MARCH 2 19

Docket No. 50-348

Mr. Alan R. Barton
Senior Vice President
Alabama Power Company
P. O. Box 2641
Birmingham, Alabama 35291

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. NPF-2 for the Farley Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated November 15, 1978, supplemented by letters dated December 21, 1978 and January 12, 1979.

The amendment includes the following revisions to the Appendix A Technical Specifications:

1. Modifications to Specification 4.2.2.2.e.2 and Bases 3/4.2.2 and 3/4.2.3 relating to the radial peaking factor limits as a function of core height.

2. Changes to Specification Table 3.3-1, Reactor Trip System Interlocks, and associated Bases relating to the setpoint for a reactor trip following a turbine trip.

3. Deletion of references to part length control rods which will be removed from the reactor during the first refueling outage.

4. Addition of a Maintenance Superintendent as a member of the Plant Operations Review Committee in Specification 6.0, Administrative Controls.

Revisions 1 and 4 above are effective as of the date of issuance of this amendment. Revisions 2 and 3 above will be effective upon startup after fueling for Cycle 2.

CP 1

7903220292 GD

OFFICE						
SURNAME						
DATE						

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

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 10 to NPF-2
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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SURNAME	CParrish	EAReeves:esp	DBrinkman	ASchwencer	D Swanson	RAVollmer
DATE	2/16/79	2/16/79	2/16/79	3/2/79	2/26/79	3/2/79

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

March 2, 1979

Docket No. 50-348

Mr. Alan R. Barton
Senior Vice President
Alabama Power Company
P. O. Box 2641
Birmingham, Alabama 35291

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2. Changes to Specification Table 3.3-1, Reactor Trip System Interlocks, and associated Bases relating to the setpoint for a reactor trip following a turbine trip.

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Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 10 to NPF-2
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

cc: Ruble A. Thomas, Vice President
Southern Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, D.C. 20036

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Dothan, Alabama 36301

Chairman
Houston Co.Commission
Dothan, Alabama 36301

John Bingham, Esquire
Balch, Bingham, Baker, Hawthorne,
Williams & Ward
600 North 18th Street
Birmingham, Alabama 35202

Edward H. Keiler, Esquire
Keiler & Buckley
9047 Jefferson Highway
River Ridge, Louisiana 70123

State Department of Public Health
ATTN: State Health Officer
State Office Building
Montgomery, Alabama 36104

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U.S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, NE
Atlanta, Georgia 30308

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. NPF-2

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

   A. The application for amendment by Alabama Power Company (the licensee) dated November 15, 1978, supplemented by letters dated December 21, 1978 and January 12, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 10, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as indicated in the letter of transmittal for this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace or revise as indicated the following pages of the Appendix "A" Technical Specifications with the enclosed pages. Revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Corresponding overleaf pages are also provided to maintain document completeness.

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3/4 2-11
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CHANNEL FUNCTIONAL TEST

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

CORE ALTERATION

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

SHUTDOWN MARGIN

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

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

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

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

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

UNIDENTIFIED LEAKAGE

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

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

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

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

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

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

reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by $\geq 1.55 \times 10^6$ lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of any one pump breaker above P-8 or the opening of two or more pump breakers below P-8. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the contained borated water volume of each water source, and

   3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside ambient air temperature is < 35°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable or misaligned from its group step counter demand position by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A revaluation of each accident analysis of Table 3.1-1 is performed within 5 days, this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions,

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) Either the THERMAL POWER level is reduced to $\leq$ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

POSITION INDICATOR CHANNELS OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

c. The provision of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one rod position indicator channel (excluding demand position indication) shall be OPERABLE for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3,*# 4*# and 5*#

ACTION: With less than the above required position indicator channels OPERABLE, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator channels shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 24 hours.

*With the reactor trip system breakers in the closed position.

#See Special Test Exception 3.10.5

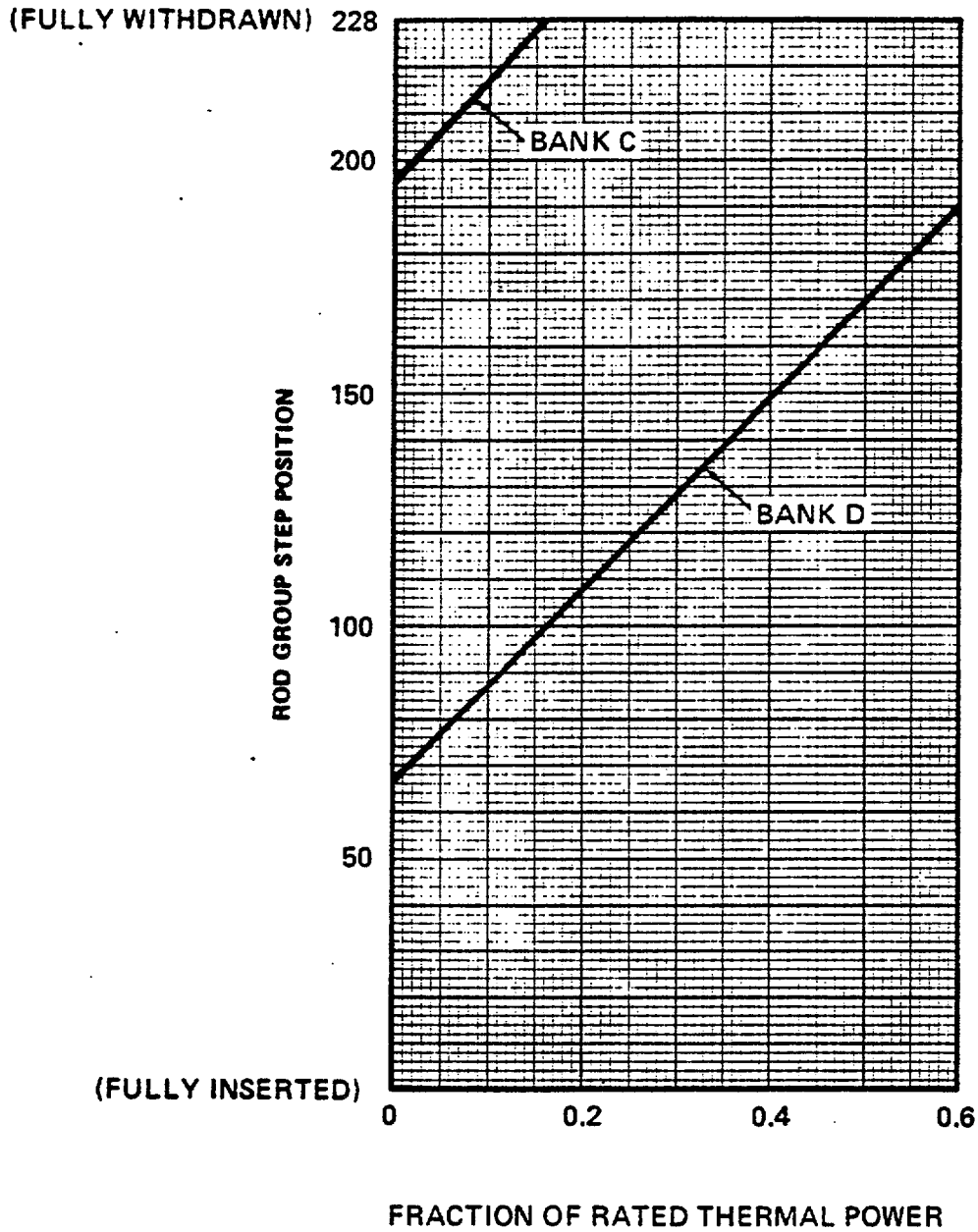


Figure 3.1-2 Rod Group Insertion Limits Versus Thermal Power Two Loop Operation

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods, and

2. For all unrodded core planes:
$F_{xy}^{RTP} \leq 1.68$ up to core elevations of 2.4 ft.

$F_{xy}^{RTP} \leq 1.75$ for core elevations from 2.4 ft. to 7.2 ft.

$F_{xy}^{RTP} \leq 1.61$ for core elevations above 7.2 ft.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive.

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

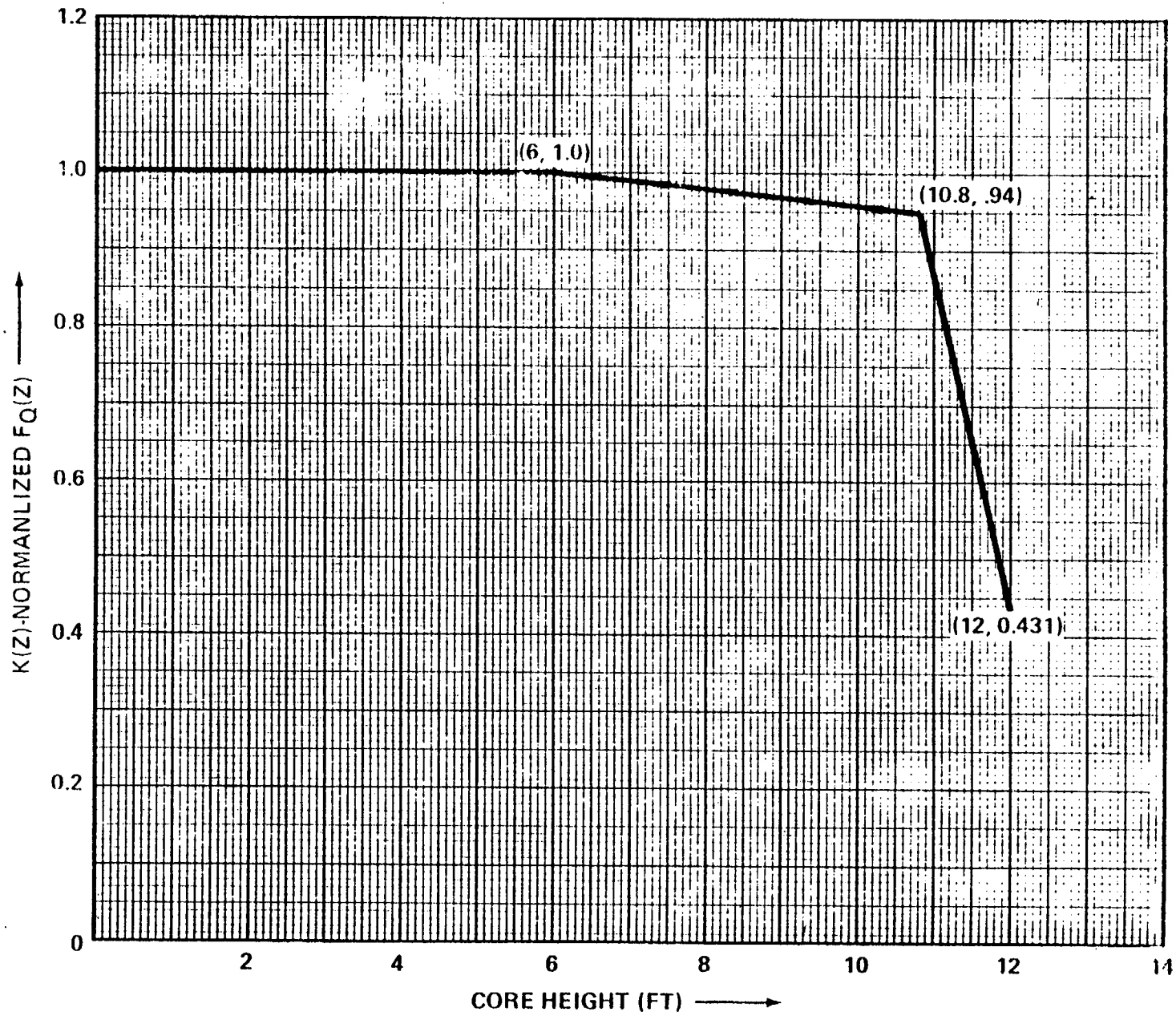


Figure 3.2-2 K(Z) Normalized $F_Q(Z)$ as a Function of Core Height

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown, control or part length rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable incore detectors to confirm that the power distribution is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels $< 6 \times 10^{-11}$ amps.	P-6 defeats the manual block of source range reactor trip
P-7	With 2 of 4 Power Range Neutron Flux Channels $\geq$ 11% of RATED THERMAL POWER or 1 of 2 Turbine impulse chamber pressure channels $\geq$ 55 psia.	P-7 defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, pressurizer low pressure, and pressurizer high level.

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-8	With 2 of 4 Power Range Neutron Flux channels ≥ 36% of RATED THERMAL POWER.	P-8 defeats the automatic block of reactor trip on low coolant flow in a single loop.
P-9	With 2 of 4 Power range Neutron Flux channels ≥ 51% of RATED THERMAL POWER.	P-9 defeats the automatic block of Reactor Trip on Turbine Trip.
P-10	With 3 of 4 Power range neutron flux channels < 8% of RATED THERMAL POWER.	P-10 prevents the manual block of: Power range low setpoint reactor trip, intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, initiate and continue boration at $\geq$ 30 gpm of between 7000 and 7700 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 30 gpm of between 7000 and 7700 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

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

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

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

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1 may be suspended during the performance of startup and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set $\leq$ 25% of RATED THERMAL POWER

APPLICABLITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing the manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ±(5)% target band about the target flux defference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulation during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. From THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours acutal time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process combuter though the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

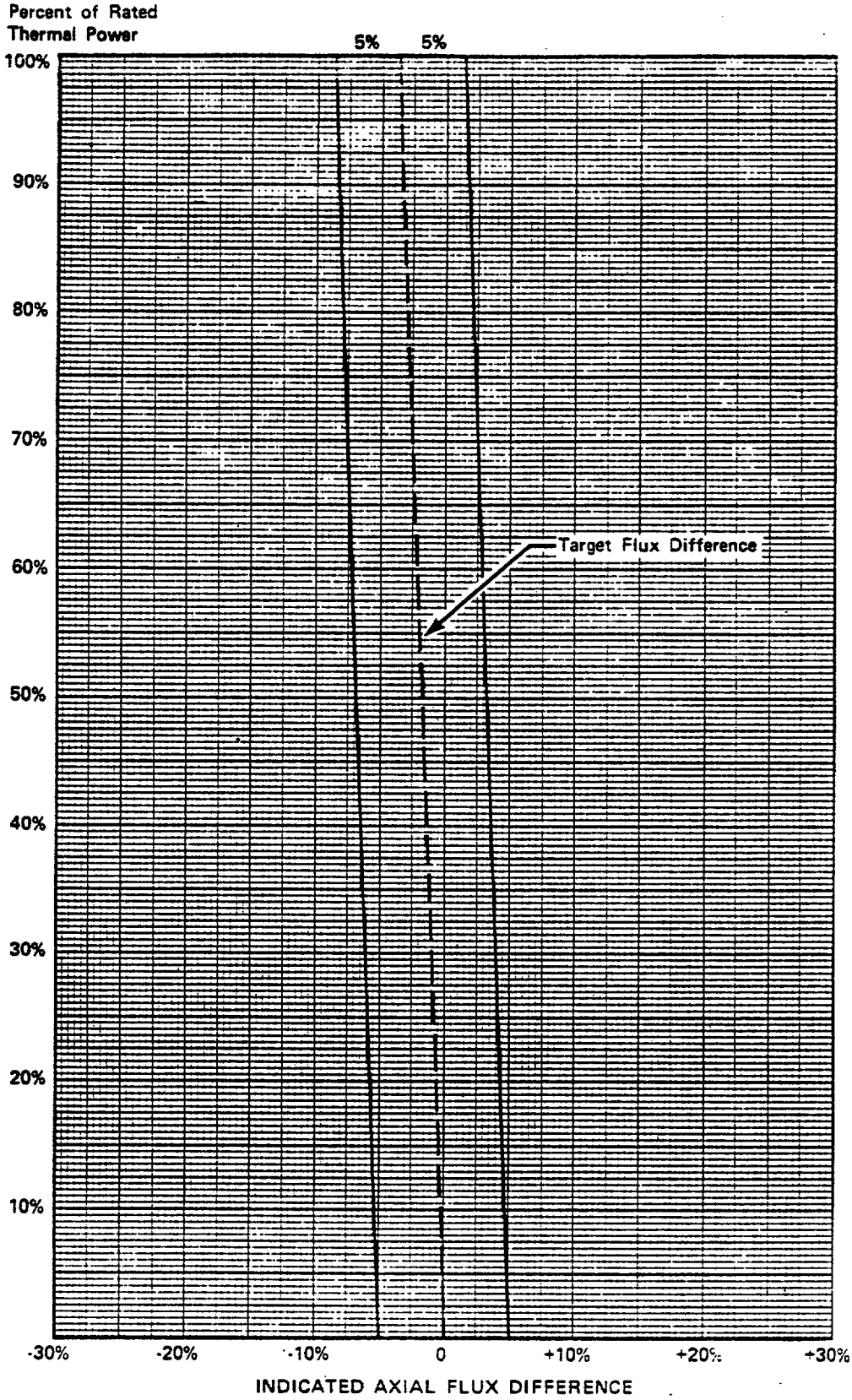


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL and RADIAL PEAKING FACTORS-

$F_Q(Z)$, $F^N_{\Delta H}$ and F_{xy} (z)

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measureable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the hot channel factor limits are maintained provided:

a. Control rods in a bank move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod banks are sequenced with overlapping banks as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

a. Abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. Although rod movement has a direct influence upon limiting F to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. Errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

The radial peaking factor, Fxy (z), is measured periodically to provide additional assurance that the hot channel factor, F_Q (z), remains within its limit. The Fxy (z) limits were determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The quardrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during start-up testing and periodically during power operation.

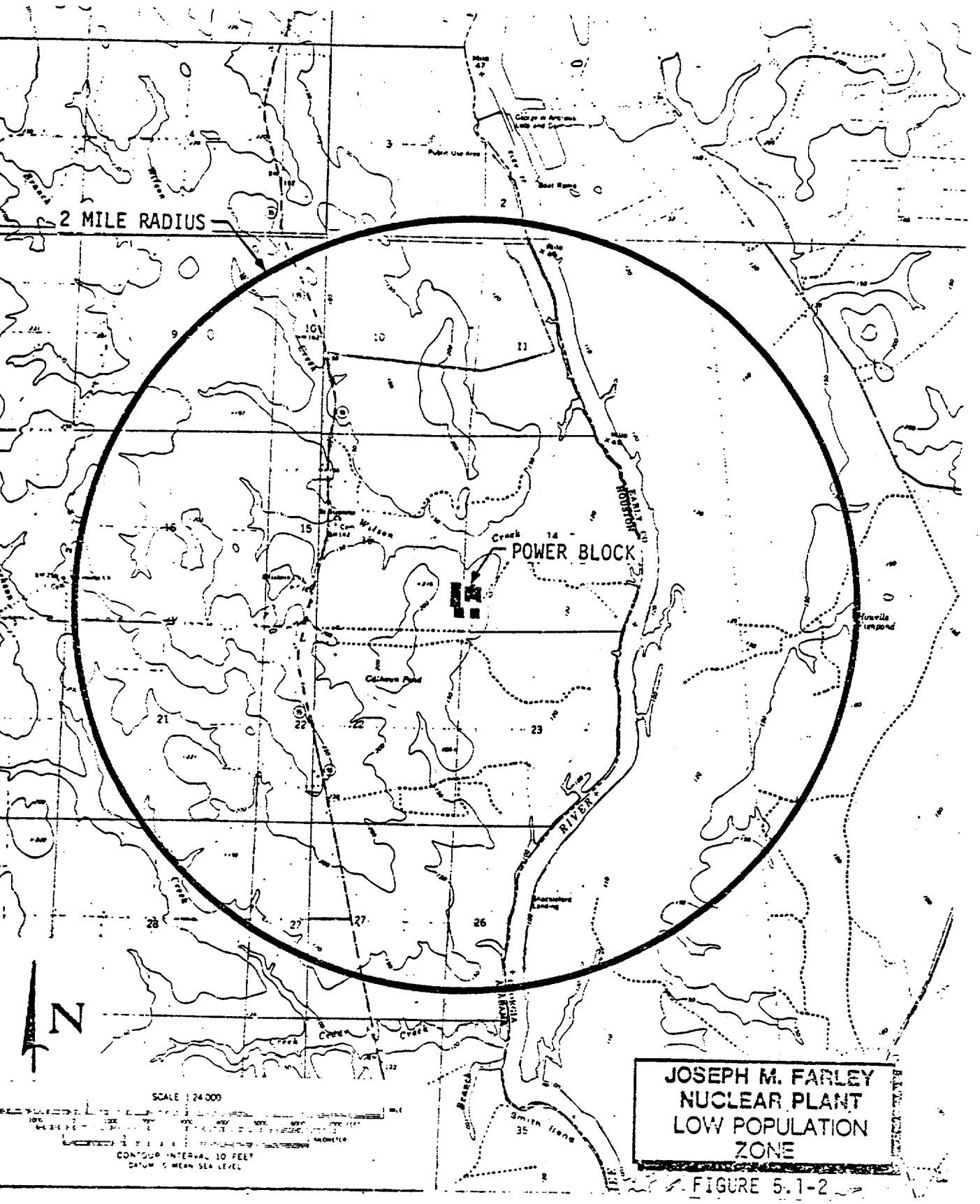
The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

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 initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters though 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.



JOSEPH M. FARLEY NUCLEAR PLANT LOW POPULATION ZONE

FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 54 psig and a temperature of 280°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

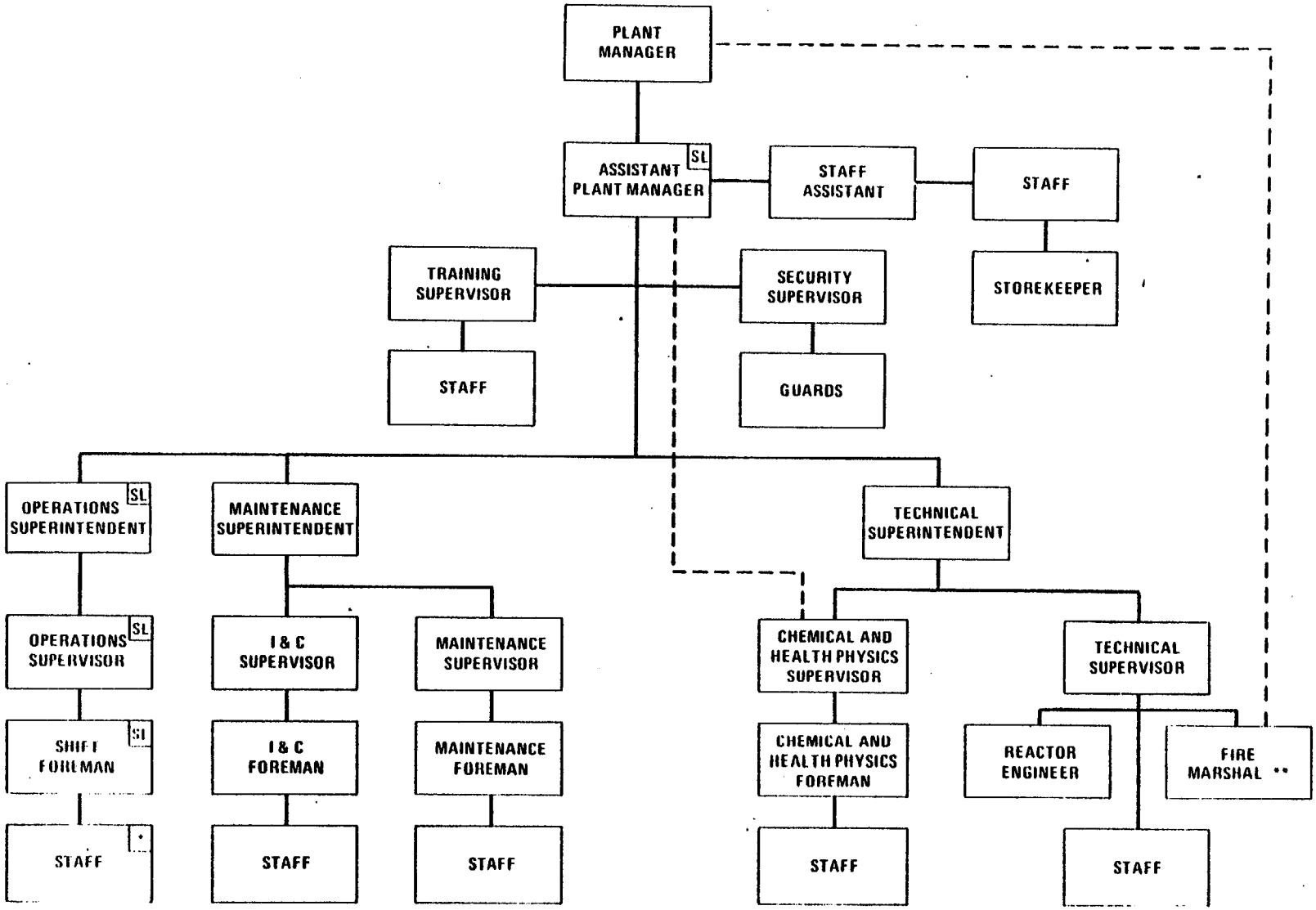
CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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



SL – Senior Operator's License
* – Reactor Operators will have Operator's License .
** – Onsite Responsibility for Fire Protection Program

Figure 6.2–2 Facility Organization – Joseph M. Farley - Unit No. 1

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

#Shift crew composition (including an individual qualified in radiation protection procedures) may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.*

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman:	Assistant Plant Manager
Member:	Technical Superintendent
Member:	Operations Suerintendent
Member: (Non-Voting)	Plant Quality Assurance Engineer
Member:	Maintenance Superintendent

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as voting members in PORC activities at any one time.

*The Minimum qualifications requirement for the Chemistry and Health Physics Supervisor shall become effective when the initial incumbent in this position is replaced.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or Vice Chairman.

QUORUM

6.5.1.5 A quorum of the PORC shall consist of the Chairman or Vice Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall review:

a. All administrative procedures and changes thereto,.

b. The safety evaluations for 1) procedures, 2) changes to procedures, equipment or systems and 3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question and defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or this Operating License.

f. Reports of violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures of instructions having nuclear safety significance or abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. All written reports concerning events requiring 24 hour notification to the Commission.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

a. Minutes of each NORB meeting shall be prepared, approved and forwarded to the Senior Vice President within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved by either the Technical Superintendent, the Operations Superintendent, the Maintenance Superintendent or the Assistant Plant Manager as applicable. The Plant Manager will approve administrative procedures, security implementing procedures and emergency plan implementing procedures. Temporary changes to procedures which clearly do not change the intent of the approved procedures will be approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved prior to implementation by the Plant Manager.

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment. Proposed test and experiments shall be approved before implementation by the Plant Manager.

d. Occurrences reportable pursuant to the Technical Specification 6.0 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Plant Manager and forwarded to the Manager of Nuclear Generation; and to the Chairman of the Nuclear Operations Review Board.

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c, and 6.5.3.1.d shall be members of the plant supervisory staff previously designated by the Plant Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review will include a determination of whether or not an unreviewed safety question is involved. Pursuant to 10 CFR 50.59 NRC approval of items involving unreviewed safety question will be obtained prior to Plant Manager approval for implementation.

RECORDS

6.5.3.2 Records of the above activities shall be provided to the Plant Manager, PORC and/or NORB as necessary for required reviews.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NORB and the Manager of Nuclear Generator.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NO. DPR-66

ALABAMA POWER COMPANY

FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

INTRODUCTION

By letter dated November 15, 1978, supplemented by letters dated December 21, 1978 and January 12, 1979, Alabama Power Company (APC) proposed changes to the Appendix A Technical Specifications for Farley Nuclear Plant (FNP), Unit No. 1. The proposals included the following:

1. Modifications to Specification 4.2.2.2.e.2 and Bases 3/4.2.2 and 3/4.2.3 relating to the radial peaking factor limits as a function of core height.

2. Changes to Specification Table 3.3-1, Reactor Trip System Interlocks, and associated Bases relating to the setpoint for a reactor trip following a turbine trip.

3. Deletion of references to part length control rods which will be removed from the reactor during the first refueling outage.

4. Addition of a Maintenance Superintendent as a member of the Plant Operations Review Committee in Specification 6.0, Administrative Controls.

DISCUSSION AND EVALUATION

1. Radial Peaking Factor (Specification 4.2.2.2.e.2 and Bases 3/4.2.2 and 3.4.2.3)

   By letter dated November 15, 1978, APC proposed a Technical Specification change to increase the radial peaking factor for RATED THERMAL POWER, F_{xy}^{RTP}, limits for the remainder of Cycle 1 and subsequent cycles of operation. Specifically, APC proposed changing Section 4.2.2.2.e.2 to increase F_{xy}^{RTP} from ≤ 1.55 for unrodded core planes to the following:

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$F_{xy}^{RTP} \leq 1.68$ up to core elevations of 2.4 feet

$F_{xy}^{RTP} \leq 1.75$ for core elevation from 2.4 feet to 7.2 feet

$F_{xy}^{RTP} \leq 1.61$ for core elevation above 7.2 feet

Thus, the radial peaking factor for unrodded core planes would become core elevation dependent. Such changes have been previously considered by the NRC staff and found acceptable.[1/]

F_{xy} is used during periodic surveillance calculations to show that the heat flux hot channel factor, F_Q, is within the current limit of 2.32. Calculations performed by Westinghouse Electric Corporation for the Farley Unit No. 1 cores were provided by APC letters of December 21, 1978 and January 12, 1979. Our review shows that the Farley cores would now and in the future exhibit flatter than the initial axial shape, with a corresponding lower axial peaking, F_z. Thus, the proposed increase in radial peaking factor, F_{xy}, would be offset by the decreased axial peaking F_z and would result in the predicted peak to core average linear heat rate ratio, F_Q, remaining within current Technical Specification limits. The F_Q limits would not be altered and the F_QXP[2/] limit envelope would remain unchanged with a maximum value of 2.32.

Similar changes in the calculational methods for F_Q have been previously supported generically[3/] for Westinghouse plants where eighteen load follow power histories were analyzed to ensure that the F_QXP limit was not exceeded. Westinghouse has described a subset of three of these eighteen cases.[4/] To account for the increased uncertainty inherent in analyzing a smaller number of cases, an uncertainty multiplier, denoted F_Q^{US} was applied. The NRC staff is currently reviewing this new method. No problems have been identified which would prohibit its use for Farley. In fact, as noted above (reference to the Trojan Plant Amendment No. 30) the NRC staff has approved the use of a subset of three cases rather than the eighteen cases.

[1/] Letter from A. Schwencer, USNRC, to C. Goodwin, Jr., Portland General Electric Company, June 22, 1978 transmitting Amendment No. 30 for Trojan Nuclear Plant.

[2/] F_QXP is the product of the total peaking factor and the fractional core power.

[3/] Letter from C. Eicheldinger, Westinghouse Electric Corporation to D. B. Vassallo, USNRC, July 16, 1975.

[4/] Letter from C. Eicheldinger, Westinghouse Electric Corporation to John F. Stolz, USNRC, April 6, 1978.

By letters dated December 21, 1978 and January 12, 1979, APC has shown, and we agree, that the three subset method used to calculate bounding values of F_QXP predict that, even with the proposed increase in F_{xy}, the F_Q limit of 2.32 will not be exceeded for the remainder of Cycle 1 and all of Cycle 2. There is no change in the core local peak power limit because F_QXP has not increased. Thus, the present total peaking F_Q limits and peak local power limits are not increased. The change does not change the margin to safety (local core power limit). Therefore, the proposed change in F_{xy} is acceptable for operation through Cycle 2. The F_Q limit envelope should be verified by the licensee for operation beyond Cycle 2.

2. Setpoint for Reactor Trip Following a Turbine Trip

By letter dated November 15, 1978, APC proposed to change the setpoint for the reactor trip following a turbine trip. The proposal would modify the turbine trip function shown in Specification Table 3.3-1, Reactor Trip System Interlocks. APC proposed to delete the turbine trip from the permissive interlock P-7 (neutron flux $\geq$ 11%) and to add a new permissive interlock, P-9 (neutron flux $\geq$ 51%). The P-9 interlock would allow a reactor trip following a turbine trip when reactor power is at or above 50% power. APC states that this change would significantly reduce the down time without compromising adequate safety margins.

APC has performed an analysis confirming that deleting the reactor trip following a turbine trip from 50% power or less has no adverse affect on plant safety. We have reviewed the APC submittal as well as the Farley Plant Final Safety Analysis Report with specific attention to Chapter 15, Accident Analyses. The previously reviewed and NRC staff approved analysis (Section 15.2.7) for a Loss of External Electrical Load and/or Turbine Trip takes no credit for the direct reactor trip following a turbine trip from 102% power. Therefore, changing the interlock setpoint from 11% to 51% power would fall well within the bounds of an accident previously analyzed and found acceptable.

We note that the proposed Reactor Protection System design changes for both the P-7 and P-9 interlocks should be made using the same design techniques as for the existing protection logic relays. The APC staff has agreed that such was its intention. APC has proposed accomplishing the modifications during the upcoming refueling outage. Thus, this part of the Technical Specification revision would become effective prior to startup for Cycle 2 operation.

Based on our review of APC's submittal and the discussion noted above, the proposed design changes and Technical Specification changes are acceptable.

3. References to Part Length Rod Cluster Control Assemblies (PLRCCA's)

By letter dated November 15, 1978 APC proposed to remove the PLRCCA's during the first refueling outage. Use of part length rods is currently prohibited during power operation. APC states that removal of these rods will reduce outage time and radiation exposure associated with surveillance testing.

Part length control rods were initially installed to suppress xenon induced power oscillations in the axial direction, should such oscillations occur. They were also intended for use in axial offset calibration tests or low power physics tests.

The Technical Specifications, as now written, require that these PLRCCA's be withdrawn and excluded from the core at all times during reactor operations. The PLRCCA's are not needed, used or assumed to be available to achieve required reactor shutdown conditions. The proposed removal, therefore, will not cause any change in required reactivity characteristics, or safety margins at full power, low power or shutdown. To the contrary, removal will eliminate the potential for part length rods remaining in the core during operation. Such an event would cause an abnormal core flux distribution.

In addition, in order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCA's were to be performed, APC proposes to install thimble plug assemblies in the spaces previously occupied by PLRCCA's. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug

is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

All components in the thimble plug assembly, except for the springs, are constructed from type 304 stainless steel. The springs are wound from Inconel x-750 for corrosion resistance and high strength.

These thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCA's have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Based on the considerations that (1) the PLRCCA's are not needed for reactor operation, (2) that removal of these assemblies will remove the chance for an abnormal flux distribution reactor shutdown, and (3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, this change and the associated Technical Specification changes are acceptable. Technical Specification changes become effective prior to startup for Cycle 2 operation.

4. Addition of Maintenance Superintendent to the Plant Operations Review Committee (PORC)

By letter dated November 15, 1978 APC proposed Administrative Control Changes to Technical Specifications 6.5.1.2, 6.5.3.1 and Figure 6.2-2. The changes would reflect recent changes to the facility organization. The Maintenance Superintendent has assumed responsibility for Maintenance and Instrumentation and Control activities. The Operations Superintendent's duties become strictly operations oriented.

The previously approved organization included one Operations Superintendent having line responsibility for three sections (Operations, I&C and Maintenance). The proposed organization places all maintenance activities under a new Maintenance Superintendent. APC states that the new organization increases the effectiveness of the review and approval process for procedures relating to nuclear safety. The change also would provide added expertise to the PORC. The Maintenance Superintendent would become a member of PORC. The change in Operations Superintendent's duties would allow more personalized operations control at the same time.

Since the changes proposed are improvements to the previously reviewed and approved organization, the Technical Specification changes are acceptable.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: March 2, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-348

ALABAMA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 10 to Facility Operating License No. NPF-2, issued to Alabama Power Company (the licensee), which revised Technical Specifications for operation of the Joseph M. Farley Nuclear Plant, Unit No. 1 (the facility) located in Houston County, Alabama.

The amendment revises the Technical Specifications relating to: (1) radial peaking factor limits as a function of core height; (2) setpoint for a reactor trip following a turbine trip; (3) part length control rods which will be removed from the reactor during the first refueling outage; and (4) addition of a Maintenance Superintendent. Revisions 1 and 4 above are effective as of the date of issuance of this amendment. Revisions 2 and 3 above will be effective upon startup after fueling for Cycle 2.

The application for the amendment complies 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated November 15, 1978, supplemented by letters dated December 21, 1978 and January 12, 1979, (2) Amendment No. 10 to License No. NPF-2, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the George S. Houston Memorial Library, 212 W. Vurdeshaw Street, Dothan, Alabama 36301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 2nd day of March, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors