may be deleted.

8.7 License Conditions N, O and P

License Conditions N, O and P required Resistance Temperature Detector (RTD) response time tests, underground cable tests and snubber functional tests prior to or during first refueling shutdown. These tests have been accomplished and the license conditions N, O and P deleted may therefore be deleted. The technical specifications continue to specify surveillance requirements for RTD response time, underground cables and snubbers. Additional verification requirements and surveillance requirements for RTD response time are discussed in Section 10.

9.0 Part Length Control Rods

Part length control rods have been removed from the reactor facility and FPL has proposed to delete reference to these rods in the technical specifications. The use of Part length rods has been prohibited by the Technical Specifications and they have been locked in the full out position during all reactor operations. Therefore, the removal of these part length rods will have no effect on the physics characteristics of the reactor. Similarily FPL has shown that there is no significant change in thermal or hydraulic effects by the use of plug assemblies to replace part length rods. Similar removal part length rods has been approved by the NRC for several other CE and Westinghouse nuclear power plants.

10.0 Summary of Findings on RTD's

By the submittal of Reference 7 FPL requested that the Resistance Temperature Detector (RTD) response time, τ, in their Technical Specification (TS) be increased from five seconds to eight seconds. We have determined that the TS change is acceptable.

The event which prompted this change was the reevaluation of the response time required every 18 months by TS 4.3.1.1.3. In this evaluation it was found that the response time of some of the RTD's was greater than five seconds. In the original tests in 1971 all RTD's were shown to have time constants of less than five seconds. However the two determinations of response time were measured using different techniques. At the present time the staff does not have sufficient data to determine with certainty whether this apparent increase in time constant represents a real degradation of the RTD's or simply represents an inconsistency between the two testing methods employed.

The change in RTD response time from five to eight seconds resulted in a recomputation of the Thermal Margin - Lower Pressure (TM-LP) trip setpoint, γ, and the Delta-T power calculator setpoints (a, τ) which have inputs from RTD. Both of these computations have been performed and the TS and Reactor Protection System (RPS) values adjusted accordingly.

The original evaluation of the RTD response time was performed by a plunge test, which to date has been the standard method for measuring response time. It would be necessary to remove the RTD's from their thermal wells in the reactor coolant piping if they are to be tested in this way during surveillance testing. To avoid doing this, the Loop Current Step Response (LCRS) technique for measuring RTD response time in-situ has been developed (References 8 and 9). This technique was used in the recent evaluation of the RTD response times.

In the surveillance testing only a quarter of the RTD's were tested, using the LCSR technique. In view of the apparent degradation observed in this small sample it is concluded that additional surveillance of the RTD response time should be performed during Cycle 2.

10.1 Setpoint Changes

The RTD response time affects the setpoint computation of three trip setpoints namely, the TM-LP setpoint, the Power Level-High setpoint, and the Local Power Density-High setpoint. These setpoints are established to assure that core thermal safety limits are not exceeded during the most severe postulated transient. A slower RTD response time means that the reactor coolant system temperatures that are input to these setpoints will lag the actual coolant temperature (during a transient) more than they did previously. Therefore, the setpoints must be revised to be accounted for this increased lag.

To determine the appropriate adjustments to the setpoints, the CE transient analyses code CESEC (Reference 10) was run for St. Lucie Cycle 2 for the anticipated limiting transients, assuming an RTD response time of eight seconds. From this array of transients the limiting value for the TM-LP coefficient ∂ and the Delta-T power calculator coefficients (a, γ) were determined. These values were intered into the Reactor Protection System.

The staff finds this approach to adjust the previously mentioned setpoints to be acceptable since the change in setpoints has maintained the same safety margin that existed previously.

10.2 The LCSR Technique

The parameter τ, called the RTD response time, is defined as the time required for the output from an RTD to reach 63.2% of its final response after being subjected to a step temperature change. In the analysis of reactor transients the temperature at an RTD is normally well approximated by a ramp function, and the appropriate time constant for analysis would be the Ramp Asymptotic Delay Time (RADT). It can be shown that in any physically realistic situation, the RADT is equal to or slightly less than τ, which makes τ a conservative estimator of the physically significant parameter, RADT.

As stated previously, the historic method for measuring is the plunge test. This process entails removing the RTD from its thermal well in the reactor coolant piping and shipping it to a laboratory where the tests are conducted. The tests are performed on the RTD assembled in two different thermal wells normally supplied by the testing laboratory. This has, in fact, been done for Millstone Unit 2 by Rosemont Engineering, who supplied the RTD's and performed the original plunge tests, and by Calvert Cliffs at the site.

The LCSR in-situ technique for measuring RTD response time relies not on temperature changes of the coolant, but rather on temperature changes of the RTD achieved by impressing an electric current through the RTD platinum element. From the time response to the current, the thermal characteristics of the RTD can be determined and the RTD response time inferred.

The method of impressing a current through the platinum element to develop an RTD temperature change was originally developed by Kerlin, et al (Reference 9). The method used at St. Lucie was an adaptation of Kerlin's method devised by Technology for Energy Corporation (TEC) (Reference 8). The principal innovation which TEC has added to the work of Kerlin is the incorporation of a semiempirical evaluation model based on the physical structure of the RTD and well. Using this model an unbiased estimate for τ can be computed using as input only the measured periods of the two slowest modes of exponential decay of the RTD. Prior to the introduction of this model, it was found that estimates of τ based on less physically explicit models underpredicted the RTD response time by as much as 30%.

10.3 Plunge Test Comparison to LCSR Test

To date TEC has made no direct comparison of LCSR results with the results of a plunge test, which is currently the accepted testing method. The staff will require such a comparison before approving the LCSR technique.

It should here be pointed out that neither the 170 DEGF plunge test nor the LCSR test measures directly the RTD response time at operating conditions. The following sources of error will appear with the two methods.

Sources of error in the plunge test:

1. Measurement errors.
2. Difference between the thermal well used in the test and the one in which the RTD will reside in the reactor coolant piping.
3. Correction for difference in fluid flow rates in test and in reactor.
4. Correction for difference in temperature at which test is conducted and temperature in reactor.
5. If a 540 DEGF plunge test is performed, then probably either oil or sand would be used instead of water, and the heat transfer coefficient would be different.

Sources of error in the LCSR test:

1. Measurement errors.
2. Inaccuracy of exponential fit with a finite number of terms.
3. Inaccuracy of the model used to infer the response time from the exponential fit.

10.4 Degradation of RTD Response Time

In Reference 8, TEC cites the cracking of the ceramic sheath around the platinum sensor and the accumulation of crud on the thermal wells to be probably the two principal mechanisms which may cause RTD response time degradation. However TEC sites no direct evidence that these mechanisms are indeed active.

It has been conjectured by FPL that the apparent degradation of the RTD response times might be merely a function of the measurement technique. However, the largest response time measured, 6.2 ± 0.5 seconds at a 90% confidence level, strongly suggests that the degradation is real. FPL has not reevaluated their LCSR results via a plunge test, but results from Millstone Unit 2 using a plunge test for surveillance show the same type of degradation as seen at FPL using the LCSR test. Based on these facts the staff judges the apparent degradation to be at least partially real, and not purely a result of the new measuring technique.

The specific response times measured in the LCSR test are as follows. The corresponding response times for the original Rosemont tests extrapolated to 540 DEGF are not available for comparison, but it is known that they were all less than five seconds.

RTD IDENTIFICATION	LCSR TIME CONSTANT (Seconds)
TE 1112 CA	4.0$\pm$0.2
TE 1112 HA	6.2$\pm$0.5
TE 1122 CA	5.5$\pm$0.2
TE 1122 HA	5.0$\pm$0.5

10.5 LCSR Verification

The RTD response time appears to possibly have suffered some degradation in 18 months of service. Therefore we assume that this degradation may be a continuing process. Until it is definitely established whether or not the RTD response time is experiencing a real degradation we required that a verification program for the LCSR technique and additional RTD surveillance testing program be conducted.

By letter dated May 19, 1978, FPL agreed to perform the above tests. The tests will be performed as follows:

1. The four RTD's tested via the LCSR technique during the recent surveillance will be removed and replaced

with spare RTD's. Four new RTD's that have not seen service will be procured. These new RTD's and the four RTD's removed from the reactor will be used for LCSR verification.

2. The response times for the removed RTD's will be measured via a 170 DEGF plunge test. For the four RTD's removed from the reactor these results will be compared with the LCSR results obtained in the recent in-situ surveillance tests and also compared with the original plunge test results obtained by Rosemont.

3. The response times for all eight RTD's will be measured via a 540 DEGF plunge test. For the four RTD's removed from the reactor these results will be compared with the LCSR results obtained in the recent in-situ surveillance tests.

4. In the same apparatus as used for the 540 DEGF plunge test the response times for all eight RTD's will be measured via the LCSR technique at 540 DEGF. The LCSR results will be compared with the plunge test results.

5. The response times for all eight RTD's will be measured via the LCSR technique at 170 DEGF. These results will be compared with the 170 DEGF plunge test results.

10.6 Additional RTD Response Time Surveillance

FPL has agreed to the following surveillance by letter dated May 1?, 1978:

1. Six months after startup all RTD's utilize in the RPS will be tested via the LCSR test. This will be baseline data for future tests.

2. Another surveillance of all RTD's will be conducted six months later. This data should be compared with that of the initial surveillance.

3. The data from these tests is scheduled to be forwarded to the NRC for evaluation by December 1, 1978. After this data is evaluated the staff will determine whether or not further surveillance is appropriate, and if so how often it should be performed.

Based on the above, we conclude that the combination of verification testing and increased surveillance of RTD response times will assure safe operation of St. Lucie Unit 1.

11.0 Physics Startup Tests

The physics startup test program as proposed by the licensee has been reviewed. The low power tests include critical boron concentration tests, temperature coefficient tests and CEA group worth tests. Power coefficient and temperature coefficient tests will be performed. After discussion, the licensee agreed to perform the CEA Symmetry Check tests on shutdown group A with an acceptance criteria of 2.5¢ deviation from the average value.

There are areas in the licensee's safety analysis particularily shutdown margin, power distribution and rod worths that warrant verification by the physics startup test program. The licensee has agreed to submit a draft physics startup test report to the NRC within 45 days of completion of the program and a final report within 60 days of completion of the program.

The entire program has been reviewed by the staff and found to be acceptable.

12.0 Technical Specifications

The licensee proposed changes to the St. Lucie Unit 1 Technical Specifications for Cycle 2 operation. The significant changes which have not been discussed previously (see Table 1) are discussed below.

12.1 Water Hole Peaking

Several changes to the Technical Specifications were required to account for water hole problem described in Section 3.2.

The total uncertainty assumed for F_r^T is 5.1% + 4.6% = 9.7% (see Section 3 for a discussion, of the component uncertainties). This is used in two places.

(1) DNBR LSSS (TM/LD Trip). The 9.7% uncertainty is assumed in the computation of the TM/LP trip setpoint Technical Specification figures 2.2-3 and 2.2-4.

(2) DNBR LCO (Monthly Incore F_r^T Surveillance). The 9.7% uncertainty is assumed in the computation of the F_r^T limits in Technical Specification figures 3.2-2 and 3.2-3.

The total uncertainty assumed for F_q^T is 5.8% + 4.6% = 10.4% (see Section 3 for a discussion of the component uncertainties). This is used in three places.

(1) KW/FT LSSS (Local Power Density Trip - ASI Tent). The 10.4% uncertainty is assumed in the generation of the Local Overpower ASI Trip Tent of Technical Specification Figure 2.2-2.

(2) KW/FT LCO - Continuous Incore Monitor. The 5.8% uncertainty is incorporated in the uncertainty parameter 1.058 of TS 4.2.1.4.b.2, and the remaining 4.6% is applied in the CECQR computer code which reduces the incore detector data.

(3) KW/FT LCO - Continuous Excore Monitor. The 10.4% uncertainty is assumed in the generation of TS figures 3.2-3 and 3.2-4.

12.2 Shutdown Margin (Sections 3.1.1.1 and 3.1.1.2)

The licensee proposed a change in the shutdown margin Technical Specifications to make the shutdown margin a function of the mode of operation. The modes of operation are given in Table 1.1 of the Technical Specifications and repeated here as Table 4.

The shutdown margins for modes 1, 2 and 3 were determined by the requirements of the worst possible cooldown event for that mode. This is the Steam Line Break Accident postulated to be initiated from that mode.

For modes 1 and 2 a shutdown margin of 3.3% was used based on the Full Power, Two Loop Steam Line Break and the Zero Power Two Loop Steam Line Break, respectively. For mode 3 the value was 4.1% based on the Hot Zero Power, One Loop Steam Line Break.

The licensee proposed a shutdown margin of 1% in Mode 4. We disagreed with the proposed number because it represents a reduction in the margin usually maintained in CE reactors. The procedure used in all other CE reactors is to choose the most conservative value of shutdown margin required for a given core and to use this value for all modes of operation for Modes 1 through 4. While we agree that it is acceptable to use the applicable Steam Line Break analysis to determine the shutdown margin for Modes 1 through 3, the value of 1% in Mode 4 represents a significant departure from past practice. Insufficient time was available to review and approve such a major change. Therefore, we did not allow the requested change in shutdown margin.

For Cycle 2 we have proposed and the licensee has accepted a value of 3.3% shutdown margin for Modes 1 through 4 when all four pumps are in operation and 4.1% when less than four pumps are in operation. This change to the shutdown margin Technical Specification is conservative and consistent with present practice with CE reactors and is therefore acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

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

TABLE 4

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 300°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 300°F$
3. HOT STANDBY	< 0.99	0	$\geq 300°F$
4. HOT SHUTDOWN	< 0.99	0	$300°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.

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

REFERENCES

1. Letter from Robert E. Uhrig, Florida Power and Light, to Victor Stello, Jr., USNRC, March 22, 1978.

2. Letter from D. Ziemann, USNRC, to Robert E. Uhrig, Florida Power and Light, December 3, 1976.

3. CEN-89(F)-P, Solution to Increased Water Hole Peaking in Operating Reactors (St. Lucie-1), Combustion Engineering, Inc., March 10, 1978.

4. Safety Evaluation Report: Interim Technical Specification Changes to Account for Power Peaking Near Water Holes in CE Reactors, cover letter from Robert L. Baer, Chief, Reactor Safety Branch, to Karl R. Goller, Assistant Director for Operating Reactors, DOR, March 3, 1978.

5. Calvert Cliffs Nuclear Power Plant Unit No. 1, Docket No. 50-317: Amendment to Operating License DPR-53 Second Cycle License Application, October 1, 1976.

6. Letter from D. Ziemann, USNRC to A. E. Lundvall, Jr., Baltimore Gas and Electric Company, March 14, 1977.

7. Florida Power and Light Submittal concerning Technical Specification changes in Resistance Temperature Detector response time, Cover letter from Robert E. Uhrig, Vice President, FPL to Victor Stello, Jr., Director, Division of Operating Reactors, February 2, 1978.

8. RESPONSE TIME OF PLATINUM RESISTANCE THERMOMETERS USING LOOP CURRENT STEP RESPONSE TECHNIQUES, Moll, Robinson, Jones, Mathis, Fisher, Technology for Energy Corporation, April 19, 1978.

9. EPRI NP-495, Project 503-3, IN-SITU TIME TESTING OF PLATINUM RESISTANCE THERMOMETERS, Kerlin, Miller, Moll, Upadhyaya, Hashemian, Arendt, University of Tennessee Nuclear Engineering Department, January 1977.

10. CENPD-199-P, C-E SETPOINT METHODOLOGY, Combustion Engineering, April 1976.

11. CENPD-139-P-A, FUEL EVALUATION MODEL, Combustion Engineering, July 1974.

12. CENPD-145, INCA - Method Analyzing In-Core Detector Data in Power Reactors, April 1975.

13. Letter to FPL from R. Reid, dated May 8, 1978.

UN: D STATES NUCLEAR REGULATORY CC ISSION

DOCKET NO. 50-335

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Facility Operating License No. DPR-67 issued to Florida Power and Light Company (the licensee), which revised the license and its appended Technical Specifications for operation of St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of its date of issuance.

The amendment authorizes:

(1) Technical Specification changes resulting from the analyses of Cycle 2 reload fuel;

(2) Technical Specification changes to include consideration of a new water hole peaking factor;

(3) Operation with sleeved Control Element Assembly (CEA) guide tubes;

(4) Deletion of certain license requirements that have been completed;

(5) Technical Specification changes authorizing the removal of all part length control element assemblies;

(6) Resistance Temperature Detector testing requirements; and

(7) Extension of time to install neutron shielding.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with Items (1) and (3) above was published in the FEDERAL REGISTER on April 18, 1978 (43 FR 16435). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. Prior public notice of the other items was not required since the items do not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated March 3 and 22, April 4, 5, 12 and 28, and May 1, 1978, as supplemented April 17 and 21, and May 11, 19, 22 and 23, 1978, (2) Amendment No. 27 to License No. DPR-67, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H

Street, N.W., Washington, D. C. and at the Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 26th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors