NOV 23 1981

RECEIVED DEC 3 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket No. 50-335

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

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your applications dated November 14, 1980 and September 28, 1981.

The amendment changes License Condition 2.C.(1) and the Technical Specifications to authorize operation of St. Lucie Unit 1 at 2700 Megawatts thermal power. The previously authorized maximum power level was 2560 Megawatts thermal.

Changes in the methodology for the core thermal hydraulic analyses, used to support the power increase have been reviewed. These changes include the use of the TORC core thermal margin code, the CE-1 critical heat flux correlation and the statistical combination of uncertainties (SCU) in the calculation of new limits for St. Lucie Unit 1.

In addition we have evaluated your reanalyses of accidents and anticipated operational occurrences. In the process of this evaluation we have found that certain items need your further attention. These items are listed below. The current status as well as your proposed actions, which are acceptable, are discussed in the referenced sections of the enclosed Safety Evaluation.

CP
1

- An alarm to alert the control room operators of a boron dilution event (III.D.7.1.1).
- An analysis of the loss of non-vital ac power taking into consideration the single failure criterion (III.D.7.2.3).
- An analysis of the seized reactor coolant pump rotor event taking into consideration, loss of offsite power and the single failure criterion (III.D.7.3.4).

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Copies of the Safety Evaluation and Environmental Impact Appraisal, and Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Original signed by:

Christian C. Nelson, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 48 to DPR-67
2. Safety Evaluation and Environmental Impact Appraisal
3. Notice of Issuance and Negative Declaration

cc: See next page

DISTRIBUTION:
Docket File
NRC PDR
LPDR
NSIC
TERA
ORB3 Rdg
DEisenhut
PMKreutzer-3
CNelson
RAClark
OELD
I&E-4
GDeegan-4
DMB-10
DBrinkman
ACRS-10
RDiggs
RBallard
Chairman, ASLAB
Gray File (+4)
L. KOPP
S. GUPTA
G. ALBERTHAL
M. THADANI

on the basis that the license conditions are not license conditions — they are not OCN

W/ no legal objection to amend & form and FR notice provided notice include appropriate N.D. statements

In correct form as is DOR Memo #5, Encl 13

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD	D:DL	
SURNAME	PMKreutzer	CNelson/pn	RAClark	TMNovak	W.CHANDLER	DGEISENHUT	
DATE	11/17/81	11/17/81	11/18/81	11/18/81	11/20/81	11/20/81	

Florida Power & Light Company

cc:
Harold F. Reis, Esquire
Lowenstein, Newman, Reis & Alexrad
1025 Connecticut Avenue, N.W.
Washington, D. C. 20036

Norman A. Coll, Esquire
McCarthy, Steel, Hector & Davis
14th Floor, First National Bank Building
Miami Florida 33131

Indian River Junior College Library
3209 Virginia Avenue
Fort Pierce, Florida 33450

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State oof Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Mr. Weldon B. Lewis
County Administrator
St. Lucie County
2300 Virginia Avenue, Room 104
Fort Pierce, Florida 33450

U.S. Environmental Protection Agency
Region IV Office
ATTN: Regional Radiation
Representative
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. Charles B. Brinkman
Manager - Washington Nuclear Operations
C-E Power Systems
Combustion Engineering, Inc.
4853 Cordell Avenue, Suite A-1
Bethesda, Maryland 20014

Mr. Jack Schreve
Office of the Public Counsel
Room 4, Holland Building
Tallahassee, Florida 32304

Resident Inspector/St. Lucie
Nuclear Power Station
c/o U.S.N.R.C.
P. O. Box 400
Jensen Beach, Florida 33457

cc w/enclosure(s) and incoming
dated: 11/14/80, 9/28/81

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. 48
License No. DPR-67

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

   A. The applications for amendment by Florida Power & Light Company (the licensee) dated November 14, 1980 and September 28, 1981 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, Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the Attachment to this license amendment, and by amending paragraphs 2.C(1) and 2.C(2) to read as follows:

   (1) Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal), provided that the construction items, preoperational tests, startup tests, and other items identified in Enclosure 1 to this license have been completed as specified in Enclosure 1. Enclosure 1 is an integral part of, and is hereby incorporated in this license.

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 48

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.

Pages

1-1
2-2
2-7
2-8
2-9
B 2-1
B 2-3
B 2-4
B 2-5
B 2-7
B 2-8
3/4 1-3
3/4 1-10
3/4 1-18
3/4 1-30
3/4 2-3
3/4 2-4
3/4 2-5
3/4 2-6
3/4 2-8
3/4 2-9
3/4 2-14
3/4 2-15
B 3/4 1-1
B 3/4 1-2
B 3/4 2-2
B 3/4 4-1
B 3/4 7-1
B 3/4 7-2

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

OPERATIONAL MODE

1.4 an OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

CONTAINMENT VESSEL INTEGRITY

1.8 CONTAINMENT VESSEL INTEGRITY shall exist when:

1.8.1 All containment vessel penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position except as provided in Table 3.6-2 of Specification 3.6.3.1,

1.8.2 All containment vessel equipment hatches are closed and sealed,

1.8.3 Each containment vessel airlock is OPERABLE pursuant to Specification 3.6.1.3, and

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

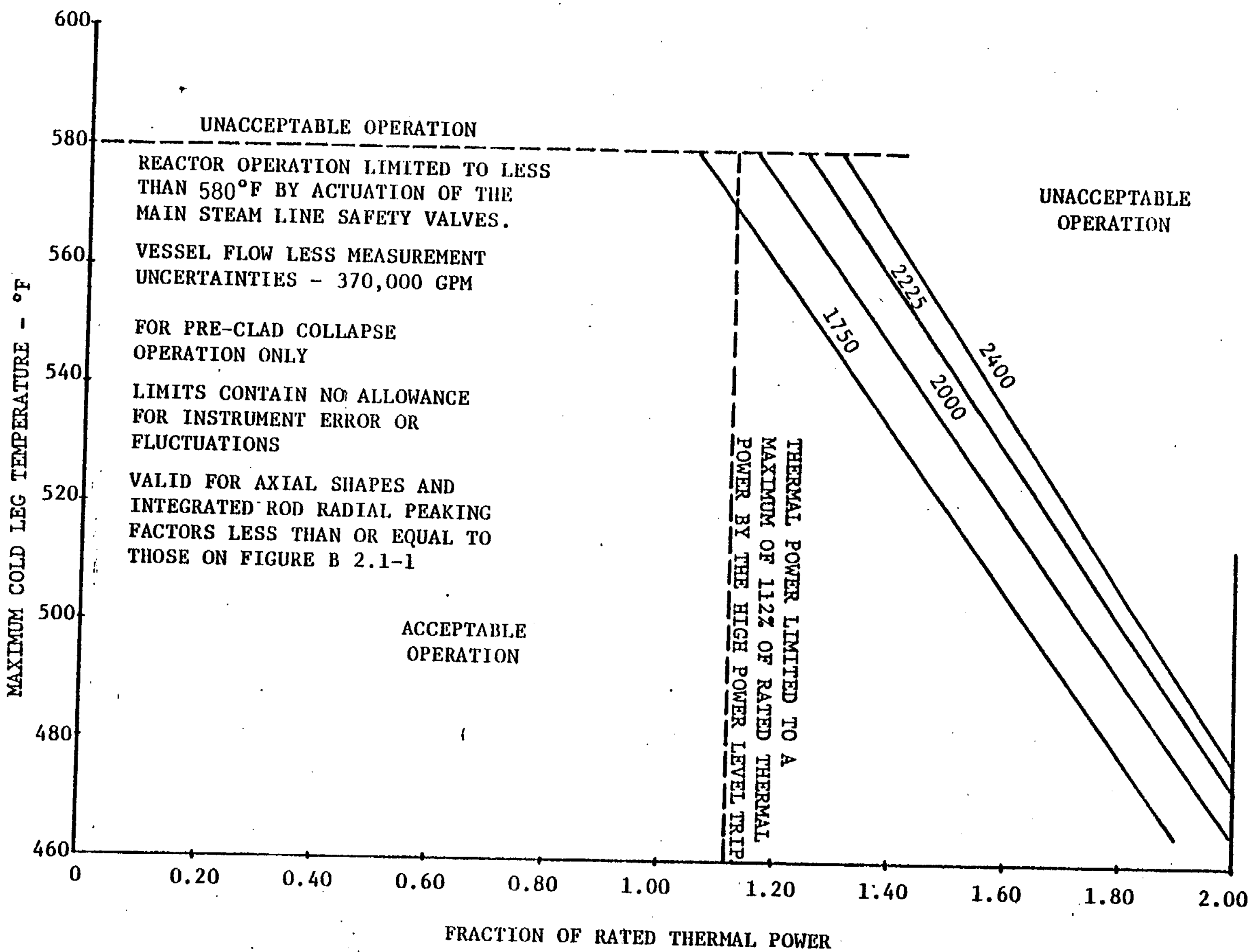

Figure 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT - FOUR REACTOR COOLING PUMPS OPERATING

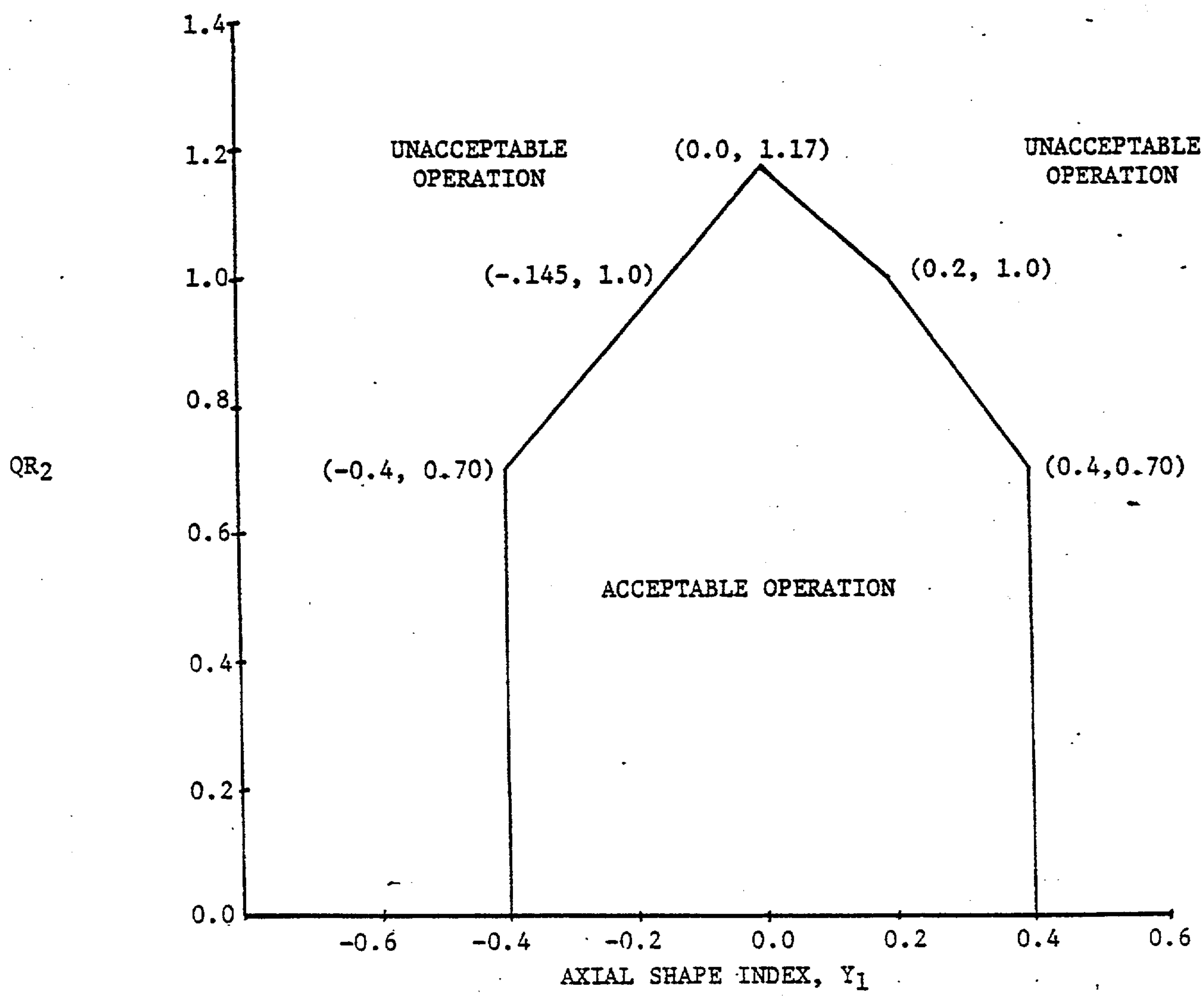

FIGURE 2.2-2

Local Power Density-High Trip Setpoint Part 2(QR_2 Versus Y_1)

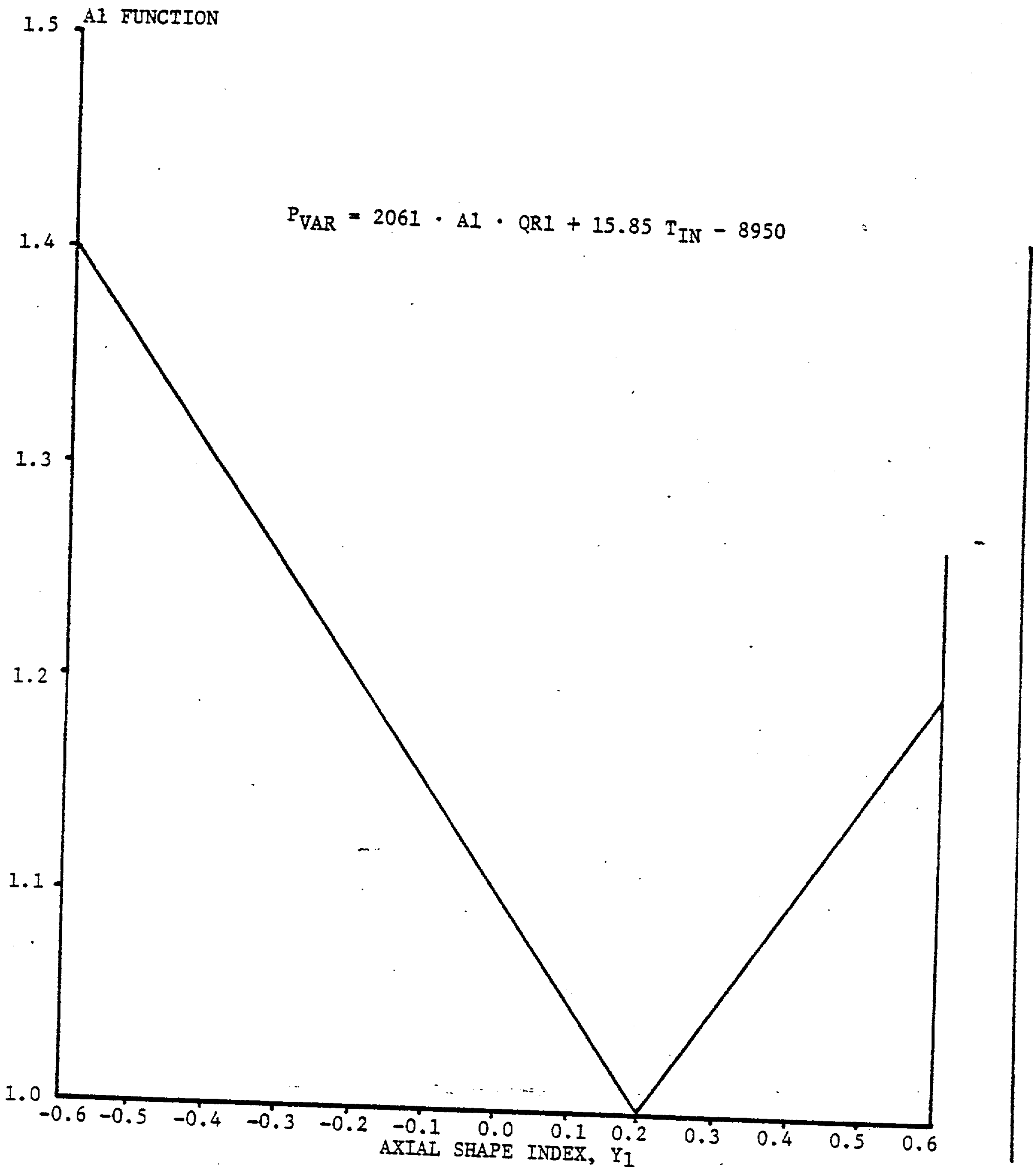

FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint

$$P_{VAR} = 2061 \cdot A1 \cdot QR1 + 15.85\ T_{IN} - 8950$$

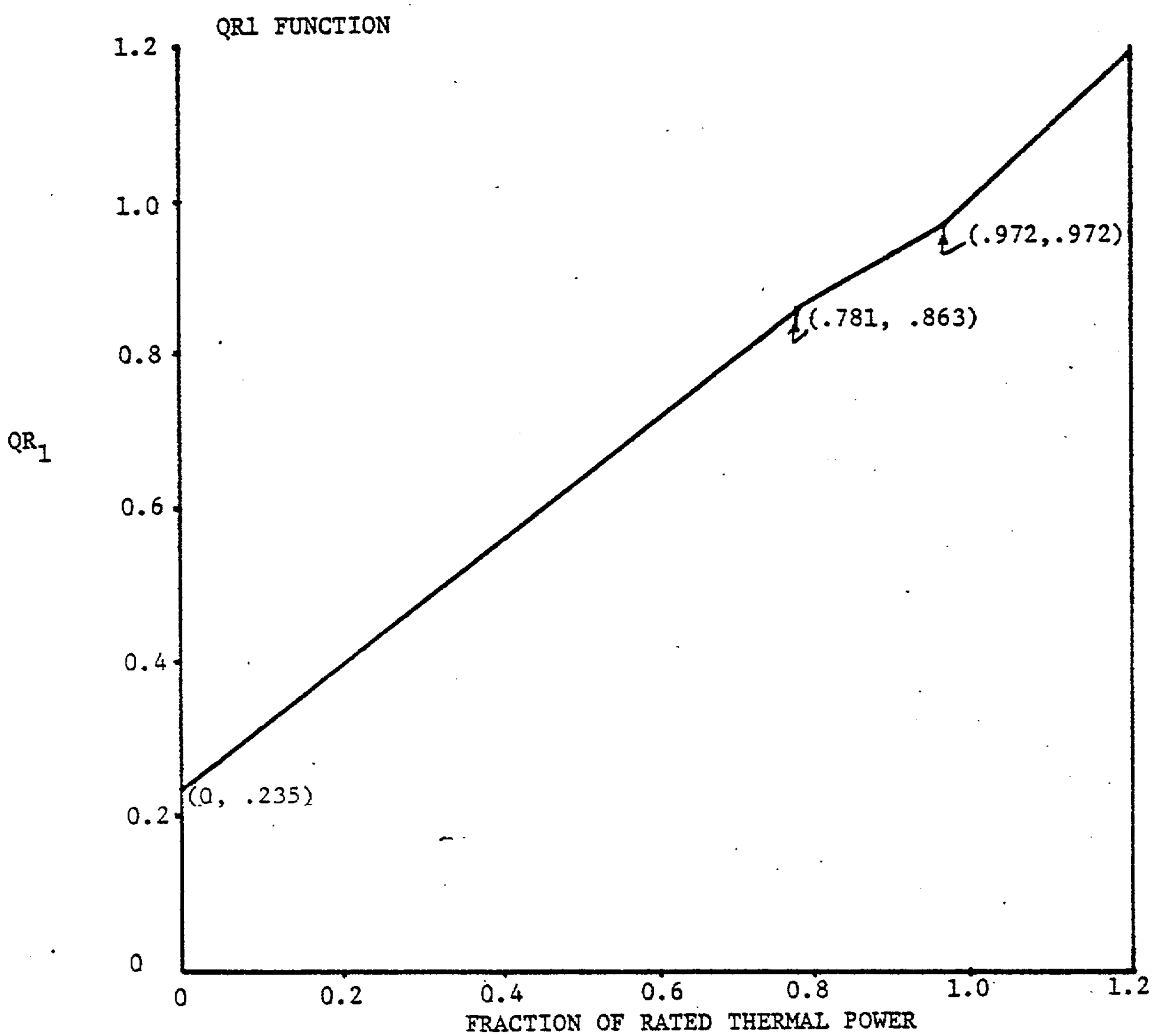

FIGURE 2.2-4

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

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 CE-1 correlation. The CE-1 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 1.23. 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 1.23 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.

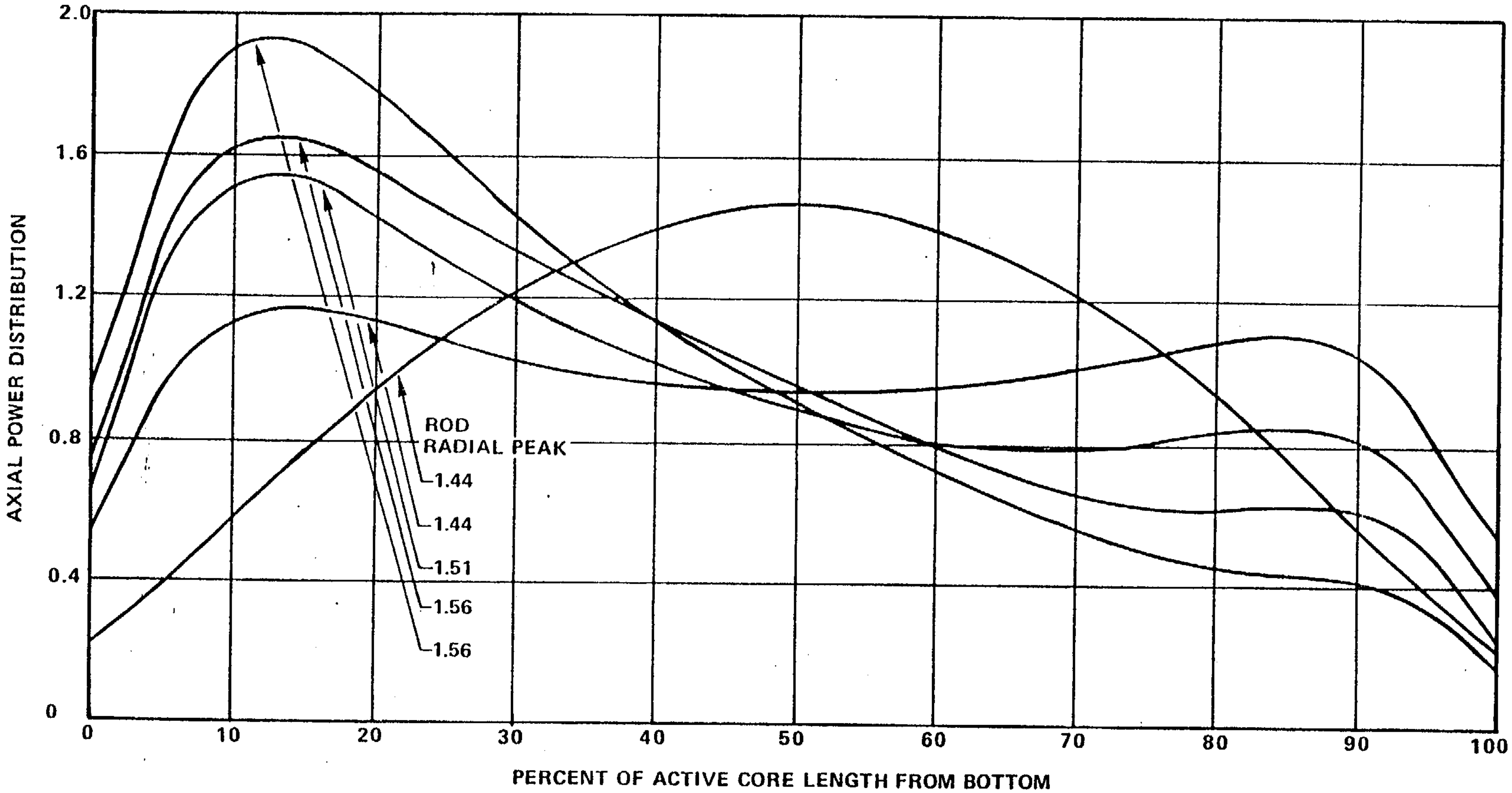

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

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 1.23 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.

2.2 LIMITING SAFETY SYSTEM SETTINGS

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 THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is consistent with the value used in the safety analysis.

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 1.23 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 1.23 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 600 psia 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 1.23.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1887 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 include appropriate allowances for equipment response time, calculational and measurement uncertainties, and processing error. A further allowance of 30 psia is included to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the DNBR limit.

Asymmetric Steam Generator Transient Protective Trip Function (ASGTPTF)

The ASGTPTF consists of Steam Generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those events associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single main steam isolation valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

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 ensuing 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.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

$\geq$ 2.0% $\Delta k/k$, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

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. 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.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

---

* Breaker racked-out.

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentation is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be $\geq$ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying $\geq$ 3000 gpm to the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Two flow paths from the boric acid makeup tanks via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System 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 2% $\Delta k/k$ at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

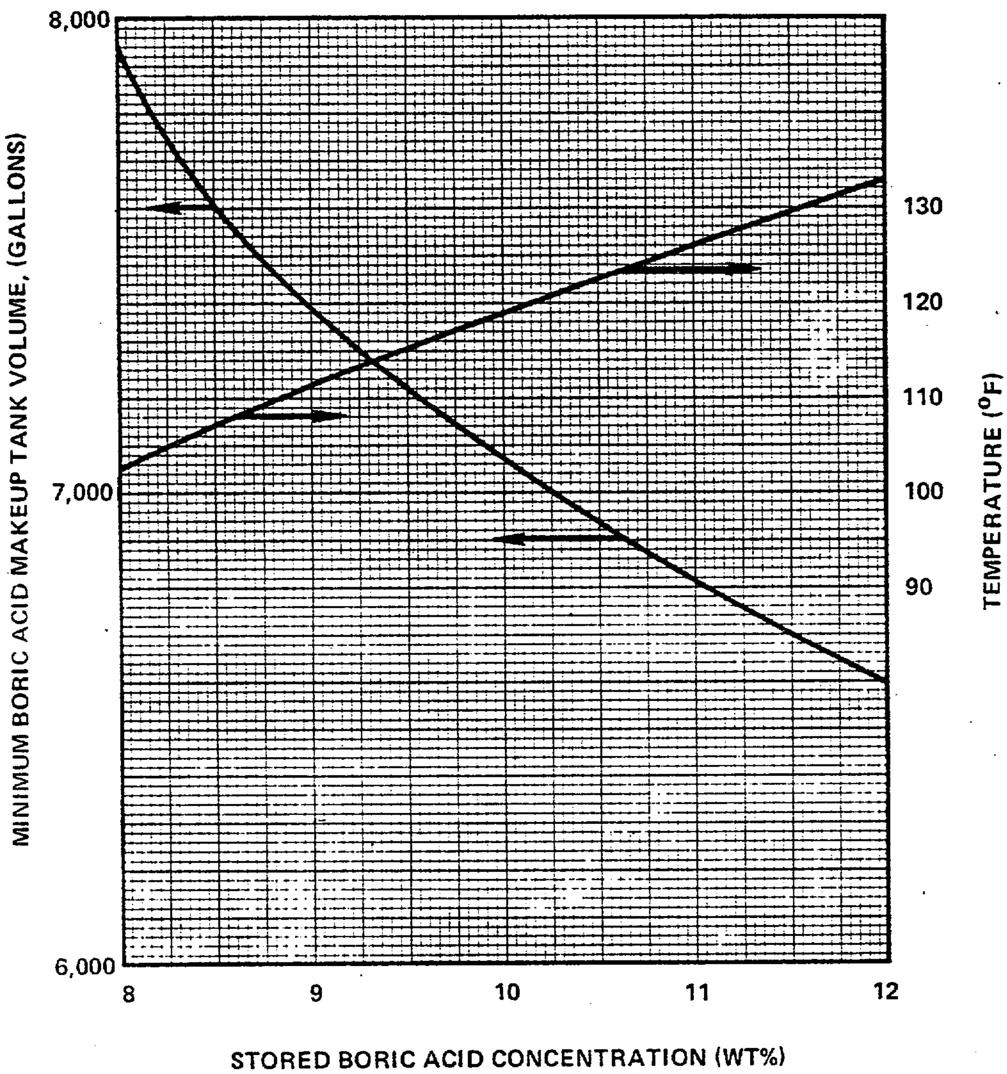

Figure 3.1-1 Minimum Boric Acid Makeup Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

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 401,800 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 2% $\Delta k/k$ 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 once per 7 days by:

1. Verifying the boron concentration in each water source,

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.

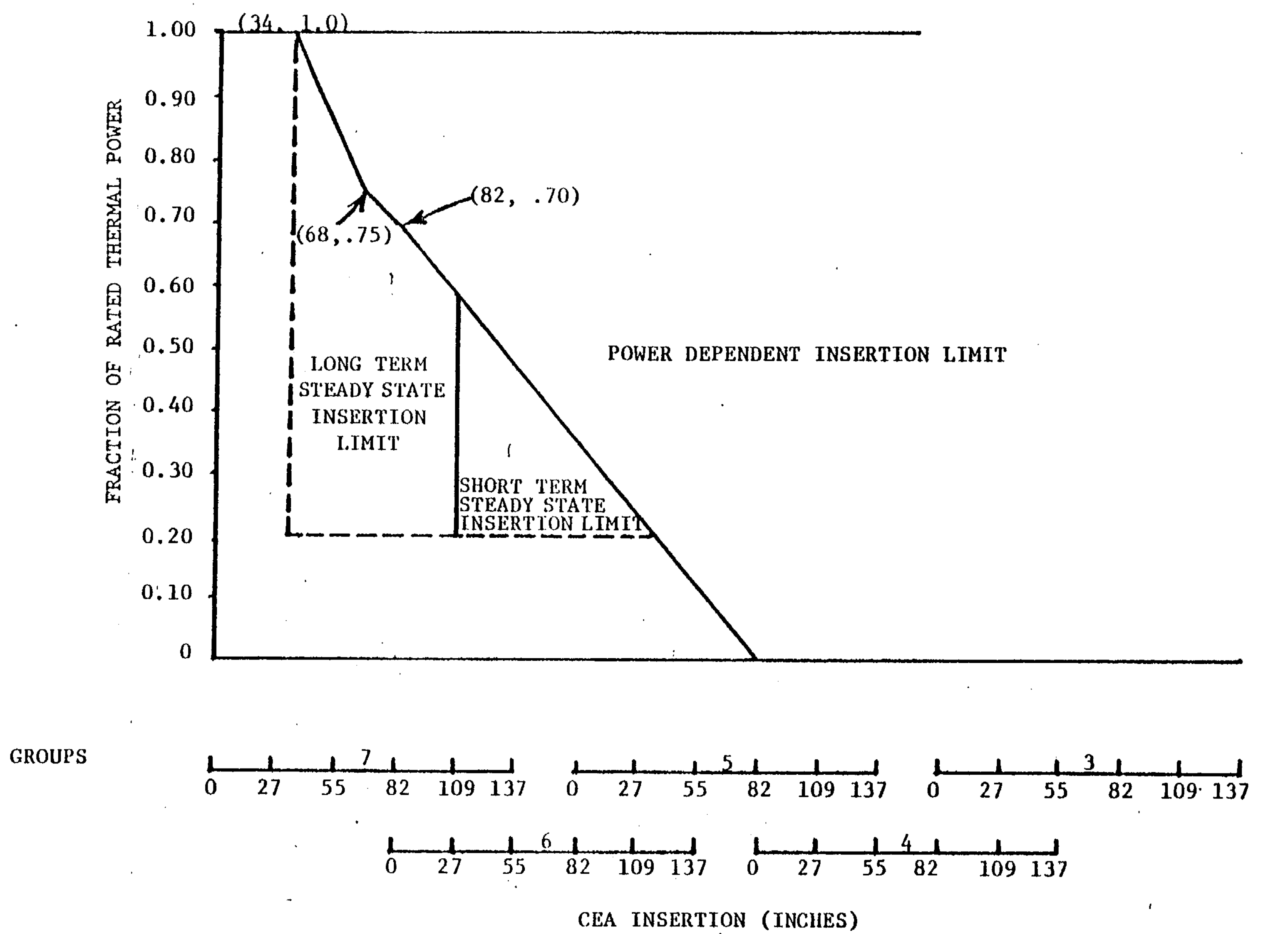

Figure 3.1-2 CEA Insertion Limits vs THERMAL POWER with 4 Reactor Coolant Pumps Operating

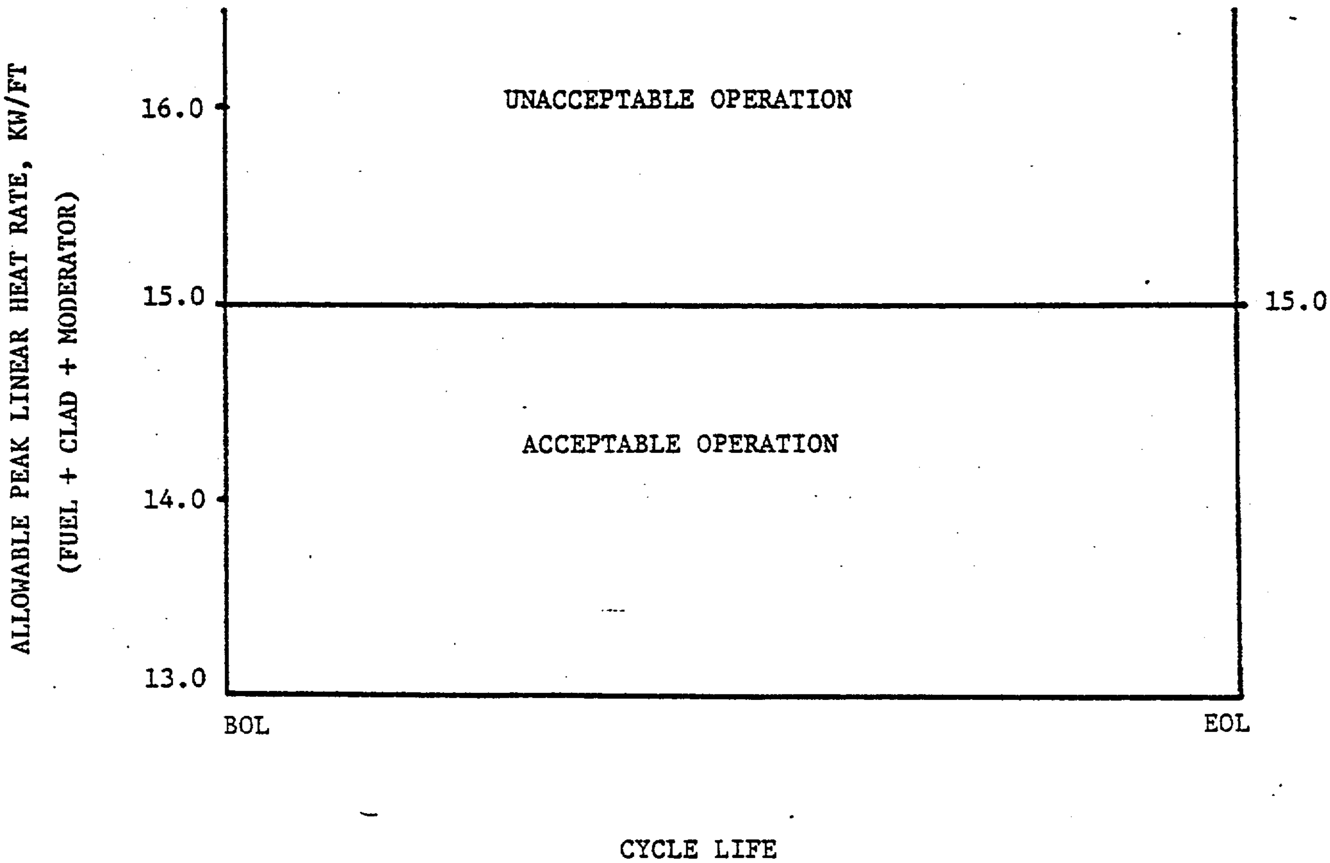

FIGURE 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

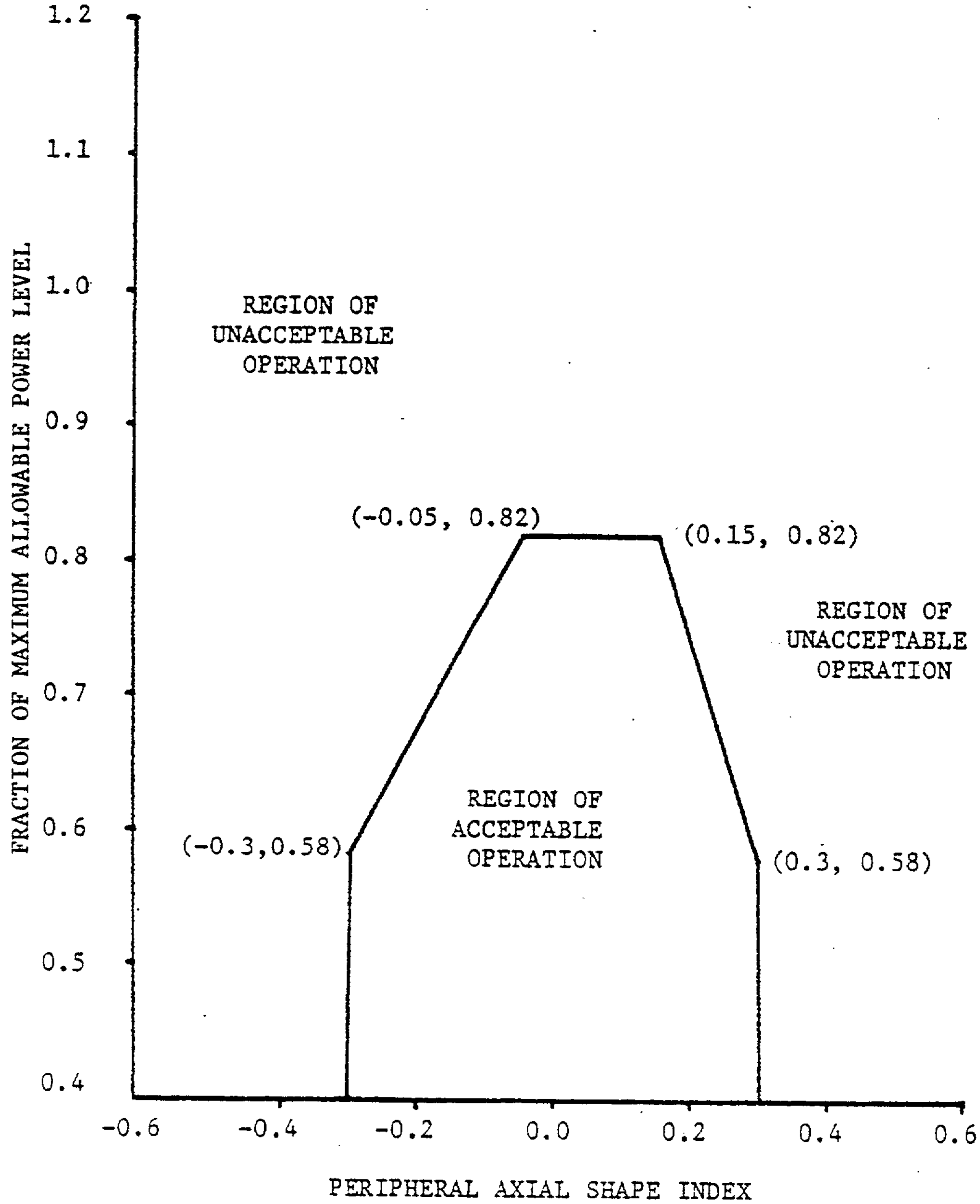

FIGURE 3.2-2

AXIAL SHAPE INDEX vs. Fraction of Maximum Allowable Power Level Per Specification 4.2.1.3

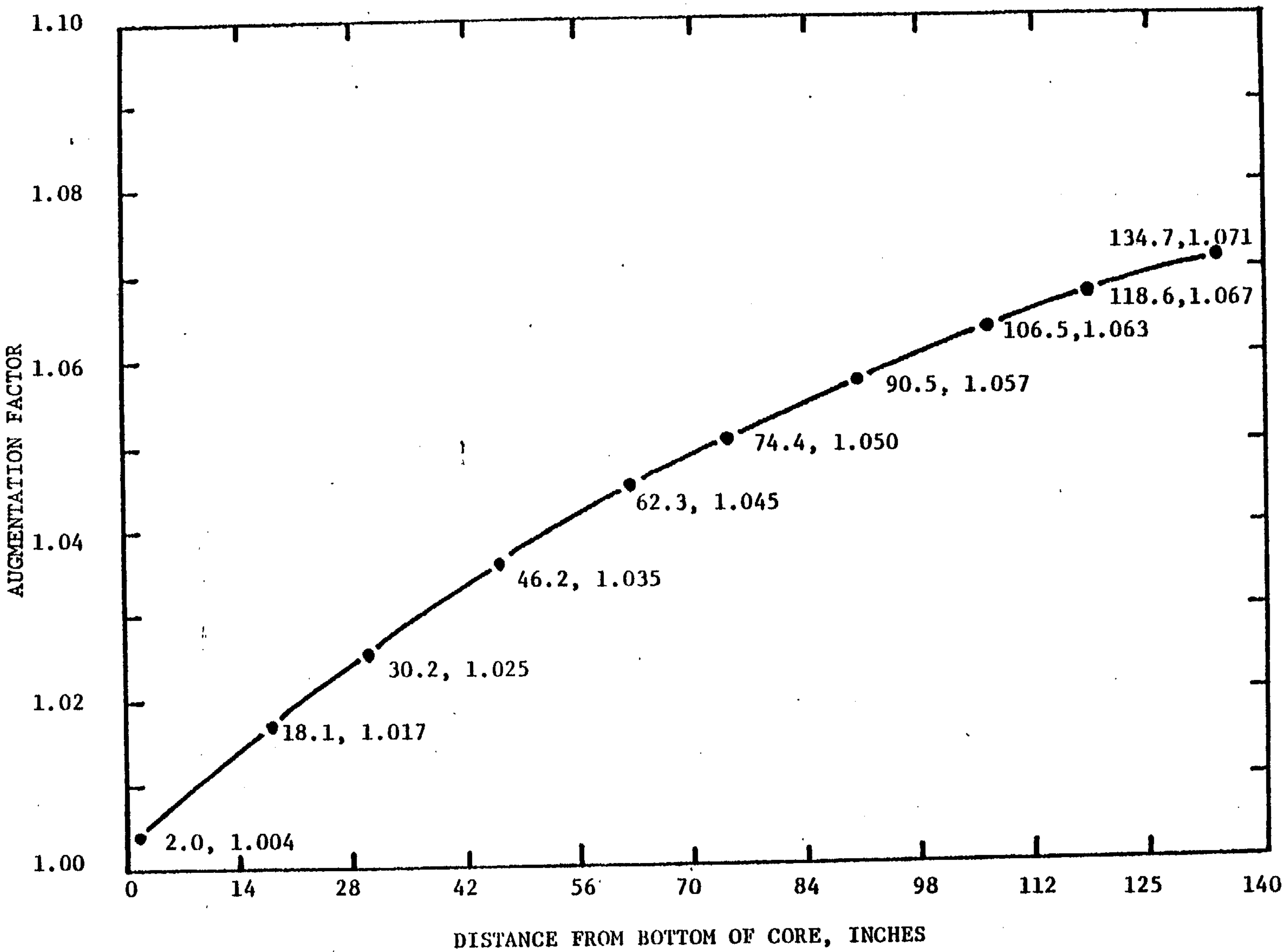

FIGURE 4.2-1

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 $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.70, 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 $F^T_{xy} = F_{xy}(1+T_q)$ when in non-LOAD FOLLOW OPERATION and by the expression $F^T_{xy} = 1.03\ F_{xy}(1+T_q)$ when in LOAD FOLLOW OPERATION. 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 > 0.03.

*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_{xy}^{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. 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_{xy}^{T} is required and the value of T_q used to determine F_{xy}^{T} shall be the measured value of T_q.

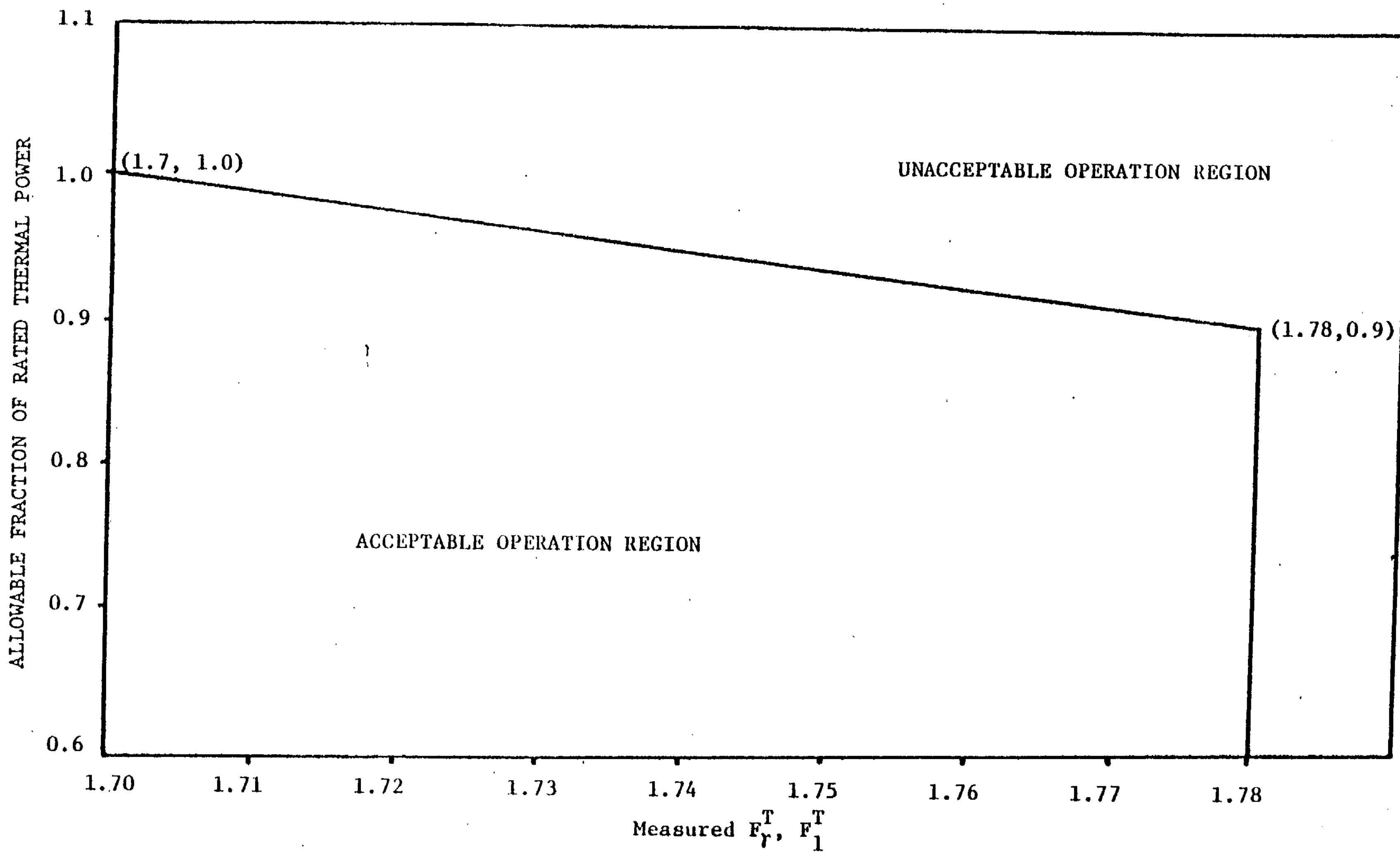

FIGURE 3.2-3
Allowable Combinations Of Thermal Power And F_r^T, F_{xy}^T

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 $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With $F_r^T > 1.70$, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to witin 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. The THERMAL POWER limit determined from Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

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 $F_r^T = F_r\ (1+T_q)$ when in non-LOAD FOLLOW OPERATION and by the expression $F_r^T = 1.02\ F_r\ (1 + T_q)$ when in LOAD FOLLOW OPERATION. 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 AXIMUTHAL POWER TILT (T_q) is > 0.03.

*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.

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	$\leq$ 549°F
Pressurizer Pressure	$\geq$ 2225 psia*
Reactor Coolant Flow Rate	$\geq$ 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.

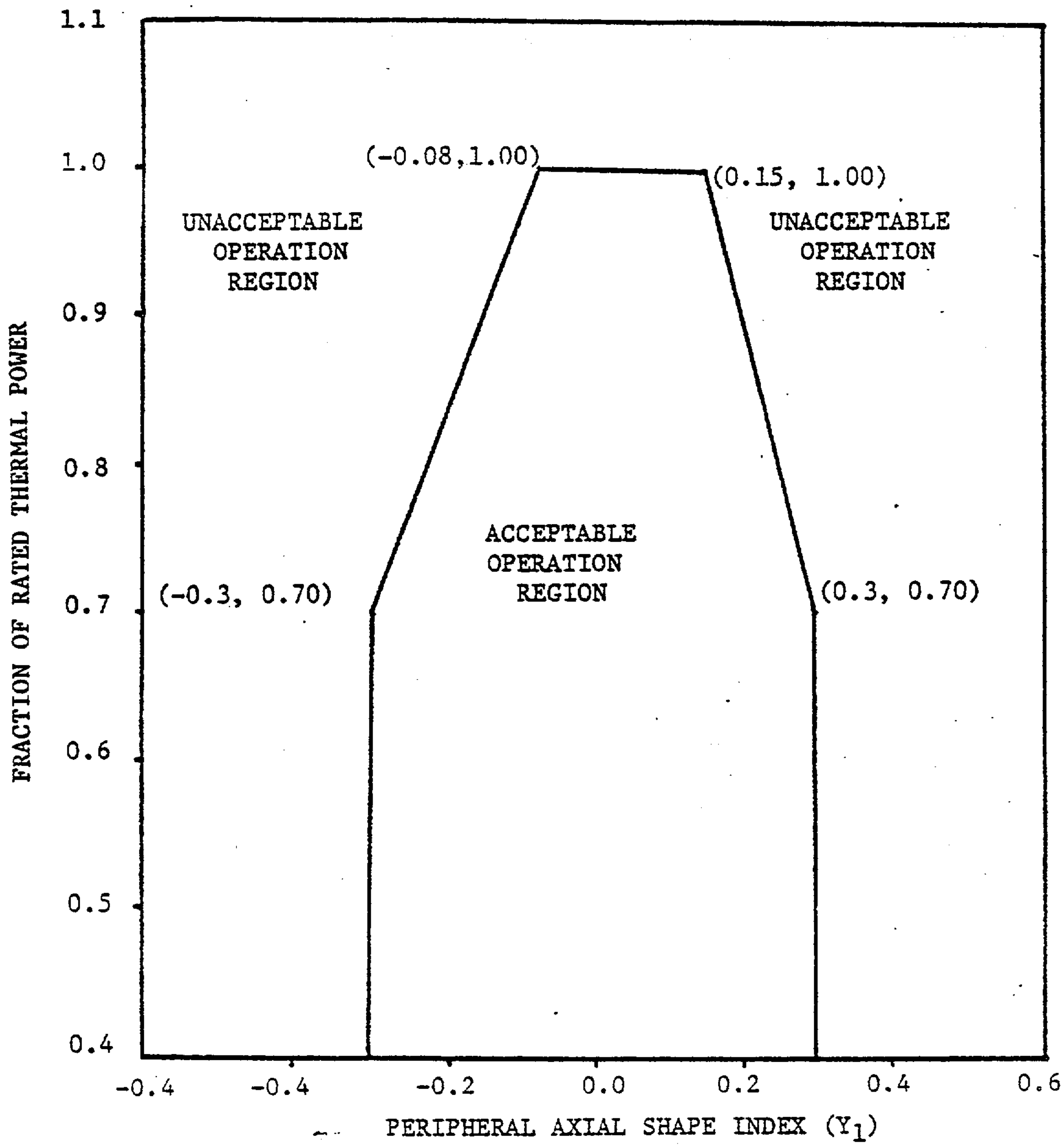

FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits With 4 Reactor Coolant Pumps Operating

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 5.0% $\Delta k/k$ 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 transient resulting from a boron dilution event with a partially drained Reactor Coolant System requires a 2% $\Delta k/k$ SHUTDOWN MARGIN and restrictions on charging pump operation to provide adequate protection. A 2% $\Delta k/k$ SHUTDOWN MARGIN is 1.0% $\Delta k/k$ conservative for Mode 5 operation with total RCS volume present, however LCO 3.1.1.2 is written conservatively for simplicity.

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 $\leq$ 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 2.0% $\Delta k/k$ 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 401,800 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.

3/4.2.1 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 1.07,* 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_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_{xy}^T 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_r^T and T_q are provided to ensure that the assumptions

---

*An uncertainty factor of 1.10 applies when in LOAD FOLLOW OPERATION.

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 1.23 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.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.23 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, and Thermal Margin/Low Pressure trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.23 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 2 x 10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

SAFETY VALVES (Continued)

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The required pressurizer heater capacity if capable of maintaining natural circulation subcooling. Operability of the heaters, which are powered by a diesel generator bus, ensures ability to maintain pressure control even with loss of offsite power.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two steam generators capable of removing decay heat, combined with the requirements of Specifications 3.7.1.1, 3.7.1.2 and 3.7.1.3 ensures adequate decay heat removal capabilities for RCS temperatures greater than 325°F if one steam generator becomes inoperable due to single failure considerations. Below 325°F, decay heat is removed by the shutdown cooling system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1025 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition and ASME Code for Pumps and Valves, Class II. The total relieving capacity for all valves on all of the steam lines is 12.38×10^6 lbs/hr which is 102.8 percent the total secondary steam flow of 12.04×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

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

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (106.5)$$

where:

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

V = maximum number of inoperable safety valves per steam line

106.5 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.192×10^6 lbs/hr.)

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.74×10^5 lbs/hr.)

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 325°F from normal operating conditions in the event of a total loss of off-site power.

Any two of the three auxiliary feedwater pumps have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 325°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 325°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The dose calculations for an assumed steam line rupture include the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

I. Introduction

By application dated November 14, 1980 (Ref. 1), Florida Power and Light Company (FPL or the licensee) requested an amendment to the license and Technical Specifications (TS) for St. Lucie Unit 1 (plant) which would allow operation at a power level of 2700 MWt. The currently authorized maximum power level is 2560 MWt. Additional submittals were made by FPL in support of this request. These are listed as References 2 through 16, 32 and 33 in section V of this evaluation.

II. Discussion

The application for the power increase (Ref. 1) was supported by analyses of plant operation at 2700 MWt using Cycle 4 parameters. Cycle 4 operation was concluded in September 1981. FPL has, in Refs. 5 and 12, updated certain analyses and the associated TS for Cycle 5 operation with NUREG-0737 related operational and design changes. These changes are automatic initiation of auxiliary feedwater flow and manual trip of reactor coolant pumps. In addition, FPL has, in Ref. 15, provided a description of the Cycle 5 core and stated that the proposed power increase analyses and TS (Ref. 1, 5 and 12) are appropriate for Cycle 5 operation at 2700 MWt. Therefore, our safety evaluation addresses the power increase analyses as updated by FPL for Cycle 5 operation.

During its 254th full committee meeting, the Advisory Committee on Reactor Safeguards (ACRS) decided not to review the St. Lucie Unit 1 power increase request since the ACRS operating license review considered this increased power level with respect to plant safety features.

Some issues involved in or related to the power increase request have already been evaluated and are the subject of separate NRC actions. These issues and our associated actions are as follows:

Issue	NRC Action
CEA Guide Tube Wear	Amendment 44 - October 14, 1981
Main Steamline Break Reanalysis	Amendment 45 - November 3, 1981
Control Room Air Intake	Amendment 38 - February 25, 1981
Asymetric Steam Generator Transient Protective Trip Function	Amendment 43 - October 14, 1981
FIESTA Code	NRC letter of April 8, 1981

Reference will be made to these actions in the appropriate sections of the safety evaluation.

III. Safety Evaluation

A. Fuel Design

The Cycle 5 core will consist of 217 fuel assemblies as follows:

Batch	No. of Assemblies	Initial Enrichment w/o U-235	No. of Shims
G^x	4	3.03	8
G*	24	3.20	8
G/	4	3.65	4
G	32	3.65	0
F	40	3.65	0
F*	48	3.03	12
E	40	3.03	0
E*	~25	2.73	0

The Cycle 5 loading pattern is illustrated in Figure 1 of Ref. 15. Batches E, F and G are identical in mechanical design. Batch F fuel was initially loaded for Cycle 4 operation and Batch G are fresh assemblies for Cycle 5. Batch E fuel was loaded for Cycle 3.

All fuel assemblies under control element assemblies (CEA) have been sleeved with a sleeve design approved by the NRC per TS 5.3.2 (Ref. 17). This satisfies our concern regarding CEA guide tube wear.

B. Nuclear Design

B.1 Evaluation

The nuclear design analysis used in Cycle 3 (reference cycle) has been used for the Cycle 4 power increase application in the same manner and with the same methods except for the use of the FIESTA computer code (Ref. 9) for the calculation of scram reactivity worths. This use of space-time kinetics methods to obtain scram worths has been approved by the NRC staff (Ref. 18).

The Cycle 4 burnup capacity is expected to be between 14,300 MWD/T and 14,900 MWD/T and the core characteristics have been examined for Cycle 3 terminations between 7250 and 8250 MWD/T. The actual termination for Cycle 3 was 7730 MWD/T, within the anticipated extremes, therefore, validating the limiting values established for the safety analyses as well as the Cycle 4 loading pattern.

The physics characterictics of Cycle 4 are shown in Table B-1 and compared to those of the reference cycle (Cycle 3).

The Cycle 4 moderator temperature coefficient (MTC) is calculated to be 0.0 for beginning of cycle and $-2.06 \times 10^{-4}\ \Delta\rho/^{o}F$ for end of cycle. These values are bounded by the values used in the safety analyses for the power increase (-2.5×10^{-4} to $+0.5 \times 10^{-4}$). Based on these analyses, FPL proposed in Ref. 1 to increase the most negative value of MTC permitted by the TS from $-2.2 \times 10^{-4}\ \Delta k/k/^{o}F$ to $-2.5 \times 10^{-4}\ \Delta k/k/^{o}F$. FPL's main steamline break (MSLB) reanalysis (Ref. 5), however, assumed a value of MTC of $-2.2 \times 10^{-4}\ \Delta k/k/^{o}F$ thereby limiting the most negative MTC allowed to that value. FPL's submittal of September 4, 1981 (Ref. 12) integrated References 1 and 5, and proposed that the current MTC limit of $-2.2 \times 10^{-4}\ \Delta k/k/^{o}F$ be retained. We have found this acceptable (Ref. 19). In addition, since the other power increase analyses used the more negative value (-2.5×10^{-4}) in a conservative manner (i.e. to give a larger positive reactivity feedback during moderator cooldown transients) the analyses remain valid.

The Doppler (fuel temperature) coefficient for Cycle 4 is slightly more negative than the value used in the reference cycle. This is a best estimate value expected to be accurate to within 15 percent. In order to assure that a conservative value was used in the safety analysis, a value 15 percent greater or less than this was used, depending upon whether a more negative or a less negative coefficient was conservative. We find the values of the Doppler coefficients to be acceptable.

At the beginning of Cycle 4, the reactivity worth of all CEAs inserted, assuming the highest worth CEA is stuck out of the core, is 7.0 percent $\Delta\rho$. The reactivity worth required for shutdown which includes the power defect from hot full power to hot zero power as well as the fact that the CEAs may be slightly inserted rather than fully withdrawn (CEA bite) is 2.4 percent $\Delta\rho$. The excess CEA worth available for normal shutdown at BOC is, therefore, 4.6 percent $\Delta\rho$. At end of Cycle 4, the calculated excess CEA worth is also 4.6 percent $\Delta\rho$. The margins available in negative reactivity at BOC and EOC are more than adequate to account for any uncertainity in nuclear calculations. We find these shutdown margins to be acceptable for Cycle 4. As a result of the MSLB reanalysis (Ref. 5) performed for Cycle 5 at 2700 MWt, FPL has proposed a required shutdown margin of 5.0 percent $\Delta k/k$. As discussed in our review of the MSLB reanalysis (Ref. 19) we have found this change acceptable.

Radial power distributions for all rods out (ARO) condition are presented for beginning, middle, and end of Cycle 4. Distributions are also presented which are representative of the upper region of the core with the insertion of the first CEA regulating group, Bank 7. Single rod power peaking values include a bias value of 4.9 percent to increase the radial peaking in fuel rods adjacent to CEA water holes. The power peaking values used in the safety analyses and the setpoint analyses are higher than those expected to occur during Cycle 4.

The augmentation factor (used to account for the power density spikes due to axial gaps caused by fuel densification) was calculated for Cycle 4 using the methodology described in Reference 20 which has been approved by the NRC staff. These augmentation factors are included in the determination of F_{xy}. The Cycle 4 calculated augmentation factors are higher than the maximum reference cycle values. These calculated values were increased for conservatism when used in the incore monitoring system, the maximum value being 1.071 as compared to the reference cycle maximum of 1.058. We find the Cycle 4 augmentation factors acceptable.

For Cycle 4 operation, the licensee has proposed measurement uncertainties of 6 percent for the total integrated radial peaking factor (F_r) and 7 percent for the total power peaking factor (F_q) for monitoring power distribution parameters. Based on our review of uncertainties in the nuclear power peaking measured by the self-powered, fixed incore detector system (Ref. 21), we find these measurement uncertainties to be acceptable.

TABLE B-1

ST. LUCIE UNIT 1 CYCLE 4 STRETCH POWER

NOMINAL PHYSICS CHARACTERISTICS

	Units	Reference Cycle	Stretch Power Cycle 4
Dissolved Boron			
Dissolved Boron Content for Criticality, CEAs Withdrawn			
Hot Full Power, Equilibrium Xenon, BOC	PPM	850	1077
Boron Worth			
Hot Full Power BOC	PPM/% $\Delta\rho$	90	104
Hot Full Power EOC	PPM/% $\Delta\rho$	80	83
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients, Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\ \Delta\rho/^{o}F$	-0.2	0.0
End of Cycle	$10^{-4}\ \Delta\rho/^{o}F$	-1.8	-2.06
Doppler Coefficient			
Hot Zero Power BOC	$10^{-5}\ \Delta\rho/^{o}F$	-1.44	-1.64
Hot Full Power BOC	$10^{-5}\ \Delta\rho/^{o}F$	-1.13	-1.26
Hot Full Power EOC	$10^{-5}\ \Delta\rho/^{o}F$	-1.22	-1.39
Total Delayed Neutron Fraction, βeff			
BOC		.0060	.0063
EOC		.0051	.0051
Neutron Generation Time, ℓ			
BOC	10^{-6} sec	28	24
EOC	10^{-6} sec	33	29

B.2 Technical Specification Changes

1. LPD/LSSS Trip

The Local Power Density Limiting Safety System Setting has been changed to reflect operation at 2700 MWt with higher radial peaking factors. The trip setpoint has been adjusted to not exceed the limit lines of Figure 2.2-2. The revised Figure 2.2-2 for Local Power Density-High Trip Setpoint is acceptable.

2. TM/LP LSSS Trip

The Thermal Margin/Low Pressure Limiting Safety System Setting has been changed to reflect operation at 2700 MWt with higher radial peaking factors. The trip setpoint has been adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. These revised figures for Thermal Margin/Low Pressure Trip Setpoint are acceptable.

3. Augmentation Factors

The incore monitoring system augmentation factors have been increased due to the higher fuel enrichment and to envelope future cycles. We find this new curve on page 3/4 2-5 acceptable.

4. Radial Peaking Factor

Total planar radial peaking factor (F^t_{xy}) and total integrated radial peaking factor (F^t_r) have been changed from 1.627 to 1.70 and from 1.64 to 1.70, respectively. This change has been evaluated in the Neutron Design section and has been found acceptable. The TS pages changed are 3/4 2-6, 3/4 2-8, and 3/4 2-9.

5. Power Dependent Insertion Limits

The PDIL is being changed to be consistent with the new LSSS. TS Figure 3.1-2, page 3/4 1-30.

C. Thermal-Hydraulic Design

C.1 Review Scope

The following reports describe the methodology changes implemented for the Cycle 4 thermal-hydraulic analyses in order to show that acceptable thermal margin is maintained at the increased power level.

(a) The TORC core thermal margin design code (Ref. 22).

This code replaces the COSMO code used in Cycle 3 and 4 analysis. This code has been approved previously by the staff (Ref. 23).

(b) CE-1 critical heat flux (CHF) correlation (Ref. 24), generic DNBR limit.

This correlation replaces the W-3 correlation used in St. Lucie 1 Cycle 3 and Cycle 4 DNBR analysis.

(c) Effects of fuel rod bow on DNBR margin (Ref. 25).

Proposed modifications on the effects of fuel rod bow on DNBR to St. Lucie 1 Cycle 3 are described in this report. This report is under review by the staff. The effects of rod bow have been considered as discussed in section C.2.2.

(d) Statistical combination of uncertainties (Refs. 6, 7 and 8).

The thermal margin methodology for St. Lucie 1 Cycle 4 power increase has been modified by the application of statistical methods instead of the application of deterministic methods applied in St. Lucie 1 Cycle 3.

The objective of this review is to confirm that the thermal hydraulic design at the stretch power rating of 2700 (compared with a design power rating of 2560 MWt for Cycle 1, 2 and 3) has been accomplished using acceptable methods, and provides acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients.

C.2 Design Methodology Evaluation

C.2.1 CE-1 Correlation (Generic Limit)

For St. Lucie 1 Cycle 4, power increase analyses, the CHF calculation has been changed from the W-3 correlation to the CE-1 correlation (Refs. 24 and 26). The CE-1 correlation has previously been approved for interim plant specific applications with a minimum DNBR limit of 1.19. Although our final generic evaluation has not been completed; the proposed limit of 1.19 for the CE-1 correlation is conservative in comparison to 14 x 14 CHF test data applicable to St. Lucie 1 and is therefore acceptable.

C.2.2 Fuel Rod Bow

The licensee has proposed a rod bow compensation of 0.6 percent on DNBR using the method described in supplement 3P to CENPD-225-P (Ref. 25) which has not yet been approved. Accordingly, it is the staff position that the approved interim method of the rod bow compensation described in Ref. 27 shall be applicable. This method permits the reduction in DNBR due to rod bowing to be offset by various credits.(Ref. 27). Using the guidelines of Ref. 27, all of the assemblies which will exceed the NRC-determined penalty threshold burnup of 24000 MWD/MTU have a maximum burnup of $\leq$37,800 MWD/MTU. The corresponding proposed DNBR penalty (Ref. 13) is 4.6 percent (the staff calculated number is 3.6 percent). The power distributions for Cycle 4 show the maximum radial peak for any of these assemblies to be at least 10 percent less than the maximum radial peak. Thus, the penalty is offset by the lower peaking of these assemblies and no power penalty for rod bowing is required for Cycle 4.

C.2.3 SCU Review

The staff, in conjunction with our contractor, Battelle Pacific Northwest Laboratories, has reviewed the SCU methodology present in CEN-123(F)-P; our evaluation is described in Appendix A to this safety evaluation. We have concluded that the SCU is acceptable with the following provisions:

1. code uncertainties of 5 percent should be included in SCU analysis;

2. pending approval of CENPD-225-P, the currently approved interim method for rod bow should be used for rod bow compensation calculation;

3. any changes in codes or correlations used in the analysis will require a re-evaluation of the SCU; and

4. there are errors in Table 3-1 of the reports (Refs. 6 and 8) which have been corrected (Ref. 28). We require that the corrected values continue to be used in future analyses.

We have concluded that the new equivalent DNBR limit is 1.23 including SCU for system parameters and excluding rod bow compensation on DNBR. Therefore the proposed DNBR limit of 1.23 is acceptable.

C.3 Comparison of Thermal Hydraulic Design Conditions

A comparison of the thermal hydraulic design conditions for St. Lucie 1 Cycle 3 and 4 is provided in Table C-1. Cycle 4 is characterized by a higher rated power level, higher design inlet temperature, and higher average linear heat rate of the fuel rods. Other differences exist in total reactor coolant mass flow, coolant flow through the core, pressure drop across the core, and average core enthalpy rise. Engineering factors on hot channel heat input and fuel densification are different for Cycle 4 compared to Cycle 3. The limiting transient (loss of flow) MDNBR value calculated with COSMO/W-3 was 1.31 for Cycle 3 compared to a MDNBR value of 1.23 calculated with TORC/CE-1 for Cycle 4 stretch power. Peak allowable linear heat generation rate is increased to 15.0 kw/ft for Cycle 4 stretch power compared to a value of 14.68 kw/ft for Cycle 3. Thus change in methodology compensated for the reduced thermal margin at the increased licensed thermal power of 2700 MWt for Cycle 4.

C.4 Technical Specification Changes

The Technical Specifications changes for the Thermal Hydraulics Section proposed for the Amendment are summarized in the following statements:

- Thechnical Specification Nos. B2.1 and B2.2, pages B2-1, B2-3, B2-5, and B2-7

W-3 DNBR correlation and MDNBR limit of 1.3 will be changed to CE-1 correlation and 1.23, respectively.

Technical Specification Table 3.2-1, page 3/4 2-14

Maximum cold leg temperature will be changed to 549°F.

Technical Specification Figure 3.2-2, page 3/4 2-4 and Figure 3.2-4, page 3/4 2-15

Figures 3.2-2 and 3.2-4 regarding Axial Shape Index will be replaced with new Figures 3.2-2 and 3.2-4, respectively, which were provided in the Cycle 4 stretch power application submittal.

Technical Specification B3/4.2.5, page B3/4 2-2 and B3/4.4.1, page B3/4 4-1

Minimum DNBR limit will be changed from "1.30" to "1.23".

Technical Specification Figure 2.1-1, page 2-2

Thermal Limit Lines have been changed to reflect 2700 MWt power operation.

These proposed modifications to the Technical Specifications in Section C.4 have been reviewed by the staff and are acceptable.

C.5 Evaluation Summary

We have reviewed St. Lucie 1 Cycle 4 power increase thermal design methodology and safety analyses as summarized below:

(a) The TORC code is acceptable for use in St. Lucie 1 safety analyses in conjunction with the CE-1 CHF correlation.

(b) The CE-1 DNBR limit for St. Lucie 1 has been evaluated. The proposed limit of 1.19 for the CE-1 correlation is conservative in comparison to 14 x 14 CHF test data and is, therefore, acceptable.

(c) Our review of SCU is complete. We have found the SCU methodology acceptable. However, a correlation cross-validation uncertainty and a 5 percent code uncertainty must be included. The approved DNBR limit is 1.23 excluding rod bow compensation.

(d) According to Section C.2.2, no rod bow compensation is required for Cycle 4.

(e) The operation of St. Lucie 1 at an increased licensed power level of 2700 MWt is acceptable.

Table C-1

St. Lucie Unit 1

Thermal-Hydraulic Parameters at Full Power

General Characteristics	Unit	Reference Cycle 3	Cycle 4 Stretch Power
Total Heat Output (core only)	MWt 10^6 BTU/hr	2560 8737	2700 9215
Fraction of Heat Generated in Fuel Rod		.975	.975
Primary System Pressure			
Nominal	psia	2250	2250
Minimum in steady state	psia	2200	2200
Maximum in steady state	psia	2300	2300
Design Inlet Temperature	°F	544	549
Total Reactor Coolant Flow (minimum steady state)	gpm 10^6 lb/hr	370,000 140.2*	370,000 139.3*
Coolant Flow Through Core	10^6 lb/hr	135.0*	134.10*
Hydraulic Diameter (nominal channel)	ft	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft^2	2.53*	2.51*
Pressure Drop Across Core (minimum steady state flow irreversible Δp over entire fuel assembly)	psi	10.3	10.4 psi
Total Pressure Drop Across Vessel (based on nominal dimensions and minimum steady state flow)	psi	33.5	33.6 psi
Core Average Heat Flux (accounts for above fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft^2	174,400	183,843
Total Heat Transfer Area (accounts for axial densification factor)	ft^2	48,860	48,872
Film Coefficient at Average Conditions	BTU/hr-ft^2 °F	5820	5820
Maximum Clad Surface Temperature	°F	657	657
Average Film Temperature Difference	°F	31	33
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kw/ft	5.83	6.14
Average Core Enthalpy Rise	BTU/lb	65*	68.7*

*Calculated at design inlet temperature, nominal primary system pressure.

Table C-1 (cont'd)

Calculational Factors	Reference Cycle 3	Cycle 4
Engineering Heat Flux Factor **	1.03	1.03
Engineering Factor on Hot Channel Heat Input **	1.03	1.02*
Inlet Plenum Nonuniform Distribution	1.05	Not applicable
Rod Pitch, Bowing and Clad Diameter**	1.065	1.065
Fuel Densification Factor (axial)	1.01	1.002
Fuel Rod Bowing Augmentation Factor on Fr	1.018	1.018
Limiting Transient (Loss of Flow) MDNBR	1.31 (COSMO/W-3)	1.23 (TORC/CE-1)
Peak Allowable Linear Heat Rate (kw/ft)	14.68	15.0

*Based on "Asbuilt" information.

**For cycle 4 these factors have been combined statistically with our uncertainty factors at 95/95 confidence/probability level (Ref. 29) to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 29.

D. Accident Analyses

We have evaluated FPL's analyses of accidents and Anticipated Operational Occurrences (AOO). The section numbering in this part of our evaluation corresponds with the event numbering in FPL's power increase request (Ref. 1).

7.1.1 Boron Dilution

Boron Dilution events are examined for all modes of operation against the acceptance criteria of SRP section 15.4.6. If operator action is required to terminate the transient, the acceptance criteria specify that a minimum time interval of 15 minutes (30 minutes if in refueling mode) must be available between the time when an alarm announces an unplanned moderator dilution and the time of loss of shutdown margin.

The St. Lucie boron dilution analyses are presented in references 1 and 16 and the technical specification changes associated with mode 5 are presented in references 1 and 14. All times presented in the analyses are the time intervals from start of the dilution event to the loss of shutdown margin. Also, as stated in reference 12, St. Lucie has indications for boron dilution, but not alarms. This is an exception to SRP 15.4.6.

We are presently evaluating the capability of operating PWRs to provide adequate protection against uncontrolled boron dilution events. Pending the results of that evaluation, we find FPL's boron dilution analysis acceptable if either:

(1) An alarm is available to alert the operator to boron dilution events; or

(2) For an unmitigated boron dilution event, (a) the DNBR does not fall below the minimum acceptable DNBR, (b) the primary system pressure does not exceed 110% of the design pressure, and (c) the pressure-temperature limits of Appendix G are not violated for all postulated unmitigated boron dilution events.

Operation of St. Lucie Unit 1 at increased power (2700 MWt) is acceptable if FPL provides a commitment to perform item 1 or 2 above prior to startup after the next (Cycle 6) refueling outage. Pending receipt of this commitment, we consider this item resolved. Operation is justified because the following indications are available to the operator to detect a boron dilution event:

(1) boronometer (on letdown line)

(2) source range indication and audible count rate meter

(3) low volume control tank level.

In addition, normal operating procedures do not align diluting sources of water, and charging pump operation is not normal for mode 5, which is the most limiting boron dilution case examined. The shutdown margin is increased to 2% $\Delta k/k$ which provides more time to react to a boron dilution event than previously existed. Finally, one charging pump is rendered inoperable when the RCS is drained below the hot leg centerline, reducing the capacity for boron dilution.

In reference 32 FPL commited to install start up flux channel alarms for the detection of boron dilution events by the next (Cycle 6) refueling outage. FPL stated that design details would be submitted 90 days prior to the Cycle 6 refueling outage. This alarm would be effective in modes 3-6. During modes 1 and 2 this event would be mitigated by one of the reactor protection system trips. In addition, during modes 1 and 2, the transient time is a number of hours. Therefore, we find FPL's commitment acceptable.

7.1.2 Startup of an Inactive Reactor Coolant Pump

The Startup of an Inactive Reactor Coolant Pump event was not analyzed for Cycle 4 power increase because Technical Specifications do not permit operation at power with less than 4 Reactor Coolant Pumps operating.

This is acceptable.

7.1.3 Excess Load Event

The Excess Load Event is evaluated in accordance with SRP section 15.1.1 to assure that the response of the primary system to the ensuing cooldown will not exceed acceptance limits for DNBR, excess power, or overpressure. SRP section 15.5.1 specifies conservative assumptions that should be used in the analysis including the initial power level, scram characteristics, core burnup, and the response of safety systems.

The St. Lucie Unit 1 analysis (ref. 12) assumed the complete opening of the steam dump and bypass valves during power operation. The assumed moderator temperature coefficient of reactivity is more negative than the Cycle 5 Technical Specification value, and the fuel temperature coefficient is the least positive value. Initial power assumed is 102%.

Reactor trip is assumed to be generated by a high power level trip (112% power) 8.4 seconds after opening of the dump valves. The analysis shows a peak linear power of 18.3 kw/ft (a value less than centerline melt) and a minimum DNBR of 1.29, which meets the acceptance limit of 1.23. Peak pressure was less than 110%. The safety injection signal was actuated by low pressurizer pressure, and reactor coolant pumps were tripped in accordance with TMI guidelines.

Based on the above conservative assumptions and acceptable results, we conclude that the excess load event has been satisfactorily analyzed for St. Lucie Unit 1.

7.1.4 Loss of Load

The Loss of Load Event causes a primary system heatup that is examined to assure that RCS pressure remains below 110% of design pressure and that DNBR limits are not reached. SRP section 15.2.1 contains acceptance criteria and review procedures for this event.

The analysis (ref. 1) assumed an initial power of 2754 MWt and temperature of 551°F. The most positive moderator coefficient was used, as was the least negative fuel temperature coefficient. These assumptions help to mask negative reactivity feedback, and increase the peak of the pressure transient. A lower-than-normal initial pressure of 2200 psi was used to delay the reactor trip signal, which was assumed to occur on a high pressure signal. The analysis with delayed trip due to low initial pressure causes a greater peak pressure than does an earlier trip with a higher initial pressure.

The results of the analysis show a peak pressure of 2572 psi with no credit given for operation of the PORV's. This value is below 110% of design pressure (2750 psia), and the minimum DNBR is 1.48, which is acceptable. Secondary side pressures are also maintained below 110% of design pressure.

Therefore, the St. Lucie Unit 1 analysis of the Loss of Load Event meets SRP section 15.2.1 acceptance criteria and is acceptable.

7.1.5 Loss of Feedwater Flow

The Loss of Feedwater flow event is evaluated to determine that the resulting primary side heatup transient does not exceed the acceptance criteria in SRP section 15.2.7. These criteria require that pressure should not exceed 110% of design pressure and that DNBR limits are met. In addition, conservative requirements on certain plant parameters and initial conditions should be observed in the analysis.

The St. Lucie Unit 1 power increase analysis (ref. 1) assumes an initial power at 2754 MWt, temperature at 551°F and primary pressure at 2200 psia. A low initial pressure was used to delay reactor trip and maximize the pressure overshoot. Two events were evaluated - one where the primary side pressure was maximized and one where steam generator dryout time was minimized. To maximize primary pressure, pressurizer spray and relief valves were inoperative as was the steam dump system. The resulting peak pressure was 2506 psia, and the DNBR was 1.52. The case where dryout time was minimized assumed operable steam dump and bypass valves, and pressurizer spray and relief valves. This case showed that approximately 15 minutes are needed to dry out a steam generator with a loss of Main Feedwater.

Both events show results in compliance with SRP section 15.2.7 and are acceptable.

7.1.6 Feedwater Malfunctions

The FW Malfunction is evaluated against the criteria in SRP section 15.1.1. Since the inadvertent opening of steam dump and bypass valves results in a greater heat removal rate than does a loss of FW heaters, or excess feedwater flow, this event is bounded by the Excess Load Event 7.1.3 and is acceptable.

7.2.1 Control Element Assembly Withdrawal Event

The CEA withdrawal event was reanalyzed for the power increase to determine the initial margins that must be maintained by the Technical Specification LCO limits such that in conjunction with the Reactor Protection System (RPS) the DNBR and fuel centerline-to-melt (CTM) design limits will not be exceeded. The reclassification of this event from the category requiring the action of Thermal Margin/Low Pressure (TM/LP) and Axial Shape Index (ASI) trips to the category where sufficient initial steady state thermal margin is built into the DNB and LHR LCOs such that credit need only be taken for either the High Power Trip (HPT) or the Variable High Power Trip (VHPT) was presented in reference 10. We have found this reclassification acceptable. Our review is attached as Appendix B. The event was reanalyzed for reactor initial conditions of zero power and full power and the licensee has stated that the DNB and CTM limits will not be exceeded.

The methods used to determine the peak fuel rod responses, and the input to that analysis, such as reactivity insertion rate, moderator and fuel temperature feedback effects, and initial axial power distribution, have been examined. The results of the analysis show that the DNB and CTM SAFDLs will not be exceeded during a CEA withdrawal event.

The staff concludes that the calculations contain sufficient conservatism, in both input assumptions and models, to assure that fuel damage will not result from CEA withdrawal transients.

7.2.2 Loss of Coolant Flow Event

The Loss of Coolant Flow Event is examined to assure that DNBR limits are not exceeded upon a complete or partial loss of coolant flow. The applicable SRP section is 15.3.1 which requires that reactor coolant and main steam pressures remain less than 110% of design pressure, and that DNBR limits not be exceeded.

The St. Lucie Unit 1 analysis of the loss of 4 coolant pumps (ref. 1) assumes nominal initial conditions at full power, inlet temperature is 549^{O}F and pressure of 2225 psi. Core parameters are conservative beginning of life values, positive moderator coefficient and the limiting axial and radial factors. The analysis shows that the low flow trip setpoint of 93% is reached in less than one second. At 2.5 seconds, the minimum DNBR of 1.23 is reached. Peak reactor pressure of 2326 psi is below the design pressure.

Since the minimum DNBR limit and peak pressure limit were not exceeded, we conclude that this evaluation meets SRP section 15.3.1 acceptance criteria and is acceptable.

7.2.3 Loss of Non-Emergency AC Power Event

The Loss of Non-Emergency AC Power Event is similar to the loss-of flow event, except that the secondary side feedwater and steam flows are also lost. With a loss of offsite power, secondary side cooling is performed by releasing steam through the atmospheric dump valves, so a site boundary dose analysis is performed. Initial conditions and assumptions were adjusted to maximize offsite doses. SRP section 15.2.6 guides our review.

Assumptions for the Loss of Non-Emergency AC Power (ref. 1) are the same as for the loss of flow event with the following exceptions: Power level is 102%, inlet temperature is 551°F, pressure is 2300 psi. In addition, the steam generator initial pressure is raised to 909 psi to increase steam release rates.

Results of the analysis show peak pressure of 2534 psia on the primary side and 1034 psi on the steam generator; both are below 110% of design pressure. The DNBR of the Loss of Flow Event (7.2.2) applies to this event since the minimum DNBR is reached before the effects of a loss of feedwater became significant. The offsite doses calculated are a small fraction of 10 CFR 100 guidelines.

FPL's analysis of this event was performed using conservative assumptions; however based on our review we find that a confirmatory analysis must be performed which considers the worst single active failure in conjunction with this event.

Operation at 2700 MWt may proceed while this confirmatory analysis is being performed. This conclusion is based on the small offsite doses calculated in the current analysis and the fact that the minimum DNBR occurs within a few seconds of event initiation. Therefore, minimum DNBR should not be affected by a single active failure and no fuel rod failures will need to be considered.

FPL has agreed to provide the requested analysis within 6 months of issuance of this amendment (ref. 33).

7.2.4 Full Length CEA Drop Event

The full length CEA drop event was reanalyzed to determine the initial thermal margins that must be maintained by the LCOs such that the DNBR and fuel centerline melt design limit will not be exceeded. The methods used to determine the peak fuel rod response, and the input to that analysis such as power distribution changes, CEA reactivities, and reactivity feed-back effects due to moderator and fuel temperature changes, have been examined.

The resulting extreme conditions of fuel power, temperature, and DNB have been compared to the acceptance criteria for fuel integrity and the analyses have shown that these limits are not exceeded.

The staff concludes that the calculations contain sufficient conservatism, in both input assumptions and models, to assure that fuel damage will not result from a full length CEA drop.

7.2.5 and 7.2.6 Part Length CEA AOO's

Part length CEAs have been removed. No evaluation is necessary.

7.2.7 Anticipated Operational Occurrences Resulting from the Malfunction of One Steam Generator

The analysis for this event was recently reviewed and approved for a power level of 2611 MWt (Ref. 31). The setpoints for this trip function remain the same for the power level increase, and the analyzed initial conditions are the same with the exceptions of the new power level, higher inlet temperatures, and a more conservative moderator coefficient. The resulting DNB for the transient is 1.42, which is acceptable. Peak steam generator pressure is 1063 psia, which is below the 110% of design pressure limit. We therefore, conclude that this analysis is acceptable.

7.3.1 CEA Ejection Event

The CEA ejection event was reanalyzed to assess the impact of changes in power peaking factors, ejected CEA worth, steady state linear heat rate, and delayed neutron fraction from the reference cycle. The analytical method employed in the reanalysis of this event is the NRC approved CE method described in Reference 30.

The most limiting key safety parameters in Cycle 4 were used to bound the most adverse conditions. These included the least negative Doppler coefficient, the most positive moderator temperature coefficient, and an EOC delayed neutron fraction to produce the highest power rise during the event.

FPL's analysis shows that both the zero power and full power cases result in peak fuel enthalpies less than the NRC limiting criterion of 280 cal/gm for pressure pulse and coolability considerations. Therefore, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur.

We conclude that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

7.3.2 Steamline Rupture Event

The steamline rupture event has been reviewed and approved for the power increase (Ref. 19) and is acceptable.

7.3.3 Steam Generator Tube Rupture

See Section E.4 of this safety evaluation.

7.3.4 Seized Rotor

The Seized Rotor Event assumes a complete stoppage of one coolant pump, which results in a rapid reduction of core flow to the three pump value. The SRP section covering this review, section 15.3.3 Acceptance Criteria, requires peak system pressure to be less than 110% of design pressure, and that a limited number of fuel failures would be allowed.

The Seized Rotor analysis (ref. 1), using the same initial conditions as the loss of coolant flow event, include conservative physics parameters, and a scram with the most reactive control rod stuck out. Results of the analysis show a peak pressure of 2306 psia and a minimum DNBR of 1.025. This minimum DNBR results in a predicted failure of 1.06% of the fuel rods using previously approved methods. The number of fuel failures is sufficiently limited to conclude that control rod insertability will be maintained, and that no loss of core cooling capability will result. Furthermore, this represents an insignificant amount of fuel failure with respect to offsite doses for this accident.

SRP acceptance criteria for the seized rotor event require consideration of a loss of offsite power coincident with turbine trip. Credit for suitable delays in loss of offsite power after the turbine trip may be assumed if justified. In addition, a worst single active failure should be considered, on either the primary or secondary systems. The current analysis does not consider a loss of offsite power and may not consider the worst single failure which is an exception to the SRP. Confirmatory analysis of the locked rotor event with a single failure and loss of offsite power is needed. FPL, in ref. 33, has agreed to provide this confirmatory analysis within 6 months of issuance of this amendment.

Interim operation at stretch power is acceptable pending resolution of this issue because the probability of a combined rotor seizure event with loss of offsite power is very low.

8.0 Loss of Coolant Accident

Loss of Coolant Accidents are examined to assure that St. Lucie Unit 1 meets the acceptance criteria of 10 CFR 50.46 using methods which are in conformance with 10 CFR 50 Appendix K. The applicable SRP section is 15.6.5. The required acceptance criteria are: 1) peak clad temperature less than 2200^{o}F; 2) peak cladding oxidation less than 17%; 3) total core wide clad oxidation less than 1%; 4) calculated geometry changes in core are such that the core remains amenable to cooling; 5) long term core cooling is maintained.

Small break LOCA's have been examined in the St. Lucie FSAR and are being re-evaluated as part of the TMI action plan. The results thus far continue to verify that the large break LOCA is a more limiting event.

The computer codes used to evaluate the large break LOCA include CEFLASH-4A for blowdown calculation, COMPERC-II for reflood, STRIKIN-II and PARCH for clad oxidation and peak clad temperature calculations.

The St. Lucie Unit 1 LOCA analysis (ref. 1) was performed for both slot and guillotine breaks in the pump discharge leg with areas of 100%, 80% and 60% of double pipe area. This location and range of break sizes traditionally generate the highest peak clad temperatures. The core parameters such as 102% power, inlet temperatures of 551^{o}F, peak pin burnup (1522 MWD/MTU) were chosen to maximize peak clad temperature.

Results of the analysis show that a peak clad temperature of 2176^{o}F was reached for the case of a double-ended guillotine break. Peak local oxidation occurred at the same rod with a value of 15.44%. Overall clad oxidation was 0.74%. These values are within acceptable limits of 10 CFR 50.46. The methods and codes used in the analysis have been previously approved.

We conclude that the applicable acceptance criteria have been met and that the St. Lucie Unit 1 LOCA analysis is acceptable.

D.2 Technical Specifications

Peak Linear Heat Rate - the allowable peak linear heat rate is increased from 14.68 kw/ft to 15.0 kw/ft to be consistent with the ECCS analysis Technical Specification Figure 3.2-1, page 3/4 2-3.

Rated Thermal Power Level - Change rated thermal power level from 2560 MWt to 2700 MWt. License paragraph 2.C.1 and TS 1.3 page 1-1.

Shutdown Margin for T-AVG Below 200^{o}F - Change required shutdown margin T-AVG below 200^{o}F from 1% $\Delta k/k$ to 2% $\Delta k/k$ and require that at least one charging pump be inoperable when, in Mode 5, the RCS is drained below the hot leg centerline. Technical Specifications 3.1.2.2 (page 3/4 1-10); 3.1.2.8 (page 3/4 1-18); B 3/4 1.1.1 & 2 (page B 3/4 1-1); and B 3/4 1.1.4 (page B 3/4 1-2).

E. Radiological Consequences of Postulated Accidents

We have evaluated FPL's proposed power increase with respect to the radiological consequences of postulated accidents.

E.1 Loss-Of-Coolant Accident (LOCA)

The design basis LOCA was evaluated in the Supplement No. 1 of the staff's safety evaluation report dated May 9, 1975. That evaluation, based on 2700 MW thermal and with the facility modified by upgrading ESF filter efficiencies and adding a NaOH spray additive system, shows that the doses resulting from a design basis LOCA will not exceed the guidelines of 10 CFR Part 100.

E.2 Fuel Handling Accidents

We have reviewed the evaluation of the consequences of the postulated fuel handling accidents in the spent fuel pool reported in the staff's safety evaluation report (SER) for St. Lucie Unit 1 licensing, dated November 7, 1974. These accidents were evaluated for a core power level of 2700 MW thermal. Therefore, the conclusions reached in the SER do not change and the dose consequences are within the guidelines of 10 CFR 100.

The staff analysis dated April 11, 1979, of the consequences of Fuel Handling Accidents inside the Containment (based on power level of 2700 Mwt) also showed that the resultant doses are within the 10 CFR 100 guidelines.

E.3 Rod Ejection Accidents

We have reviewed the evaluation of the rod ejection accidents presented in the staff's SER and find that the dose consequences were calculated for fission product release through the containment and through steam generator leakage for a core thermal power level of 2700 MW. Based on our review we conclude that the calculated doses reported in the SER meet the guidelines of 10 CFR Part 100 and, therefore, are acceptable.

E.4 Steam Generator Tube Rupture

In reference 12 FPL submitted an analysis of the steam generator tube rupture (SGTR) event. This was reanalyzed for Cycle 5 to include the effects of the NUREG-0737 related changes discussed in Section II of this evaluation. In addition to evaluating FPL's analysis we performed an independent calculation of the doses from this event.

A steam generator tube rupture (SGTR) accident releases primary coolant to the secondary side of a steam generator, thus providina a pathway for iodine and noble gases from the primary coolant to be released to the environment. The staff evaluated the radiological consequences of the release to the environment, both with and without loss of offsite power, and both with a consequential iodine spike (i.e., a temporarv rapid increase in rate of fuel rod leakage) and with a pre-existing iodine spike.

The applicant's description of the steam generator tube failure accident was reviewed, including the assumptions of the thermal hydraulic transient, the sequence of events, the bases for operator action in isolating the steam generator, and the effects of offsite power loss. The signals available to the operator are sufficient to ensure that the affected steam generator will be isolated within 30 minutes, thus limiting the release of radionuclides to the environment. The descriptions of the plant transients and sequence of events are sufficient to ensure that the most conservative type of SGTR was selected, namely, a continuous leak from the rupture for some time before a reactor scram, and loss of offsite power coincident with the scram.

The doses that the applicant calculated to result from this accident meet the guidelines of Standard Review Plan Setion 15.6.3 and 10 CFR Part 100. The staff independently calculated the doses from this accident and determined that the rupture location which would result in the greatest release would be the top of the tube bundle, where scrubbing of iodine by the secondary side liquid would be at a minimum. For a leak at the top of the tube bundle, iodine from the primary side could be released either in the vapor or in droplets formed during the flashing that occurs at the rupture. There is some scrubbing of the iodine by the two-phase mixture of secondary fluid above the top of the tube bundle, and it was assumed that the effect of this partial scrubbing could be bounded by taking the fraction of iodine released to be the flashing fraction or 10%, whichever is greater. The flow rate from the rupture was determined by assuming a double-ended guillotine break, and basing the pressure drop on entrance and exit losses, and viscous pressure drop for both one- and two-phase flow.

The condenser is available until the reactor scrams on low pressurizer level, and then we have assumed that atmospheric dump valves (ADV) are used for heat removel thus providing a more direct path to the environment. After the affected steam generator is isolated at thirty minutes, heat is removed only through the unaffected steam generators's ADV's, until the operator can initiate shutdown cooling at two hours, ten minutes. A leak of one gpm (technical specifications limit) to the unaffected steam generator is assumed to occur. Other assumptions that were used in this calculation are listed in Table E-2.

The calculated doses are summarized in Table E-1 where Case 1 is that based on no pre-accident iodine spike (only a coincident iodine spike), and Case 2 is calculated assuming a pre-accident iodine spike. Only the doses for loss of offsite power following reactor scram are presented; doses with offsite power available are less.

The staff concludes that the distances to the exclusion area and to the low population zone outer boundaries for the St. Lucie site, in conjunction with the operation of the dose mitigatin ESF systems, are sufficient to provided reasonable assurance that the calculated radiological consequences of a postulated steam generator tube failure accident at Unit 1 do not exceed: (a) the exposure guidelines as set forth in 10 CFR Part 100, Section II, for the accident with an assumed pre-accident iodine spike (Case 2) and (b) 10 percent of these exposure guidelines, for the accident generated iodine spike (Case 1).

The staff conclusion is based on (1) the staff review of the licensee's analysis of the radiological consequences, (2) the independent dose calculations by the staff using conservative assumptions, including atmospheric dispersion factors as presented in Table E-2, and (3) the Technical Specification limit for primary to secondary leakage in the steam generators.

E.5 Control Room Habitability

Amendment 38 to the St. Lucie Unit 1 license, dated February 25, 1981, contains our evaluation of FPL's proposed change to the control room outside air intake limit; an increase from 100 to 450 CFM. Our evaluation concludes that the control room ventilation system is acceptable for normal and emergency operation and the radiological doses resulting from accidents will meet the guidelines of GDC 19. Since the release associated with the design basis LOCA has not changed for the power increase, our conclusions regarding control room habitability remain valid.

E.6 Summary

The potential radiological consequences of design basis accidents have been evaluated at the proposed power level of 2700 MWt and are acceptable.

TABLE E-1

RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE ACCIDENT

	0-2 Hour Doses, Exlcusion Area Boundary, rems		0-30 Day Doses, Low Population Zone, rems	
	Thyroid	Whole Body	Thyroid	Whole Body
Case 1, no pre-accident spike	1.1	> 1	0.4	> 1
Case 2, pre-accident iodine spike	10	> 1	4	> 1

TABLE E-2

ASSUMPTIONS AND BASES FOR STEAM GENERATOR TUBE FAILURE DOSES

For Case 1, no pre-accident iodine spike

Rupture location: top of tube bundle	
Length of tube for pressure drop calculation:	40 feet
Velocity of water in ruptured tube, based on one phase, initial	150 feet/sec.
Total mass flow rate of water out both sides of double-ended rupture, initial	28.6 lbs/sec.
Fraction of iodine in leaked coolant that becomes airborne (flashed or as aerosol)	
prior to scram, fraction that reaches condenser	0.136
after scram, fraction released to environment	0.1
Decontamination factor for condenser	10.
Concentration of iodine in coolant, initial	1.0 µCi/g
increasing during pressure-transient spike	18.5 µCi/g-hr
Fraction of iodine in primary coolant that mixes with secondary water that is converted to organic iodine	0.01
Fraction of organic iodine released to environment	1.0
Duration of leak, prior to scram	9.6 minutes
Time that safety valves or atmospheric dump valves are open in affected steam generator (isolation of affected steam generator occurs at 30 minutes)	20.4 minutes
Time that atmospheric dump valves are open in unaffected steam generator	121 minutes
Primary to secondary leak rate to unaffected steam generator	1.0 gpm

For Case 2, with a pre-accident iodine spike

As above, except:

Concentration of iodine in primary coolant, initial	60 µCi/g
increasing at	18.5 µCi/(g-hr)

Assumptions for whole-body dose calculations

All noble-gases, as Dose Equivalent Xe-133, that are released out tube rupture are released to environment. Concentration is at maximum allowable by technical specifications (100/E) µCi/gm.

X/Q values

0-2 hours at 1560 meters = 1.6×10^{-4} sec/m^3
0-8 hours at 1610 meters = 6.7×10^{-5} sec/m^3

F. Safety 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.

Principal Reviewers:

Larry Kopp	CPB (Physics), DSI
Suresh Gupta	CPB (Thermal Hydraulics), DSI
Gus Alberthal	RSB, DSI
Mohan Thadani	AEB, DSI
Chris Nelson	ORB#3, DL

IV. Environmental Impact Appraisal

A. Description of Proposed Action

By letter dated November 14, 1981, Florida Power and Light Company (the licensee) requested an amendment to Operating License No. DPR-67 for St. Lucie 1 to allow operation at 2700 MW thermal power level.

B. Environmental Impact of Proposed Action

The NRC has evaluated the potential environmental impact associated with the proposed license amendment as required by the National Environmental Policy Act (NEPA) and 10 CFR Part 51.

We have reviewed the Final Environmental Statement (FES) of June 1973, related to the operation of St. Lucie Unit 1. Although initial plant operation was to be at 2560 MW thermal, the FES considered plant operation at 2700 MW thermal. Therefore, the environmental impacts, both radiological and non-radiological, of plant operation at 2700 MW thermal have been reviewed and found to be acceptable.

The licensee has not, as part of this change, requested any modifications to the Appendix B environmental technical specifications. Therefore, approval of the power increase will not authorize an increase in radioactive effluents from the plant.

C. Conclusion and Basis for Negative Declaration

On the basis of the NRC evaluation and information supplied by the licensee, it is concluded that the implementation of the proposed amendment to Operating License DPR-67 will have no environmental impact other than that which has already been predicted and described in the Commission's Final Environmental Statement for the Facility dated June 1973.

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for the proposed license amendment and that a Negative Declaration to that effect should be issued.

Dated:

V. References

1. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Application for Stretch Power, dated November 14, 1980.

2. Letter from R. E. Uhrig (FPL) to R. A. Clark (NRC), Additional Information, dated June 11, 1981.

3. Letter from R. E. Uhrig (FPL) to R. A. Clark (NRC), Additional Information, dated June 24, 1981.

4. Letter from R. E. Uhrig (FPL) to R. A. Clark (NRC), Additional Information, dated July 6, 1981.

5. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), proposed license amendment regarding Shutdown Margin, Main Steam Isolation System Changes and Control Element Assembly Sleeving, dated July 23, 1981.

6. CEN-123(F)-P, "Statistical Combination of Uncertainties, Part 1," dated December 1979.

7. CEN-123(F)-P, "Statistical Combination of Uncertainties, Part 2," dated January 1980.

8. CEN-123(F)-P, "Statistical Combination of Uncertainties, Part 3," dated February 1980.

9. CEN-122(F), "FIESTA, A One Dimensional, Two Group Space-Time Kinetics Code for Calculating PWR Scram Reactivities," November 1979.

10. CEN-126(F), "CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected System," January 1980.

11. Letter from R. E. Uhrig (FPL) to R. A. Clark (NRC), Additional Information, dated August 13, 1981.

12. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Analyses on Excess Load Event and Steam Generator Tube Rupture, dated September 4, 1981.

13. Letter from R. E. Uhrig (FPL) to R. A. Clark (NRC), "Responses to NRC Questions," dated September 18, 1981.

14. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Propose Technical Specification Changes re. Mode 5 Shutdown Margin Requirements, dated September 28, 1981.

15. Letter from R. E. Uhrig (FPL) to R. A. Clark (NRC), Stretch Power Operation During Cycle 5, dated October 8, 1981.

16. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Additional Information re. Shutdown Margin Requirements, dated October 8, 1981.

17. Amendment 44 to St. Lucie Unit 1 Operating License DPR-67, CEA Guide Tube Sleeves, dated October 14, 1981.

18. Letter from R. A. Clark (NRC) to R. E. Uhrig (FPL), NRC Staff evaluation of CEN-122(F), FIESTA, dated April 8, 1981.

19. Amendment 45 to St. Lucie Unit 1 Operating License DPR-67, Shutdown Margin and Steam Generator Pressure-Low Trip Setpoints, dated November 3, 1981.

20. CENPD-139, "C-E Fuel Evaluation Model Topical Report," July 1, 1974.

21. CENPD-153, Revision 1, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System," May 1980.

22. CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," dated July 1975.

23. Letter from K. Kniel (NRC) to A. E.Scherer (CE), "With Evaluation of Topical Report CENPD-161-P," dated September 14, 1976.

24. CENPD-162-P-A (Proprietary) and CENPD-162-A (non-Proprietary), "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution," dated April 1975.

25. Supplement 3-P (Proprietary) to CENPD-225P, "Fuel and Poison Rod Bowing," dated June 1979.

26. CENPD-207-P, "C-E Critical Heat Flux; Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 2, Nonuniform Axial Power Distribution," dated June 1976.

27. "Interim Safety Evaluation Report on Effects on Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (Revision 1)," dated February 16, 1977.

28. CEN-123(F)-P, "C-E Response to NRC Second Round Questions on the Statistical Combination of Uncertainties Program."

29. CEN-124(B)-P, Combustion Engineering, "Statistical Combination of Uncertainties Methodology, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," dated January 1980.

30. CENPD-190-A, "CEA Ejection, C-E Method for Control Element Assembly Ejection Analysis," dated July 1976.

31. Amendment 43 to St. Lucie Unit 1 Operating License DPR-67, Asymmetric Steam Generator Transient Protective Trip Function, dated October 14, 1981.

32. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Boron Dilution Alarm, dated November 13, 1981.

33. Letter from R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), Seized Rotor and Loss of Non-Vital AC Power Analyses, dated November 18, 1981.

APPENDIX-A

Statistical Combination of Uncertainties (SCU) for St. Lucie Unit 1

The licensee has defined the input data required for a detailed thermal-hydraulic analysis by type: (1) system parameters which describe the physical system and are not monitored during reactor operation and (2) state parameters, which describe the operational state of the reactor and are monitored during operation. There is a degree of uncertainty in the value used for each of the input parameters used in the design safety analyses. This uncertainty has been handled in the past by assuming that each variable affecting DNB is at its extreme most adverse limit of its uncertainty range. The assumption that all factors are simultaneously at their most adverse values leads to conservative restrictions in reactor operation. The licensee has proposed in three parts of the CEN-123(F)-P (Refs. 1, 2, and 3) a new methodology to statistically combine uncertainties in the calculation of new limits for St. Lucie 1. These limits will ensure with at least 95 percent probability and 95 percent confidence level that neither DNB nor fuel centerline melt will occur. Part 1 describes the application of the SCU to the development of the local power density (LPD) and thermal margin/low pressure (TM/LP) limiting safety system settings (LSSSs). These are used in the analog reactor protection system to protect against fuel centerline melt and DNB, respectively. Part 2 uses SCU methods to develop a new DNB limit. Part 3 uses SCU methods to define limiting conditions for operations (LCOs).

A.1 PART ONE

Part 1 of the report (Ref. 1) defines the methods used to statistically combine uncertainties applicable to the LSSSs and evaluates the aggregate of these uncertainties as they determine the reactor protection against DNB and fuel centerline melt. The report further defines those uncertainties that have to be considered and evaluates their probability distributions.

A.1.1 Thermal-Hydraulic Summary and Evaluation of Part 1

The methods by which the licensee determines the setpoints in the St. Lucie 1 reactor protection system are given in CENPD-199-P (Ref. 4). The statistical combination of variables does not alter these methods. The same variables are considered and, once the uncertainties have been identified, statistically combined, and applied to the setpoint variables, the development of the setpoints proceeds as has been done in the past to develop the LSSSs.

Basically, ordered pairs of values of the peripheral shape axial index and the power to the specified fuel design limit are plotted. A lower bound is drawn under the "flyspeck" data such that all the core power distributions analyzed are accommodated. This in itself retains much of the conservatism of the past practices, since all of the data points lie above the lower bound and must lie well above. The lower bound is then reduced by uncertainties derived from the statistical combination and the generation of the trips proceeds much as has been the past practice.

The variables considered in the LSSS determination are listed in Table 3-1 of Part 1 of the report (Ref. 1) together with values of their uncertainties. There are errors in Table 3-1 of the report (Ref. 1). Corrected values have been supplied (Ref. 5). Corrected values provided in Reference 5 must continue to be used in future calculations for reloads.

The bases of the uncertainty values of Table 3-1 are given in Appendix A of Reference 1. More information (Ref. 5) has been provided in response to a request for more detailed justification. The source and magnitude of the uncertainty estimates were reviewed and found to be acceptable. The method of combining the various uncertainties on a single variable will produce valid estimates of the total. The calculations were spot-checked and found to be correct.

A.1.2 Statistical Summary and Evaluation of Part 1

Uncertainties associated with DNB and LPD limiting system safety settings are combined statistically. A stochastic simulation technique is used to estimate the probability distribution function (pdf) of DNB overpower (p/fdn) and power to fuel design limit on linear heat rate (P/fdℓ) for a specific axial power distribution. The simulations are carried out for a number of axial power distributions characterized by peaking factors and normalized axial shapes. For each axial shape, the pdf's of P/fdn and P/fdℓ are estimated. For each pdf the ratio of the mean value to the lower 95/95 probability/confidence limit is computed. The statistically combined uncertainty is taken as the maximum ratio over all axial shapes used.

Evaluation of the statistical validity of the uncertainty combination methodology requires examination of the following points:

1. Sampling Method
   - design of the simulation experiment
   - number of samples (simulation runs)
   - random number generator
2. Uncertainty distributions of independent variables
   - distribution form, e.g., Gaussian, uniform
   - statistical analysis method

These points will be discussed in order.

1. Sampling Method

For the TM/LP LSSS the input parameters subject to uncertainty are:

- primary coolant inlet temperature
- pressurizer pressure
- primary coolant flow
- ΔT/flux power
- radial peaking factor
- ASI correction terms.

The simulation is carried out by selecting a peripheral axial shape index and the corresponding axial shape. For the selected axial shape at least 500 simulation trials are carried out, with each trial using one sampled value from each input parameter distribution. The sampling is carried out using the SIGMA code and a Latin Hypercube Sampling (LHS) design. The LHS design with 500 trials will produce accepable estimates of the distribution of P/fdn.

The SIGMA is described in Section 4.4.1.1 of Reference 1 and CE's response to the first round questions (Ref. 6). The sample generation procedures depart somewhat from standard statistical practice. For example, the sample mean from a Gaussian distribution when the standard deviation is estimated from the same sample follows a student's t-distribution. SIGMA handles this by sampling a variance from a X^2 distribution and then sampling from a Gaussian distribution using the sampled variance. As a second example, SIGMA generates normal deviates using an approximation to the inverse Gaussian distribution function. Standard statistical methodology produces normal deviates by a transformation of uniform deviates. However, in the instances where SIGMA does not use standard techniques, the methods used will produce similar or more conservative results.

The random number generator used in the simulation trials was identified (Ref. 14) and test of autocorrelation, length of monotonic runs, and runs above and below mean were given. Since some random number generators can introduce inadvertent correlation, the use of a thoroughly tested generator is essential. The tests indicate that the generator is satisfactory. The method used to select axial power distributions is described in Berté, Filstein and Goldstein (Ref. 7). The method is divided into two parts. The first part is an algorithm for summarizing the distribution of axial shapes as a frequency distribution of hypercubes. The second part is a method of sample selection called Least Discrepancy Sampling (LDS), used to select a sample from the frequency distribution of hypercubes. The sampling procedure LDS does not preserve statistical properties of the sampled population and is, therefore, not acceptable. However, LDS was not used in selecting axial shapes. Instead, the sample was selected using simple random sampling or stratified sampling. Either of these methods is acceptable.

2. Uncertainty Distributions

For the most part, the methodology used to obtain uncertainty distributions on the independent parameters is acceptable. Distributions were not assumed to be Gaussian without being tested, and where data from several sources could not be pooled, conservative variance estimates were used.

A signal processing system is approximated by a first order Taylor series and the Central Limit Theorm (CTL) is applied to the approximation. The application of the CLT in Appendix A3 (Ref. 1) is justified by stating that the variances of the independent variables are small in relation to their overall ranges. However, the criterion that is necessary is that the variances be small relative to the size of the region of adequate approximation. Our review concluded that the necessary criterion is satisfied.

The error analysis performed on the shape annealing factor data has no statistical validity. Inspection of the data in Table 4 of Appendix A3 (Ref. 1) shows that the data from St. Lucie 1 is from a different population than the data from the other reactors in the table. Both the mean and the variance, after correction for cycle and channel effects, are larger for the St. Lucie 1 data. The incorrect error analysis attempted to account for the larger variance by using a multiplicative error structure. However, the standard deviation apparently increases faster than the mean, so the multiplicative structure does not remove the systematic component of the error.

Additional data on shape annealing factors for St. Lucie 1 was provided and analyzed in Reference 5 . The analysis concluded that the existing uncertainty estimate was conservative for St. Lucie 1.

This analysis of the St. Lucie 1 data has some statistical faults. However, these faults lead to an overestimate of the uncertainty so that the conclusion remains valid. Thus, the existing stochastic simulation of the axial shape index uncertainty is acceptable.

A.2 PART TWO

The licensee's approach for SCU is to adopt a single set of "most adverse state parameters" and generate a MDNBR response surface of the system parameters, which is, in turn, applied in Monte Carlo methods to combine numerically the system parameter probability distribution functions with the CHF correlation uncertainty. Our review of the SCU methodology includes the selection of the most adverse state parameters, the elimination of some system parameters from the response surface, the uncertainties of system parameters in the response surface and the statistical method used in calculating the final equivalent MDNBR limit.

(1) Most Adverse State Parameters

Generation of the actual response surface simultaneously relating MDNBR to both system and state variables would require an inordinate number of detailed TORC analyses. The licensee's solution to this problem is to select one single set of state parameters for use in developing the system variable response surface. The problem then becomes one of selecting a single set of state parameters, termed the most adverse state parameter set, that leads to conservatism in the system parameter response surface; i.e., the resultant MDNBR uncertainty is maximized. Calculations are performed with the detailed TORC code to determine the sensitivity of the system parameters at several set of operating conditions (state parameters). By tabulating the results of the sensitivity studies and through an examination of tables and exercise of engineering judgment, the "most adverse is listed in Section 3.1.5 of the CEN-123(F)-P report (Ref. 2).

Our review has found that the values of these parameters, such as system pressure, inlet coolant temperature and primary flow rate, are very likely at their most adverse values.

In Section 1.1, it is stated that the MDNBR is a smoothly varying function of the state parameters. This is not the case for the ASI. The ASI enters the calculation of MDNBR by the selection of a value of ASI from a finite collection of axial shapes and corresponding ASI's. Because the correspondence between ASI and axial shape is a multi-valued relationship, MDNBR cannot be a continuous function of ASI. Thus, a relatively small perturbation in ASI could lead to a large change in MDNBR. The data presented in CEN-123(F)-P indicate the possibility of an ASI that is considerably more adverse than the ASI selected as most adverse. In response (Ref. 8) to our question (Ref. 9) the licensee provided additional evaluations of the sensitivity of MDNBR near the most adverse ASI. With this additional information, the ASI selected as most adverse can be accepted as leading to conservative estimates of the sensitivity of MDNBR to system parameter variation. We, therefore, conclude that the licensee has achieved the goal of finding the most adverse set of state parameters.

(2) System Parameter Uncertainties

The CEN-123(F)-P report lists each of the system variables and then either provides the rationale for eliminating the variable from the statistical combination or provides the appropriate uncertainty value. Our review of thise variables follows:

(i) Radial Power Distribution

Conservatism in the thermal margin modeling is listed as a reason that uncertainty in the radial power distribution need not be considered. A subsequent response to questions (Ref. 8) outlined the proprietary calculational technique currently being used to maintain the conservatism. The technique was reviewed and found to be satisfactory. The elimination of the radial power distribution uncertainty is justified.

(ii) Inlet Flow Distribution

The sensitivity studies in CEN-123(F)-P (Ref. 2) have shown that MDNBR in the limiting hot assembly is unaffected by changes in the inlet flow of assemblies which are diagonally adjacent to the hot assembly. Therefore, only the inlet flow to the hot assembly and its contiguous neighbors are included in the analysis. We find this approach acceptable.

(iii) Exit Pressure Distribution

The sensitivity study provided in Table 3.10, CEN-123(F)-P (Ref.2) has shown the insensitivity of MDNBR with respect to the variation in exit pressure distribution. Therefore, we conclude the elimination of the exit pressure distribution uncertainty from the MDNBR response surface acceptable.

(iv) Enthalpy Rise Factor

Enthalpy rise factor is used to account for the effect on hot channel enthalpy rise of the fuel manufacturing deviation from nominal values of fuel dimension, density, enrichment, etc. The enthalpy rise factor is determined in accordance with an approved quality assurance procedure (Ref. 10). This involves a 100 percent recording of the relevant data which are then collected into a histogram. The mean and standard deviation are determined with 95 percent confidence. We find this procedure and the uncertainty listed in Table 5.1 (Ref. 2) acceptable.

(v) Heat Flux Factors

Manufacturing tolerance limits and fuel specifications which conservatively define the probability distribution function of the heat flux factor are used. We find the mean and the standard deviation of heat flux factor used in the analysis are conservative and, therefore, acceptable.

(vi) Clad O.D.

Proprietary measured clad diameter mean and standard deviations are given based on as-built data. The minimum systematic clad O.D. and its standard deviation are used in the development of the heat flux factor since this gives the most adverse effect on DNB. The minimum clad O.D. and its standard deviation are used in wetted perimeter calculations which penalizes the MDNBR. This double accounting of the clad O.D. uncertainty introduces conservatism in the analysis and is acceptable.

(vii) Systematic Pitch Reduction

As-built data are used to determine proprietary mean and standard deviations of gap width. The minimum mean and its standard deviation are chosen for combination with maximum clad O.D. to give the minimum pitch. The use of the minimum gap width is a conservative approach and is acceptable.

(viii) Fuel Rod Bow

The methodology for calculating rod bow compensation is discussed in Section C.2.2 of this SER. The rod bow compensation is applied directly as a multiplier to the MDNBR limit and the approach is acceptable.

(ix) CHF Correlation

The DNBR limit associated with the CE-1 correlation as discussed in Section C.2.1 is imposed to account for only the uncertainty of the correlation. Other uncertainties associated with plant system parameters and measurements of operating state parameters are accounted for, separately, through accompanying uncertainty factors.

In our review of the correlation prediction uncertainty, we also applied a cross-validation technique, where the test data are divided into two equal portions. The parameters of the correlation are estimated separately on each half. The estimated correlation from one half is then used to predict the data from the other half. Based on results of the cross validation technique, we conclude that the standard deviation of the measured to predicted CHF ratio should be increased by 5 percent. This increase in correlation uncertainty should be included in the derivation of the DNBR limit.

(x) Code Uncertainty

Uncertainty exists in all subchannel codes. Our evaluation result of the CE-1 DNBR limit using the COBRA IV code differs slightly from the applicant's analysis using the TORC code. This is, to a great extent, a result of the inherent calculational uncertainties in the two codes. The applicant contends that since the same TORC code is used for both CHF test data analysis and CHF calculations in the reactor, the code uncertainty is implicitly included in the minimum DNBR limit that is used for reactor application. However, we find the argument not valid since the CHF test section, being a small number of representative pins, differs from the reactor fuel assemblies in the large reactor core. Even though the heated shrouds are used in test assembly, the two-phase frictional pressure drop and diversion cross flow phenomena, etc., result in uncertainties in thermal hydraulic conditions predicted in the test assembly and reactor core. Information to quantify these uncertainties are not easily obtained and have not been provided. Therefore, consistent with past practice, we have imposed a 4 percent uncertainty for the subchannel codes and 1 percent uncertainty for transient codes which predict conservatively against data. These code uncertainties are imposed only when SCU is used for design analysis. The code uncertainties should be included in the SCU to assess the effect of the uncertainties on DNBR limit.

(3) Response Surface of System Parameters

The use of a response surface to represent a complicated, multivariate function is an established statistical method. A response surface relating MDNBR to system parameters is created. Conservatism is achieved by selecting the "most adverse set" of state parameters that maximizes the sensitivity of MDNBR to system parameter variations. The response surface includes linear, cross-product, and quadratic terms in the system parameters. Data to estimate the coefficients of the response surface are generated in an orthogonal central composite design using the TORC code with the CE-1 CHF correlation. The resulting MDNBR response surface is described in Table 4-2 of CEN-123(F)-P (Ref. 2).

The licensee has calculated the coefficient of determination associated with the response surface to be 0.9995 and the standard error of 0.003408. We conclude that the response surface prediction MDNBR is acceptable.

(4) Derivation of Equivalent MDNBR Limit

The probability distribution function (pdf) of MDNBR is estimated using the response surface in a Monte Carlo simulation. The simulation also accounts for uncertainty in the CHF correlation. The estimated MDNBR pdf is approximately normal, and a 95/95 probability/confidence limit is assigned using normal theory.

The SIGMA code is used in a simulation to estimate the distribution of MDNBR. SIGMA is reviewed in the statistical evaluation of Part 1 of CENPD-123(F)-P (Ref. 1). The results of the simulation were compared to results obtained using an analytical propagation of variance. The two methods are in close agreement. Therefore, we conclude the use of Monte Carlo simulation and SIGMA code acceptable.

In our review of the statistical methodology used in deriving the final equivalent MDNBR limit (Section 6.1, Reference 2), we discovered that an incorrect number of degrees of freedom is used in calculating the error associated with the response surface at 95 percent confidence level. However, since the error associated with the response surface is very small, the error results in minimal effect on DNBR limit.

The derivation of the SCU - equivalent MDNBR limit is generally acceptable except for the omissions of the CE-1 correlation cross-validation uncertainty and code uncertainty. As described in Item 2-ix, the standard devaition of the measured/predicted CHF ratio should be increased by 5 percent resulting from cross-validation of the test data. This increased uncertainty results in an increase of MDNBR by 0.005. Secondly as described in Item 2-x, a 5 percent code uncertainty should be included in the response surface. Assuming this uncertainty equal to two standard deviations, and combining the standard deviation with the standard deviation of the response surface by root sum square method, the MDNBR limit will increase by a factor of 1.008, i.e., an increase of 0.01 in MDNBR limit.. With the generic MDNBR limit of 1.19 for the CE-1 correlation, the SCU-equivalent MDNBR becomes 1.234. As was explained in Section 2.2, no rod bow DNBR compensation is required for Cycle 4, therefore, the licensee's proposed final MDNBR limit value of 1.23 is correct and is acceptable to the staff.

A.3 PART THREE

Part 3 of the report describes the method for statistically combining the uncertainties involved in the calculation of the limits for DNB, linear heat rate (LHR), and limiting condition for operation (LCO). The methods outlined parallel those given in Part 1 to develop the statistical combination method for LSSSs. For this reason the comments on the discussion for Part 1 of this review also apply to Part 3.

The differences between Part 1 and Part 3 of this report arise in the development of those distributions which impact LCO's differently than they impacted the LSSS's, in particular to determine whether statistically combining uncertainties affects the selection of initial conditions for the transient analyses. Also it is necessary to examine the sensitivity of the required over power margin (ROPM) to the initial condition to determine the magnitude of variations of ROPM within the range of the uncertainties.

A.3.1 Thermal-Hydraulic Evaluation, Part 3

The uncertainty distributions which are different for the LCO determinations described in Part 3 from the LSSS determinations described in Part 1 have to do with the ASI. Different ex-core neutron flux detectors are used to monitor the ASI for LCO determinations than are used for LSSS determinations. They are designed control channel instruments rather than the safety channel designation of the instruments used for LSSS evaluations. The control channel instruments are at different angular locations than are the safety channel instruments. Some of their specific uncertainty values are different. The techniques used to generate the safety channel uncertainties were also used for the control channels, and the results shown in Table A1-1 (Ref. 3) are satisfactory.

The licensee has determined that the reactor coolant system (RCS) depressurization event gives the maximum pressure bias term for the entire range of system parameters allowed by the Technical Specifications LCO. The methods and initial conditions used in this analysis are selected in the same manner as is currently done (Ref. 4). No changes in the determination of the TM/LP trip for protection against design basis events is required as a result of the change of combining uncertainties from deterministic to statistical.

The licensee has also determined that none of the design basis events has a margin degradation from time of trip signal to time of peak kW/ft greater than the bias already included in the LPD trip system. Therefore, the method of combining uncertainties, statistical or deterministic, has no impact on the initial conditions selected for analysis.

The four pump loss of flow event (LOF) and the control element assembly (CEA) drop events characterize those events for which RPS trips or sufficient initial steady-state margin is necessary. For both events, the maximum variation in the ROPM was determined. This margin variation is added to the cycle specified ROPM calculated for nominal conditions to establish the LCO.

The analysis of these events contains several conservative assumptions. For the four pump LOF event they are:

1. The magnetic flux decay in the holding coils was assumed to be 0.5 second. Field tests show a more realistic 0.4 second.

2. A low flow response time of 0.5 second was assumed. Field tests show that this is conservative by at least 0.1 second.

3. CEA drop time of 3.1 seconds was assumed. A more realistic value would be 2.9 seconds.

4. The flow coastdown did not take credit for the coastdown assist feature.

For the CEA drop event the conservative assumptions are:

1. A bounding value of the integrated radial peaking factor was assumed which was conservative by 2 percent. The analysis also assumed a minimum CEA drop worth which does not produce the maximum radial peaking factor change.

2. No credit was taken for the lowering of the margin requirement for increasing pressurizer pressure which would occur.

3. The moderator temperature coefficient assumed was the most negative allowed by Technical Specifications.

Best estimate calculations were made for both cases which showed that the conservatism is considerable.

There are errors in Table 3-1 of the report (Ref. 3). Corrected values have been supplied (Ref. 5). Subsequent reloads will require that the corrected values provided in Reference 5 be used in calculations. Based on our review, we find the licensee's method for statistically combining the uncertainties involved in the calculation of limits for DNB, LHR and LCO's acceptable.

References

1. CEN-123(F)-P, "Statistical Combination of Uncertainties, Part 1", dated December 1979.
2. CEN-123(F)-P, "Statistical Combination of Uncertainties, Part 2", dated January 1980.
3. CEN-123(F)-P, "Statistical Combination of Uncertainties, Part 3", dated February 1980.
4. CENPD-199-P, "C-E Setpoint Methodology", dated April 1976.
5. CEN-123(F)-P, "CE Response to NRC Second Round Questions on the SCU"
6. C-E Response to NRC First Round Questions on the Statistical Combination of Uncertainties, Part 2 of CENPD-124(B)-P.
7. F. J. Berte, F. L. Filstein and R. Goldstein, "Representative Sampling in Reactor Data Analysis", TIS-6532, Combustion Engineering.
8. Combustion Engineering, Response to Questions of PNL of March 1981.
9. Letter from G. M. Hesson to H. Balukjian dated March 27, 1981.
10. CENPD-210-A, Combustion Engineering, "Quality Assurance Program", Revision 3, dated November 1977.

APPENDIX-B

Topical Report Evaluation

CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected Systems
CEN-126(F)-P

Summary of Report

This report describes proposed new methods to be used for the analysis of the sequential Control Element Assembly Group Withdrawal (CEAW) event for Combustion Engineering cores with analog protected systems. These methods are intended to allow the reclassification of the CEAW event from the category requiring the Thermal Margin/Low Pressure (TM/LP) and the Axial Shape Index (ASI) trips to a category where sufficient initial steady state thermal margin is built into Departure from Nucleate Boiling (DNB) and Linear Heat Rate (LHR) Limiting Conditions for Operation (LCO's) to ensure that Specified Acceptable Fuel Design Limits (SAFDL's) are not exceeded. This reclassification relies on the High Power Trip (HPT) or the Variable High Power Trip (VHPT) to mitigate the consequences of this event instead of the TM/LP and ASI trips which are presently required.

A detailed analysis is presented in the report and is used to determine the initial conditions which cause the largest DNB and Centerline Temperature Melt (CTM) margin degradation during the CEAW transient when credit is taken only for the HPT or the VHPT. This analysis includes

sensitivity studies for the following key parameters:

(1) CEA withdrawal rate,

(2) gap thermal conductivity,

(3) initial power level,

(4) moderator temperature coefficient (MTC) of reactivity, and

(5) integrated radial peaking factor (maximum for given power level)

Best estimate calculations for DNB required overpower margin and Peak Linear Heat Generation Rate (PLHGR) are presented and compared with the safety analysis calculations to quantify the degree of conservatism.

Summary of Review

We have reviewed the material presented in the subject report with regard to the completeness with which it demonstrates that the CEAW event can be reclassified to a category where sufficient initial steady state margin is built into DNB and LHR LCO's to ensure that the HPT or the VHPT and not the TM/LP and ASI trips can mitigate the consequences of this event. The event is still classified as an Anticipated Operational Occurrence (AOO) and, therefore, the DNB and CTM SAFDL's must not be violated. We reviewed the analytical models employed, the input parameters and initial conditions assumed, the conservatism in the assumptions of the analysis, and the results of the analysis. In addition, we had our Technical Assistance consultants at Brookhaven National Laboratory perform an independent review of the material presented by CE.

The reactor system response to the CEAW event was simulated using the digital computer codes CESEC (Ref. 1), TORC (Ref. 2), and QUIX (Ref. 3). In addition, the shielding code SHADRAC (Ref. 4) and the one-dimensional transport code ANISN (Ref. 5) were used to determine excore detector response during a CEAW event. We find the use of these codes acceptable for the analysis presented in the topical report.

We have reviewed the initial power levels assumed in the analyses and concur that the complete spectrum from Hot Zero Power (HZP) to 102 percent of full power was investigated. In addition, we find the parametric analysis in gap thermal conductivity, CEA withdrawal rate, and moderator temperature coefficient, as a function of initial power level an acceptable method for determining the peak conditions for the transient.

The postulated initial reactor coolant flow, pressure, and inlet temperature are consistent with the CEA and power configuration and we agree that they cover the extremes of postulated conditions so as to produce the maximum margin degradation.

A beginning of cycle (BOC) Doppler coefficient is used. Considering the time in cycle and temperature conditions of the fuel, we concur that the BOC value in conjunction with a 15 percent reduction due to calculational uncertainties is conservative and acceptable.

We have reviewed the scram reactivity and delay times used and agree that they have been conservatively chosen to maximize the time required to reduce the increases in power, heat flux, and coolant temperature.

The integrated radial peaking factors used have been conservatively selected to be the maximum for a given power level based on the CEA insertions allowed by the Power Dependent Insertion Limit (PDIL) at that power level.

We have reviewed the determination of margin degradation which is based on calculating the Required Overpower Margin (ROPM) that must be provided from the time of CEAW event initiation to the time of minimum DNBR and maximum LHR and find the method acceptable. Included in this review was an evaluation of the key reactor state parameters used in the analysis and their range of values.

As an aid in evaluating the conservatisms in the DNB ROPM and the PLHGR calculations, best estimate calculations were performed and compared with the calculations used for the safety analysis. Based on our review of these comparisons, we find the above mentioned calculations suitably conservative and, therefore, acceptable.

The bases for acceptance of the results of a CEAW event is that the minimum transient DNBR not be less than 1.19 (based on the CE-1 correlation) and that the maximum fuel centerline temperature does not exceed the UO_2 melt temperature. The minimum DNBR acceptance criterion is met for all cases in the CEAW study. The fuel centerline melt SAFDL is not exceeded if the PLHGR does not exceed a steady state limit. A limit of 21 kw/ft is used in this study. For some of the CEAW events analyzed, the power rise causes the steady state limit of 21 kw/ft to be exceeded. In these cases, the total energy generated and the corresponding temperature rise at the hot spot are calculated to determine the maximum fuel centerline temperature reached during the transient. We concur that for rapid power spikes of short duration, a time at power is more significant than the PLHGR achieved. We have reviewed the procedures described in the report to calculate the fuel centerline temperatures and find them acceptable.

Evaluation Procedure

The review of the CEAW topical report has been conducted within the guidelines provided by the Standard Review Plan (NUREG 75/087). Sufficient information has been presented in the report and in responses to our questions to permit the conclusions described in the Regulatory Position.

Regulatory Position

Based on our review of the areas described above, we conclude that the subject report is an acceptable reference for the method of analyzing a CEAW event for St. Lucie Unit 1. We concur that the results presented support reclassification of the CEAW event from the category requiring the TM/LP and ASI trips to the category where sufficient initial thermal margin is built into the LCO's to ensure that DNB and LHR SAFDL's are not exceeded when only the HP or VHP trips are credited as possible trips to mitigate the event. This reclassification infers that the CEAW event is no longer the limiting event for the calculation of the pressure bias factor used in establishing the TM/LP setpoints although this bias term for the TM/LP trip is still required and determined for other transients as described in CENPD-199-P (Ref. 6).

References

1. CENPD-107, CESEC Topical Report, July 1974.

2. CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.

3. System 80 PSAR, CESSAR, Volume 1, Appendix 4A, Amendment no. 3, June 3, 1974.

4. G30-1365, "SHADRAC, Shield Heading and Dose Rate Attenuation Calculation," March 25, 1966.

5. K-1693, "A User's Manual for ANISN," March 30, 1967.

6. CENPD-199-P, "C-E Setpoint Methodology," April 1976.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the Technical Specifications for operation of the St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of the date of issuance.

The amendment changes License Condition 2.C.(1) and the Technical Specifications to authorize operation of St. Lucie Unit 1 at 2700 Megawatts thermal power. The previously authorized maximum power level was 2560 Megawatts thermal.

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 this action was published in the Federal Register on January 15, 1981 (46 FR 3686). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility dated June 1973.

For further details with respect to this action, see (1) the applications for amendment dated November 14, 1980 and September 28, 1981, (2) Amendment No. 48 to License No. DPR-67, and (3) the Commission's related Safety Evaluation and Environmental Impact Appraisal. 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 Licensing.

Dated at Bethesda, Maryland this 23rd day of November, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing