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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. NPF-2

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

   A. The application for amendment by the licensee, dated July 15, 1991, as supplemented September 10, 1991, and January 10, 1992, 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 an safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. 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. 92 , are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Lloyd T. Walter

for Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 11, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 92

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
2-2	2-2
2-5	2-5
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-6	B 2-6
3/4 1-4	3/4 1-4
3/4 1-19	3/4 1-19
3/4 2-4	3/4 2-4
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-14	3/4 2-14
3/4 2-15	3/4 2-15
3/4 3-10	3/4 3-10
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-30	3/4 3-30
3/4 4-2	3/4 4-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 4-1	B 3/4 4-1
6-19	6-19

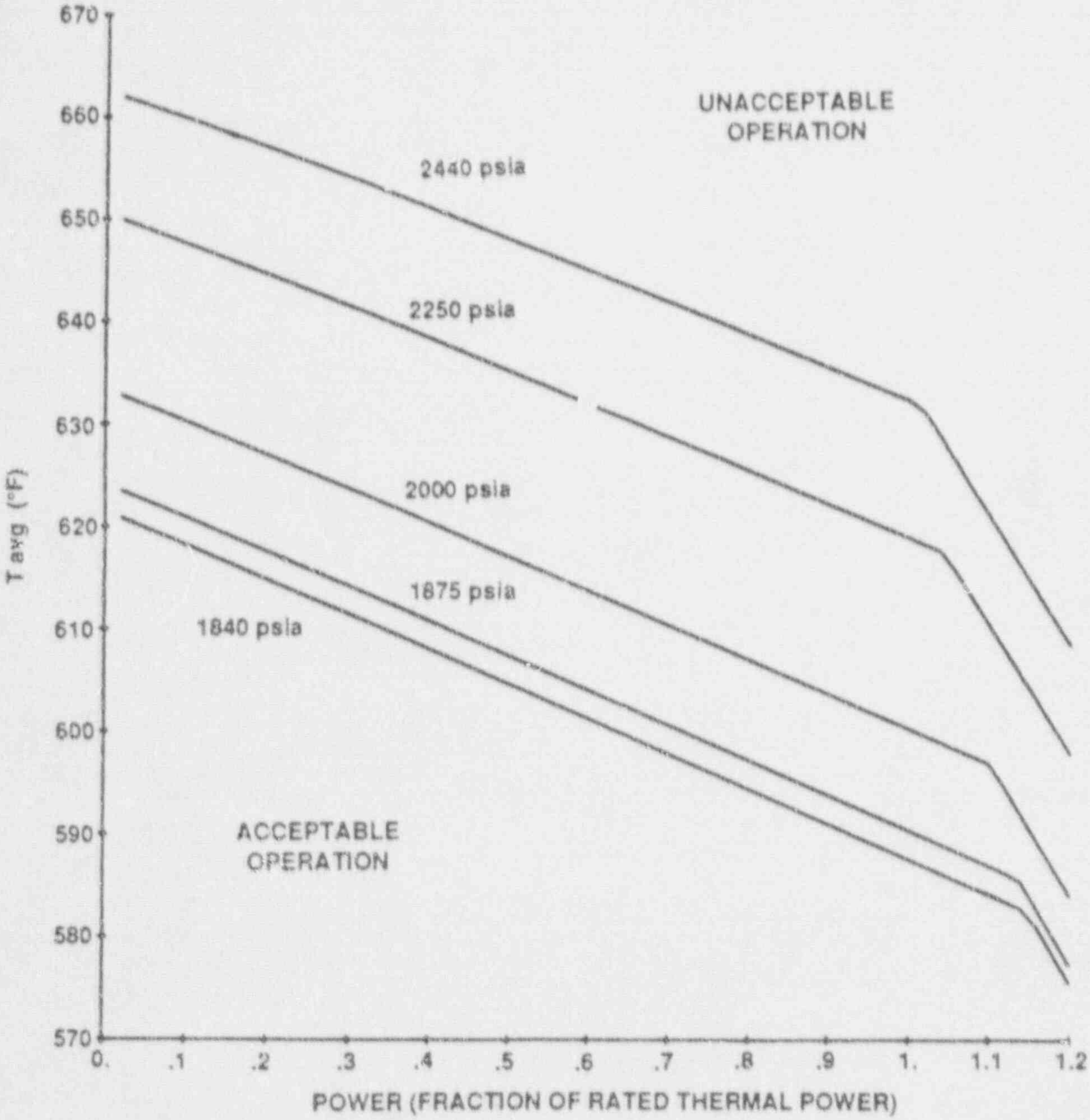

Figure 2.1-1 Reactor Core Safety Limits
Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER
	High Setpoint - $\leq$ 109% of RATED THERMAL POWER	High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 6
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of minimum measured flow per loop*	$\geq$ 88.5% of minimum measured flow per loop*

*Minimum measured flow is 89,290 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature ΔT

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_o \left[K_1 - K_2 \frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} \left(T \left(\frac{1}{1 + \tau_6 s}\right) - T'\right) + K_3 (P - P') - f_1 (\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

$T' \leq 577.2$°F (Maximum Reference T_{avg} at RATED THERMAL POWER);

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

$\frac{1 + \tau_1 s}{1 + \tau_2 s}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 30 sec, τ_2 = 4 sec;

$\frac{1 + \tau_4 s}{1 + \tau_5 s}$ = The function generated by the lead-lag controller for ΔT dynamic compensation;

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 = \tau_5$ = 0 sec;

$\frac{1}{1 + \tau_6 s}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 sec;

s = Laplace transform operator, sec^{-1};

Operation with 3 loops

K_1 = 1.14;

K_2 = 0.0250;

K_3 = 0.001275;

Operation with 2 loops

K_1 = (values blank pending

K_2 = NRC approval of

K_3 = 2 loop operation)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -39 percent and +13 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -39 percent, the ΔT trip setpoint shall be automatically reduced by 1.92 percent of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +13 percent, the ΔT trip setpoint shall be automatically reduced by 2.17 percent of its value at RATED THERMAL POWER.

Note 2: Overpower ΔT

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s}\right)\left(\frac{1}{1 + \tau_6 s}\right) T - K_6 \left(T \left(\frac{1}{1 + \tau_6 s}\right) - T''\right) - f_2(\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T'' = Reference T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 577.2°F);

K_4 = 1.07;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

K_6 = 0.00165/°F for $T > T''$, $K_6 = 0$ for $T \leq T''$;

$\frac{\tau_3 s}{1+\tau_3 s}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

τ_3 = Time constant utilized in the rate lag controller for T_{avg}, τ_3 = 10 sec;

$\frac{1 + \tau_4 s}{1 + \tau_5 s}$ = The function generated by the lead-lag controller for ΔT dynamic compensation;

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 = \tau_5 = 0$ sec;

$\frac{1}{1 + \tau_6 s}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ sec;

s = Laplace transform operator, sec^{-1};

$f2(\Delta I) = 0$ for all ΔI.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.8 percent.

Note 4: Pressure value to be determined during initial startup testing. Pressure value of $\leq$ 55 psia to be used prior to determination of revised value.

Note 5: Pressure value to be determined during initial startup testing.

Note 6: The channel's maximum trip point shall not exceed its computed trip point by more than 2.3 percent.

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. 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 correlations which have 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 DNB thermal design criterion is that the probability of DNB not occurring on the most limiting rod is at least 95 percent (at a 95 percent confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figures 2.1-1 and 2.1-2 show the reactor core safety limits for a range of THERMAL POWER, Reactor Coolant System pressure and average temperature which satisfy the following criteria:

a. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

b. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). Each curve reflects the most limiting result using either low-parasitic (LOPAR) fuel or VANTAGE 5 fuel. The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.25 and 1.24 for the typical and thimble cells, respectively.

c. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

The curves of Figures 2.1-1 and 2.1-2 are based on the most limiting result using an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.65 for VANTAGE 5 fuel and an $F^N_{\Delta H}$ of 1.55 for LOPAR fuel and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$F^N_{\Delta H}$ = 1.65 [1 + 0.3 (1-P)] for VANTAGE 5 fuel and

$F^N_{\Delta H}$ = 1.55 [1 + 0.3 (1-P)] for LOPAR fuel

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

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, pressurizer and the reactor coolant system piping and fittings are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. 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 analysis.

Manual Reactor Trip

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

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 8 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the DNB design criterion is met for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for multiple dropped rods. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

latter trip will ensure that the DNB design criterion is met during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature delta T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 66% RATED THERMAL POWER will ensure that the DNB design criterion is met during normal operational transients and anticipated transients with 2 loops in operation.

Steam Generator Water Level

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

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall 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 greater than or equal to 1.55×10^6 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 under-frequency to more than one reactor coolant pump. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less than or equal to 0.7×10^{-4} delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), condition for power levels up to 70% THERMAL POWER with a linear ramp to 0 delta k/k/°F at 100% THERMAL POWER.

b. Less negative than -4.3×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3.a - MODES 1 and 2* only#
Specification 3.1.1.3.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.3.a above operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.3.b above, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.0

See Special Test Exception 3.10.3

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (225 to 231 steps, inclusive)* shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

*The fully withdrawn position used for determining rod drop time shall be greater than or equal to the fully withdrawn position used during subsequent plant operation.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq [\frac{2.45}{P}]\ [K(Z)]$ for $P > 0.5$ for VANTAGE 5 fuel |

$F_Q(Z) \leq [4.9]\ [K(Z)]$ for $P \leq 0.5$ for VANTAGE 5 fuel and |

$F_Q(Z) \leq [\frac{2.32}{P}]\ [K(Z)]$ for $P > 0.5$ for LOPAR fuel |

$F_Q(Z) \leq [4.64]\ [K(Z)]$ for $P \leq 0.5$ for LOPAR fuel |

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

and K(Z) is the function obtained from Figure (3.2-2) for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

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

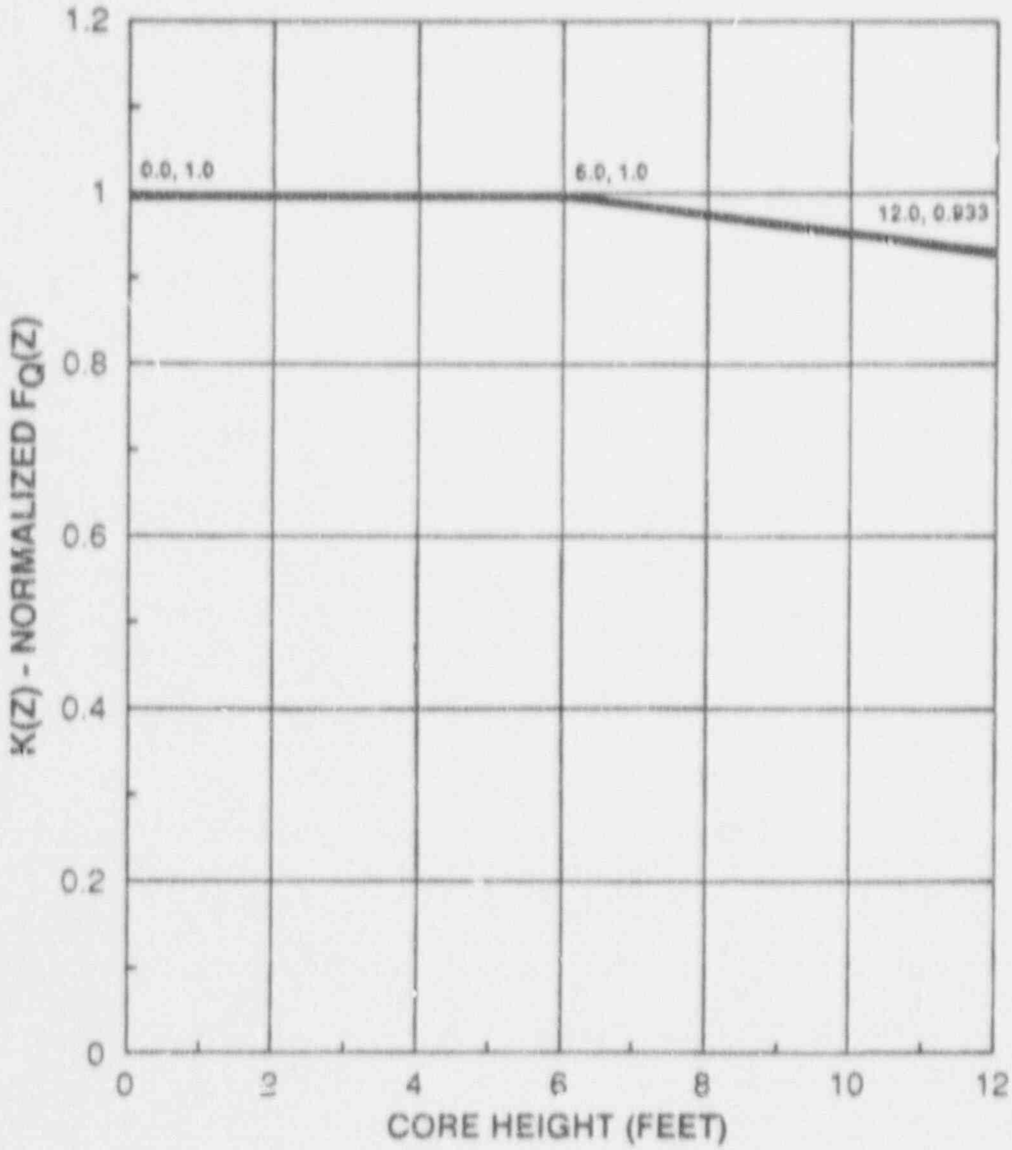

Figure 3.2-2 K(Z) Normalized $F_Q(Z)$ as a Function of Core Height

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$F^N_{\Delta H} \leq 1.65\ [1 + 0.3\ (1-P)]$ for VANTAGE 5 fuel and

$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$ for LOPAR fuel

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

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. 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 $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate.

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 less than 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 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.

4.2.5.3 The indicated RCS flow rate shall be verified to be within the acceptable limit at least once per 31 days.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops in Operation	LIMITS: 2 Loops in Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 580.7°F	(**)
Indicated Pressurizer Pressure	$\geq$ 2205 psig*	(**)
Indicated Reactor Coolant System Total Flow Rate	$\geq$ 267,880 gpm***	(**)

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

** Values blank pending NRC approval of 2 loop operation.

*** Value includes a 2.4% flow uncertainty (0.1% feedwater venturi fouling bias included).

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	
	a. High	≤ 0.5 seconds*
	b. Low	Not Applicable
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Power Range, Neutron Flux, High Negative Rate	Not Applicable
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	Not Applicable
7.	Overtemperature ΔT	≤ 6.0 seconds*
8.	Overpower ΔT	Not Applicable
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High-High	$\leq$ 27 psig	$\leq$ 28.3 psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation Logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High-High	$\leq$ 27 psig	$\leq$ 28.3 psig
c. Purge and Exhaust Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	≤ 16.2 psig	≤ 17.5 psig
d. Steam Flow in Two Steam Lines--High, Coincident with T_{avg}--Low-Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load with $T_{avg} \geq 543°F$	≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load with $T_{avg} \geq 540°F$
e. Steam Line Pressure--Low	≥ 585 psig	≥ 575 psig
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water Level--High-High	≤ 75% of narrow range instrument span each steam generator	≤ 76% of narrow range instrument span each steam generator

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0[1]/12.0[4]
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0[6]
d. Containment Isolation-Phase "A"	$\leq$ 17.0[4]
e. Containment Purge Isolation	$\leq$ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0[4]/87.0[1]
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 12.0[4]/22.0[5]
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0[6]
d. Containment Isolation-Phase "A"	$\leq$ 17.0[4]/27.0[5]
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0[4]/87.0[5]
5. Steam Flow in Two Steam Lines-High Coincident with T_{avg}--Low-Low	
a. Steam Line Isolation	Not Applicable
6. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 12.0[4]/22.0[5]
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0[6]
d. Containment Isolation-Phase "A"	$\leq$ 17.0[4]/27.0[5]
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0[4]/87.0[5]
h. Steam Line Isolation	$\leq$ 7.0

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant Loops listed below shall be OPERABLE and in operation when the rod control system is operational or at least two Reactor Coolant Loops listed below shall be OPERABLE with one Reactor Coolant Loop in operation when the rod control system is disabled by opening the Reactor Trip Breakers or shutting down the rod drive motor/generator sets:*

1. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant pump,
2. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant pump.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required Reactor Coolant Loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one Reactor Coolant Loop in operation and the rod control system operational, within 1 hour open the Reactor Trip Breakers or shut down the rod drive motor/generator sets.

c. With no Reactor Coolant Loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required Reactor Coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required Reactor Coolant Loop(s) shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication at least once per 12 hours.

* All Reactor Coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) meeting the DNB design criterion during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and 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 for manufacturing tolerances on fuel pellets and rods and measurement uncertainty.

$F^N_{\Delta H}$ 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

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.45 for VANTAGE 5 and 2.32 for LOPAR 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.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the +(5)% target band about the target flux difference, 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 cumulative 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. For 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 actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through 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 2 or more 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.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, NUCLEAR ENTHALPY HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that 1) the design limit on peak local power density is not exceeded, 2) the DNB design criterion is met, and 3) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

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

b. Control rod banks are sequenced with overlapping groups as described in Specification 3.1.3.6.

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.

BASES

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

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate 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 $F^N_{\Delta H}$ contains an 8% allowance for uncertainties. The 8% allowance is based on the following considerations:

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

b. Although rod movement has a direct influence upon limiting F_Q 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 distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

The radial peaking factor $F_{xy}(Z)$, is measured periodically to provide additional assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor limit report per Specification 6.9.1.11 was 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 quadrant 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 startup 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.

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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

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 meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 580.7°F is based on the average of two control board readings and an indication uncertainty of 2.5°F. The indicated pressure value of 2205 psig is based on the average of two control board readings and an indication uncertainty of 20 psi. The indicated total RCS flow rate is based on one elbow tap measurement from each loop and an uncertainty of 2.4% flow (0.1% flow is included for feedwater venturi fouling).

The 12 hour surveillance of Tavg and pressurizer pressure through the control board readings are sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The 18 month surveillance of the total RCS flow rate is a precision measurement that verifies the RCS flow requirement at the beginning of each fuel cycle and ensures correlation of the flow indication channels with the measured loop flows. The monthly surveillance of the total RCS flow rate is a reverification of the RCS flow requirement using loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of the fuel cycle. The 12 hour RCS flow surveillance is a qualitative verification of significant flow degradation using the control board indicators and the loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of each fuel cycle.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all Reactor Coolant Loops in operation, and meet the DNB design criterion during all normal operations and anticipated transients. In MODES 1 and 2 with one Reactor Coolant Loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two Reactor Coolant Loops provide sufficient heat removal capability for removing core heat even in the event of a bank withdrawal accident; however, a single Reactor Coolant Loop provides sufficient decay heat removal capacity if a bank withdrawal accident can be prevented; i.e., by opening the Reactor Trip Breakers or shutting down the rod drive motor/generator sets.

In MODE 4, a single reactor coolant or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the Reactor Coolant Loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

In MODE 5, single failure considerations require two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more Reactor Coolant System cold legs less than or equal to 310°F are provided to prevent Reactor Coolant System pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The Reactor Coolant System will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the Reactor Coolant Pumps to when the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity) and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Commission, pursuant to 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the change was implemented.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) for all core planes containing bank "D" control rods and all unrodded core planes shall be established and documented in the Radial Peaking Factor Limit Report before each reload cycle (prior to MODE 2) and provided to the Commission, pursuant to 10 CFR 50.4, upon issuance. In the event that the limit would be submitted at some other time during core life, it will be submitted upon issuance, unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

ANNUAL DIESEL GENERATOR RELIABILITY DATA REPORT

6.9.1.12 The number of tests (valid or invalid) and the number of failures to start on demand for each diesel generator shall be submitted to the NRC annually. This report shall contain the information identified in Regulatory Position C.3.b of NRC Regulatory Guide 1.108, Revision 1, 1977.

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. NPF-8

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

   A. The application for amendment by the licensee, dated July 15, 1991, as supplemented September 10, 1991, and January 10, 1992, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 85 , are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 85

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
2-2	2-2
2-5	2-5
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-4	B 2-4
B 2-5	B 2-5
B 2-6	B 2-6
3/4 1-4	3/4 1-4
3/4 1-19	3/4 1-19
3/4 2-4	3/4 2-4
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-14	3/4 2-14
3/4 2-15	3/4 2-15
3/4 3-10	3/4 3-10
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-30	3/4 3-30
3/4 4-2	3/4 4-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 4-1	B 3/4 4-1
6-19	6-19

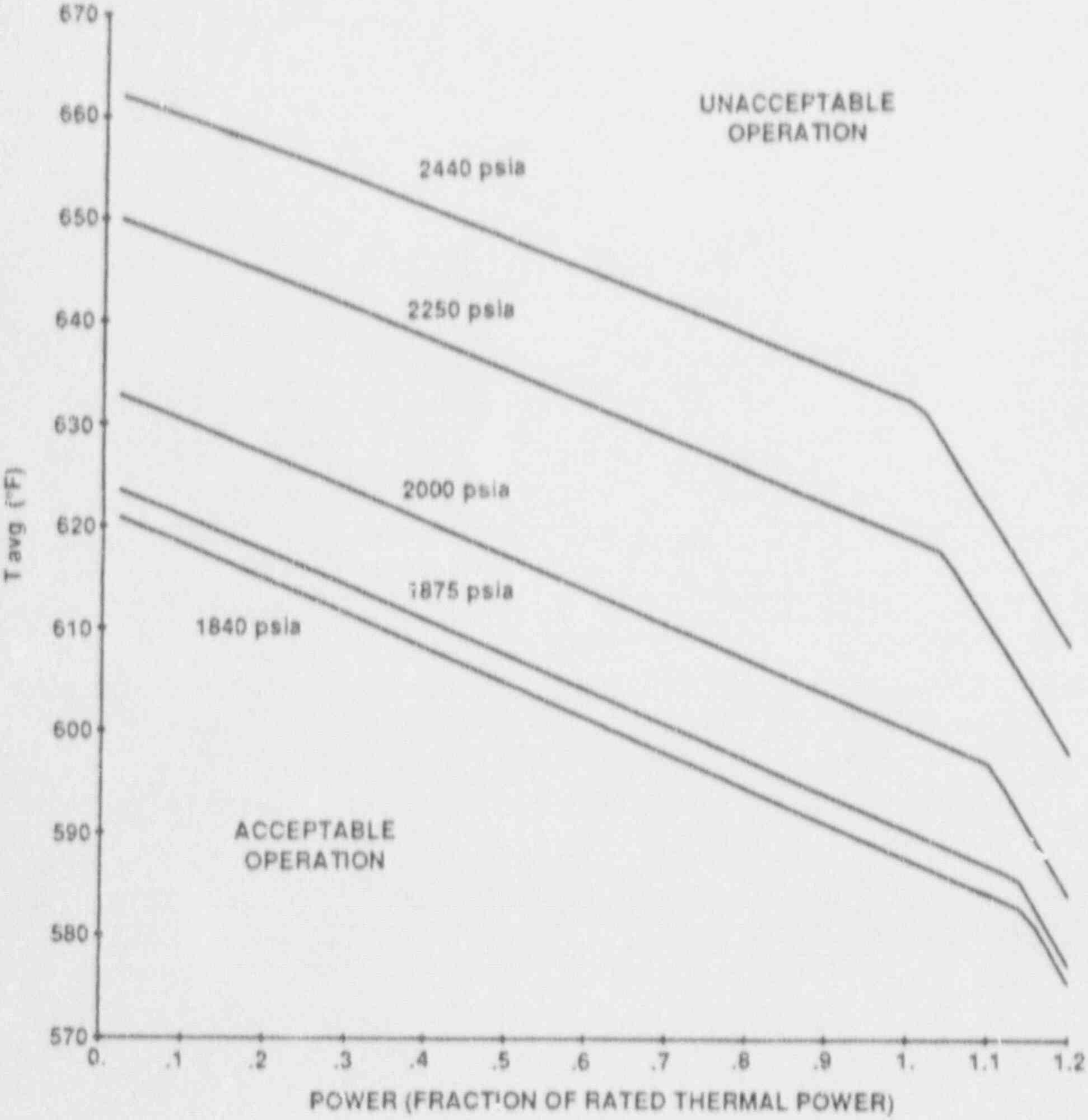

Figure 2.1-1 Reactor Core Safety Limits
Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER
	High Setpoint - $\leq$ 109% of RATED THERMAL POWER	High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 6
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of minimum measured flow per loop*	$\geq$ 88.5% of minimum measured flow per loop*

*Minimum measured flow is 89,290 gpm per loop.

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature ΔT

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_o \left[K_1 - K_2 \frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} \left(T \left(\frac{1}{1 + \tau_6 s}\right) - T'\right) + K_3 (P - P') - f_1 (\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

$T' \leq$ 577.2°F (Maximum Reference T_{avg} at RATED THERMAL POWER);

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

$\frac{1 + \tau_1 s}{1 + \tau_2 s}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 30 sec, τ_2 = 4 sec;

$\frac{1 + \tau_4 s}{1 + \tau_5 s}$ = The function generated by the lead-lag controller for ΔT dynamic compensation;

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 = \tau_5$ = 0 sec;

$\frac{1}{1 + \tau_6 s}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 sec;

s = Laplace transform operator, sec^{-1};

Operation with 3 loops

K_1 = 1.14;

K_2 = 0.0250;

K_3 = 0.001275;

Operation with 2 loops

K_1 = (values blank pending

K_2 = NRC approval of

K_3 = 2 loop operation)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -39 percent and +13 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -39 percent, the ΔT trip setpoint shall be automatically reduced by 1.92 percent of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +13 percent, the ΔT trip setpoint shall be automatically reduced by 2.17 percent of its value at RATED THERMAL POWER.

Note 2: Overpower ΔT

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s}\right)\left(\frac{1}{1 + \tau_6 s}\right) T - K_6 \left(T \left(\frac{1}{1 + \tau_6 s}\right) - T"\right) - f_2(\Delta I)\right]$$

where: ΔT = Measured ΔT by RTD instrumentation;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

$T"$ = Reference T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 577.2°F);

K_4 = 1.07;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

K_6 = 0.00165/°F for $T > T"$, $K_6 = 0$ for $T \leq T"$;

$\frac{\tau_3 s}{1 + \tau_3 s}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

τ_3 = Time constant utilized in the rate lag controller for T_{avg}, τ_3 = 10 sec;

$\frac{1 + \tau_4 s}{1 + \tau_5 s}$ = The function generated by the lead-lag controller for ΔT dynamic compensation;

τ_4 & τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 = \tau_5 = 0$ sec;

$\frac{1}{1 + \tau_6 s}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ sec;

s = Laplace transform operator, $\sec^{-1}$;

$f2(\Delta I) = 0$ for all ΔI.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.8 percent.

Note 4: Pressure value to be determined during initial startup testing. Pressure value of ≤ 55 psia to be used prior to determination of revised value.

Note 5: Pressure value to be determined during initial startup testing.

Note 6: The channel's maximum trip point shall not exceed its computed trip point by more than 2.3 percent.

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. 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 correlations which have 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 DNB thermal design criterion is that the probability of DNB not occurring on the most limiting rod is at least 95 percent (at a 95 percent confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figures 2.1-1 and 2.1-2 show the reactor core safety limits for a range of THERMAL POWER, Reactor Coolant System pressure and average temperature which satisfy the following criteria:

a. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

b. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). Each curve reflects the most limiting result using either low-parasitic (LOPAR) fuel or VANTAGE 5 fuel. The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.25 and 1.24 for the typical and thimble cells, respectively.

c. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

The curves of Figures 2.1-1 and 2.1-2 are based on the most limiting result using an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.65 for VANTAGE 5 fuel and an $F^N_{\Delta H}$ of 1.55 for LOPAR fuel and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$F^N_{\Delta H} = 1.65\ [1 + 0.3\ (1-P)]$ for VANTAGE 5 fuel and

$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$ for LOPAR fuel

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

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, pressurizer and the reactor coolant system piping and fittings are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. 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 analysis.

Manual Reactor Trip

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

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 8 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the DNB design criterion is met for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for multiple dropped rods. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit, thermowell, and RTD response time delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 3 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 2 loop operation exclusive of the Overtemperature delta T setpoint. Two loop operation above the 3 loop P-8 setpoint is permissible after resetting the K1, K2, and K3 inputs to the Overtemperature delta T channels and raising the P-8 setpoint to its 2 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting) under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 10 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will ensure that the DNB design criterion is met during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature delta T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 66% RATED THERMAL POWER will ensure that the DNB design criterion is met during normal operational transients and anticipated transients with 2 loops in operation.

Steam Generator Water Level

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

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall 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 greater than or equal to 1.55×10^6 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 setpoints assure a reactor trip signal is generated before the low flow trip setpoint

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less than or equal to 0.7×10^{-4} delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), condition for power levels up to 70% THERMAL POWER with a linear ramp to 0 delta k/k/°F at 100% THERMAL POWER.

b. Less negative than -4.3×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3.a - MODES 1 and 2* only#
Specification 3.1.1.3.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.3.a above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.3.b above, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.0

See Special Test Exception 3.10.3

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (225 to 231 steps, inclusive)* shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

*The fully withdrawn position used for determining rod drop time shall be greater than or equal to the fully withdrawn position used during subsequent plant operation.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq [\frac{2.45}{P}]\ [K(Z)]$ for P > 0.5 for VANTAGE 5 fuel |

$F_Q(Z) \leq [4.9]\ [K(Z)]$ for $P \leq 0.5$ for VANTAGE 5 fuel and |

$F_Q(Z) \leq [\frac{2.32}{P}]\ [K(Z)]$ for P > 0.5 for LOPAR fuel |

$F_Q(Z) \leq [4.64]\ [K(Z)]$ for $P \leq 0.5$ for LOPAR fuel |

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

and K(Z) is the function obtained from Figure (3.2-2) for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

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

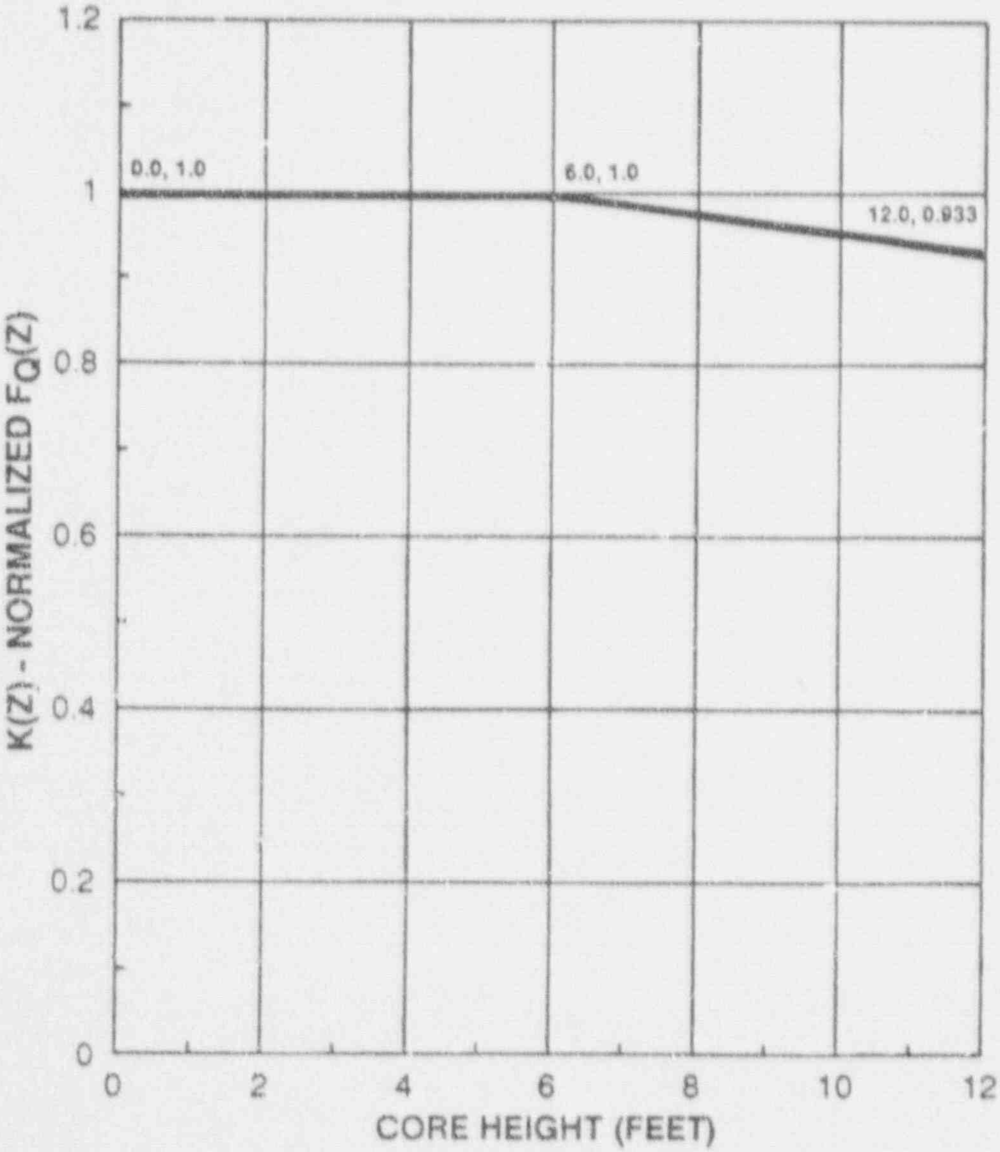

Figure 3.2.2 K(Z) Normalized $F_Q(Z)$ as a Function of Core Height

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$F^N_{\Delta H} \leq 1.65\ [1 + 0.3\ (1-P)]$ for VANTAGE 5 fu- and |

$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$ for LOPAR fuel |

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

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. 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,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

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. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate.

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 less than 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 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.

4.2.5.3 The indicated RCS flow rate shall be verified to be within the acceptable limit at least once per 31 days.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops in Operation	LIMITS: 2 Loops in Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 580.7°F	(**)
Indicated Pressurizer Pressure	$\geq$ 2205 psig*	(**)
Indicated Reactor Coolant System Total Flow Rate	$\geq$ 267,880 gpm***	(**)

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

** Values blank pending NRC approval of 2 loop operation.

*** Value includes a 2.4% flow uncertainty (0.1% feedwater venturi fouling bias included).

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	
	a. High	≤ 0.5 seconds*
	b. Low	Not Applicable
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Power Range, Neutron Flux, High Negative Rate	Not Applicable
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	Not Applicable
7.	Overtemperature ΔT	≤ 6.0 seconds*
8.	Overpower ΔT	Not Applicable
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High-High	$\leq$ 27 psig	$\leq$ 28.3 psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation Logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High-High	$\leq$ 27 psig	$\leq$ 28.3 psig
c. Purge and Exhaust Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 16.2 psig	$\leq$ 17.5 psig
d. Steam Flow in Two Steam Lines--High, Coincident with T_{avg}--Low-Low	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load with $T_{avg} \geq 543°F$	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load with $T_{avg} \geq 540°F$
e. Steam Line Pressure--Low	$\geq$ 585 psig	$\geq$ 575 psig
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water Level--High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. AUXILIARY FEEDWATER		
a. Automatic Actuation Logic	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥ 17% of narrow range instrument span each steam generator	≥ 16% of narrow range instrument span each steam generator
c. Undervoltage - RCP	≥ 2680 volts	≥ 2640 volts
d. S.I.	See 1 above (all SI Setpoints)	
e. Trip of Main Feedwater Pumps	N.A.	N.A.
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 3255 volts bus voltage*	≥ 3222 volts bus voltage* ≤ 3418 volts bus voltage*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3675 volts bus voltage*	≥ 3638 volts bus voltage* ≤ 3749 volts bus voltage*
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, P-11	≤ 2000 psig	≤ 2010 psig
b. Low-Low T_{avg}, P-12		
(Increasing)	544°F	≤ 547°F
(Decreasing)	543°F	≥ 540°F
c. Steam Generator Level, P-14	(See 5. above)	
d. Reactor Trip, P-4	N.A.	N.A.

* Refer to appropriate relay setting sheet calibration requirements.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0(1)/12.0(4)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0(6)
d. Containment Isolation-Phase "A"	$\leq$ 17.0(4)
e. Containment Purge Isolation	$\leq$ 5.0
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0(4)/87.0(1)
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 12.0(4)/22.0(5)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0(6)
d. Containment Isolation-Phase "A"	$\leq$ 17.0(4)/27.0(5)
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0(4)/87.0(5)
5. Steam Flow in Two Steam Lines-High Coincident with T_{avg}--Low-Low	
a. Steam Line Isolation	Not Applicable
6. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 12.0(4)/22.0(5)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 32.0(6)
d. Containment Isolation-Phase "A"	$\leq$ 17.0(4)/27.0(5)
e. Containment Purge Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 77.0(4)/87.0(5)
h. Steam Line Isolation	$\leq$ 7.0

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant Loops listed below shall be OPERABLE and in operation when the rod control system is operational or at least two Reactor Coolant Loops listed below shall be OPERABLE with one Reactor Coolant Loop in operation when the rod control system is disabled by opening the Reactor Trip Breakers or shutting down the rod drive motor/generator sets:*

1. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant pump,
2. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant pump.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required Reactor Coolant Loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one Reactor Coolant Loop in operation and the rod control system operational, within 1 hour open the Reactor Trip Breakers or shut down the rod drive motor/generator sets.

c. With no Reactor Coolant Loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required Reactor Coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required Reactor Coolant Loop(s) shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication at least once per 12 hours.

* All Reactor Coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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) meeting the DNB design criterion during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and 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 for manufacturing tolerances on fuel pellets and rods and measurement uncertainty.

$F^N_{\Delta H}$ 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

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.45 for VANTAGE 5 and 2.32 for LOPAR 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.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the +(5)% target band about the target flux difference, 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 cumulative 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. For 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 actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through 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 2 or more 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.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, NUCLEAR ENTHALPY HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that 1) the design limit on peak local power density is not exceeded, 2) the DNB design criterion is met, and 3) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

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

b. Control rod banks are sequenced with overlapping groups as described in Specification 3.1.3.6.

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.

BASES

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

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate 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 $F^N_{\Delta H}$ contains an 8% allowance for uncertainties. The 8% allowance is based on the following considerations:

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

b. Although rod movement has a direct influence upon limiting F_Q 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 distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

The radial peaking factor $F_{xy}(Z)$, is measured periodically to provide additional assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor limit report per Specification 6.9.1.11 was 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 quadrant 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 startup 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.

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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

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 meet the DNB design criterion throughout each analyzed transient. The indicated T_{avg} value of 580.7°F is based on the average of two control board readings and an indication uncertainty of 2.5°F. The indicated pressure value of 2205 psig is based on the average of two control board readings and an indication uncertainty of 20 psi. The indicated total RCS flow rate is based on one elbow tap measurement from each loop and an uncertainty of 2.4% flow (0.1% flow is included for feedwater venturi fouling).

The 12 hour surveillance of Tavg and pressurizer pressure through the control board readings are sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The 18 month surveillance of the total RCS flow rate is a precision measurement that verifies the RCS flow requirement at the beginning of each fuel cycle and ensures correlation of the flow indication channels with the measured loop flows. The monthly surveillance of the total RCS flow rate is a reverification of the RCS flow requirement using loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of the fuel cycle. The 12 hour RCS flow surveillance is a qualitative verification of significant flow degradation using the control board indicators and the loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of each fuel cycle.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all Reactor Coolant Loops in operation, and meet the DNB design criterion during all normal operations and anticipated transients. In MODES 1 and 2 with one Reactor Coolant Loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two Reactor Coolant Loops provide sufficient heat removal capability for removing core heat even in the event of a bank withdrawal accident; however, a single Reactor Coolant Loop provides sufficient decay heat removal capacity if a bank withdrawal accident can be prevented; i.e., by opening the Reactor Trip Breakers or shutting down the rod drive motor/generator sets.

In MODE 4, a single reactor coolant or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the Reactor Coolant Loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

In MODE 5, single failure considerations require two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more Reactor Coolant System cold legs less than or equal to 310°F are provided to prevent Reactor Coolant System pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The Reactor Coolant System will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the Reactor Coolant Pumps to when the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity) and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Commission, pursuant to 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the change was implemented.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) for all core planes containing bank "D" control rods and all unrodded core planes shall be established and documented in the Radial Peaking Factor Limit Report before each reload cycle (prior to MODE 2) and provided to the Commission, pursuant to 10 CFR 50.4, upon issuance. In the event that the limit would be submitted at some other time during core life, it will be submitted upon issuance, unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

ANNUAL DIESEL GENERATOR RELIABILITY DATA REPORT

6.9.1.12 The number of tests (valid or invalid) and the number of failures to start on demand for each diesel generator shall be submitted to the NRC annually. This report shall contain the information identified in Regulatory Position C.3.b of NRC Regulatory Guide 1.108, Revision 1, 1977.