2) Logic Channel

A logic channel is a group of relay contact matrices which operate in response to the analog channels signals to generate a protective action signal.

~~f~~F. Instrumentation Surveillance

1) Channel Check

Channel check is a qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

2) Channel Functional Test

A channel functional test consists of injecting a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

3) Channel Calibration

Channel calibration consists of the adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarm, or trip, and shall be deemed to include the channel functional test.

~~g~~G. Shutdown

1) Hot Shutdown

The reactor is in the hot shutdown condition when the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 15.3.10 and $T_{avg}$ is at or greater than 540°F.

2) Cold Shutdown

The reactor is in the cold shutdown condition when the reactor has a shutdown margin of at least 1 percent $\Delta k/k$ and reactor coolant temperature is ≤200°F.

3) Refueling Shutdown

The reactor is in the refueling shutdown condition when the reactor is subcritical by at least 5 percent $\Delta k/k$ and $T_{avg}$ is ≤140°F. A refueling shutdown refers to a shutdown to move fuel to and from the reactor core.

4) Shutdown Margin

Shutdown margin is the instantaneous amount of reactivity by which the reactor core would be subcritical if all withdrawn control rods were tripped into the core but the highest worth withdrawn RCCA remains fully withdrawn. If the reactor is shut down from a power condition, the hot shutdown temperature should be assumed. In other cases, no change in temperature should be assumed.

~~h~~H. <u>Power Operation</u>

The reactor is in power operating condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates greater than 2 percent of rated power.

~~i~~I. <u>Refueling Operation</u>

Refueling operation is any operation involving movement of core components (those that could affect the reactivity of the core) within the containment when the vessel head is removed.

~~j~~J. <u>Rated Power</u>

Rated power is here defined as a steady state reactor core output of 1518.5 MWT.*

~~k~~K. <u>Thermal Power</u>

Thermal power is defined as the total core heat transferred from the fuel to the coolant.

* For Unit 2: If the Reactor Coolant System raw measured total flow rate is <174,000 gpm but ≥169,500 gpm, Unit 2 shall be limited to ≤98% rated power.

~~l~~L. <u>Reactor Critical</u>

The reactor is said to be critical when the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

~~m~~M. <u>Low Power Operation</u>

The reactor is in the low-power operating condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates less than or equal to 2% of rated power.

~~n~~N. <u>Fire Suppression Water System</u>

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source; pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard post indicating valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

~~o~~O. Dose Equivalent I-131

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

~~p~~P. $\bar{E}$ - Average Disintegration Energy

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

Q. Core Operating Limits Report

The Core Operating Limits Report is the document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with specification 15.6.9.1.D. Plant operation within these operating limits is addressed in individual specifications.

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, ~~and~~ coolant temperature, reactor coolant system flow, and F(ΔI) during operation.

Objective:

To maintain the integrity of the fuel cladding.
Specifications:

1. ~~The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.~~ During power operation, the departure from nucleate boiling ratio (DNBR) shall be maintained ≥1.17 for the WRB-1 DNB correlation.

2. During power operation, the peak fuel centerline temperature shall be maintained <5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

~~Basis:~~
~~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 excess cladding temperature 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.~~

~~This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.~~

~~The DNB design basis is as follows: there must be at least a 95 percent probability at a 95 percent confidence level that DNB will not occur during steady state operation, normal operational transients, and anticipated transients and is an appropriate margin to DNB for all operating conditions.~~

~~The curves of Figure 15.2.1-1 and 15.2.1-2 are applicable for a core of 14 x 14 OFA. The curves also apply to the reinsertion of previously depleted 14 x 14 standard fuel assemblies into an OFA core. The use of these assemblies is justified by a cycle specific reload analysis. The WRB-1 correlation is used to generate these curves. Uncertainties in plant parameters and DNB correlation predictions are statistically convoluted to obtain a DNBR uncertainty factor. This DNBR uncertainty factor establishes a value of design limit DNBR. This value of design limit DNBR is shown to be met in plant safety analyses, using values of input parameters considered at their nominal values.~~

~~These curves represent the loci of points of thermal power, Reactor Coolant System pressure and average temperature for which the calculated DNBR is no less than the design limit DNBR or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Appropriate rod bow penalties have been included in the generation of these curves. The effects of fuel densification and possible clad flattening have also been taken into account.~~

~~An allowance is included in these curves for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:~~

~~$F^N_{\Delta H}$ = Full Power $F^N_{\Delta H}$ [1 + 0.3 (1 - P)]~~
~~where P is a fraction of rated thermal power.~~

~~The hot channel factors are sufficiently large to account for the degree of malpositioning of full length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached. The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than the design limit DNBR.~~

Basis:

The proper functioning of the Reactor Protection System and the Steam Generator Safety Valves prevents violation of the reactor core safety limits.

The fuel cladding must not sustain damage as a result of steady state operation, normal operational transients or anticipated transients. The reactor core safety limits are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criteria) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

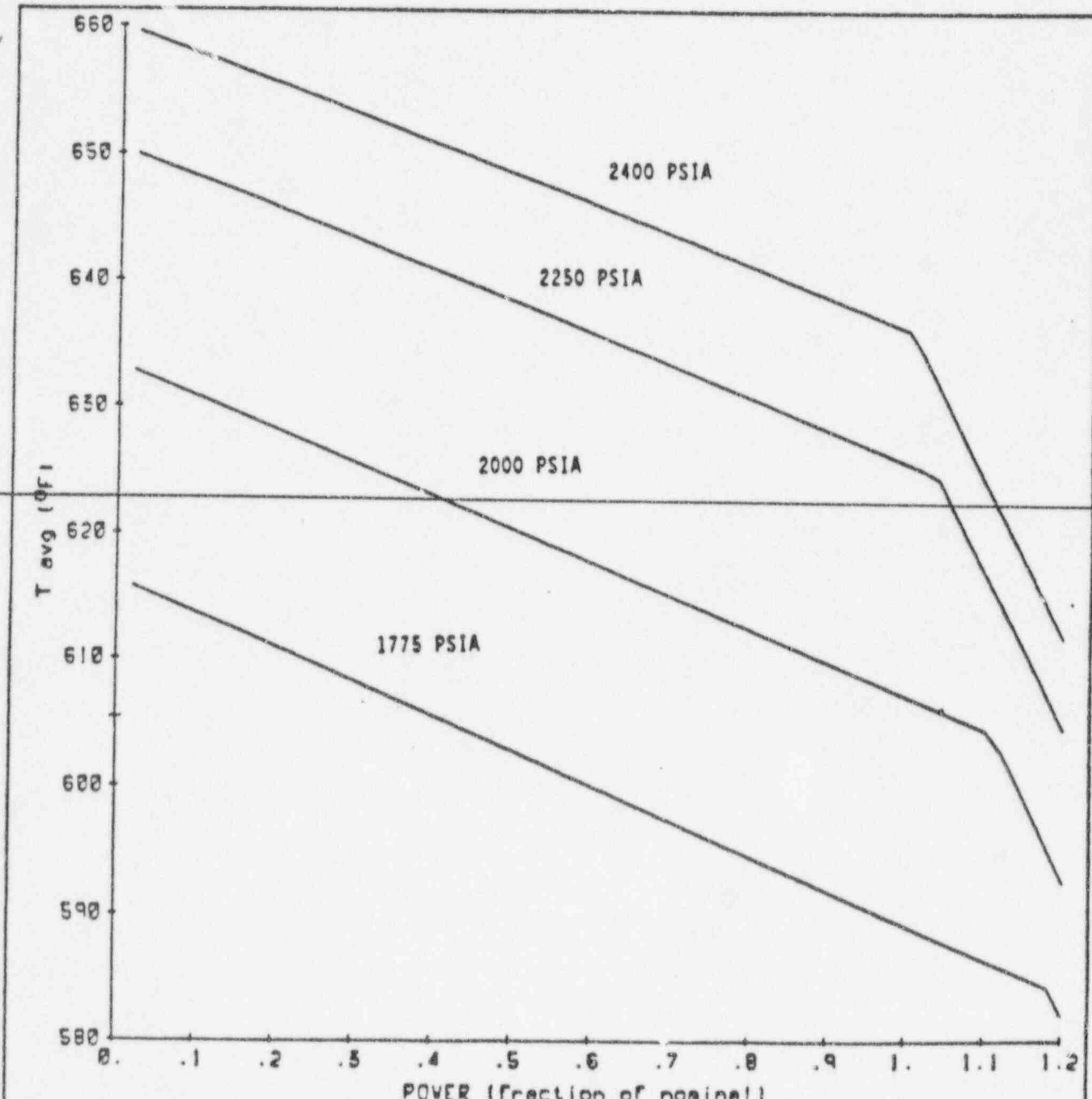
The Reactor Protection System setpoints are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, flow rate, axial flux difference and thermal power level that would result in a DNBR of less than the DNBR limit. Automatic enforcement of these reactor core safety limits is provided by the Reactor Protection System and Steam Generator Safety Valves. The reactor core safety limits represent a design requirement for establishing the Reactor Protection System setpoints.

An additional criterion is applied to the Overtemperature and Overpower Delta-T setpoints of the Reactor Protection System to ensure that the Reactor Core Safety Limits are not violated. The average enthalpy of the hot leg must be less than or equal to the saturation enthalpy. This ensures that the Delta T used in these Reactor Protection System setpoints is proportional to the thermal power.

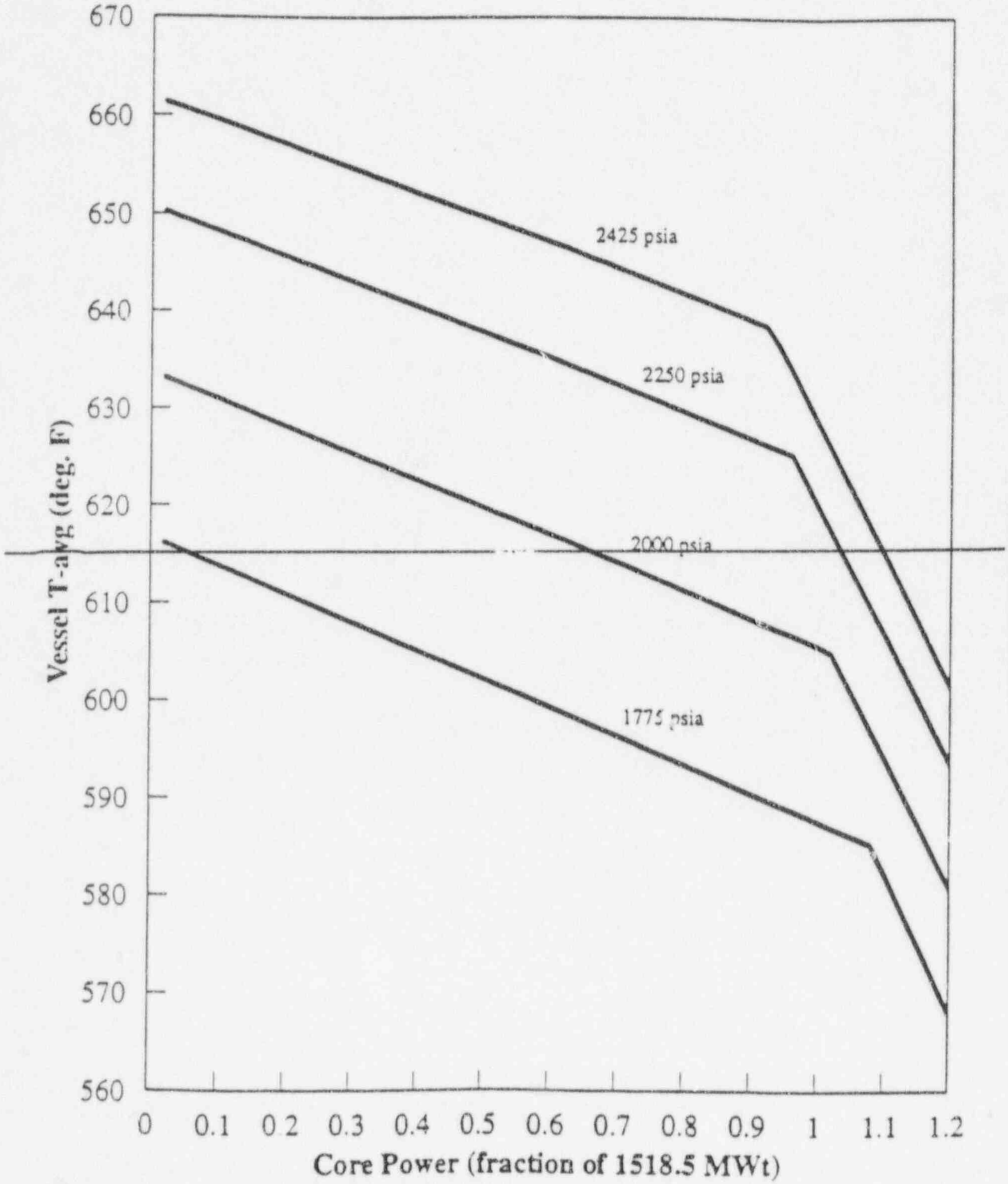
Core exit quality must also be within the limits of the DNBR correlation used in the safety analysis.

The Reactor Core Safety Limits apply only during power operation because it is the only time that the reactor is generating significant thermal power. Automatic protection functions are required to be operable during power operation to ensure operation within the Reactor Core Safety Limits. The Steam Generator Safety Valves or automatic protection actions serve to prevent RCS heat-up to the Reactor Core Safety Limits conditions or to initiate a reactor trip which forces the reactor into Hot Shutdown. Setpoints for the reactor trip functions are specified in 15.2.3.

~~Figure 15.2.1-1~~
~~REACTOR CORE SAFETY LIMITS~~
~~POINT BEACH UNIT 1~~



~~Figure 15.2.1-2~~
~~REACTOR CORE SAFETY LIMITS~~
~~POINT BEACH UNIT 2~~



(3) Low pressurizer pressure - ≥1865 psig for operation at 2250 psia primary system pressure
≥1790 psig for operation at 2000 psia primary system pressure

(4) Overtemperature

$$\Delta T \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P-P') - f(\Delta I)\right)$$

where

$\Delta T_o$ = indicated ΔT at rated power, °F
T = average temperature, °F
T′ ~~≤ 573.9°F (Unit 1)~~ specified in the COLR
~~T′ ≤ 570.0°F (Unit 2)~~
P = pressurizer pressure, psig
P′ ~~= 2235 psig~~ specified in the COLR
$K_1$ ~~≤ 1.30~~ specified in the COLR
$K_2$ ~~= 0.0200~~ specified in the COLR
$K_3$ ~~= 0.000791~~ specified in the COLR
$\tau_1$ ~~= 25 sec~~ specified in the COLR
$\tau_2$ ~~= 3 sec~~ specified in the COLR
$\tau_3$ ~~= 2 sec for Rosemont or equivalent RTD~~ specified in the COLR
~~= 0 sec for Sostman or equivalent RTD~~
$\tau_4$ ~~= 2 sec for Rosemont or equivalent RTD~~ specified in the COLR
~~= 0 sec for Sostman or equivalent RTD~~

~~and~~ f(ΔI) ~~is an even 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, where $q_T$ and $q_B$ are the percent power in the top and bottom halves of the core respectively, and $q_T + q_B$ is total core power in percent of rated power, such that:~~

~~(a) for $q_T - q_B$ within -17, +5 percent, f(ΔI) = 0.~~

~~(b) for each percent that the magnitude of $q_T - q_B$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power for Unit 1, or by an equivalent of 3.1 percent of rated power for Unit 2.~~
specified in the COLR

~~(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.~~

(5) Overpower

$$\Delta T\left(\frac{1}{1+\tau_3 S}\right)$$

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

where

$\Delta T_o$ = indicated ΔT at rated power, °F

T = average temperature, °F

T′ ~~≤ 573.9°F (Unit 1)~~ specified in the COLR

~~T′ ≤ 570.0°F (Unit 2)~~

$K_4$ ~~≤ 1.089 of rated power~~ specified in the COLR

$K_5$ ~~= 0.0262 for increasing T~~ specified in the COLR

~~= 0.0 for decreasing T~~

$K_6$ ~~= 0.00123 for T ≥ T′~~ specified in the COLR

~~= 0.0 for T < T′~~

$\tau_5$ ~~= 10 sec~~ specified in the COLR

$\tau_3$ ~~= 2 sec for Rosemont or equivalent RTD~~ specified in the COLR

~~0 sec for Sostman or equivalent RTD~~

$\tau_4$ ~~= 2 sec for Rosemont or equivalent RTD~~ specified in the COLR

~~0 sec for Sostman or equivalent RTD~~

(6) Undervoltage - ≥75 percent of normal voltage

(7) Indicated reactor coolant flow per loop - ≥90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point ≥55.0 HZ

(b) Low voltage set point ≥75 percent of normal voltage.

With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the reactor core safety limits ~~as shown on Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2~~. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced[6][7].

The overpower, overtemperature and pressurizer pressure system setpoints include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints will not exceed the core safety limits ~~as shown in Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2~~.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower $\Delta T$ trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident[4].

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis[8]. The low loop flow signal is caused by a condition of less than 90 percent flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening

## F. MINIMUM CONDITIONS FOR CRITICALITY

Specification:

1. ~~Except during low power physics tests, the reactor shall not be made critical when the moderator temperature coefficient is more positive than 5 pcm/°F.~~ The moderator temperature coefficient shall be within the limits specified in the COLR.

~~2. Reactor power shall not exceed 70 percent of Rated Power if the moderator temperature coefficient is positive.~~

~~3~~2. During an approach to criticality, at least one (1) count per second, attributable to neutrons, shall register on a narrow range source range nuclear instrument.

~~4~~3. In no case shall the reactor be made critical (other than for the purpose of low level physics tests) to the left of the reactor core criticality curve presented in Figure 15.3.1-1.

~~5~~4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis:

During the early part of the fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below 70 percent of rated thermal power.[(1)(2)] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative when ≥70 percent of rated thermal power. Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within th. limits ~~shown~~ specified in the COLR during Rated Power operation:

1. $T_{avg}$ ~~shall be maintained below 578°F~~.

2. Reactor Coolant System (RCS) pressurizer pressure. ~~shall be maintained:~~
~~a. Unit 1: ≥2205 psig during operation at 2250 psia, or~~
~~≥1955 psig during operation at 2000 psia.~~
~~b. Unit 2: ≥1955 psig during operation at 2000 psia.~~

3. Reactor Coolant System raw measured Total Flow Rate (See Basis).

~~a. Unit 1 ≥ 181,800 gpm Unit 1~~
~~b. Unit 2 ≥ 174,000 gpm Unit 2*~~

Basis:

The reactor coolant system total flow rates for Units 1 and 2 are ~~of 181,800 gpm is~~ based on an assumed measurement uncertainty of 2.1 percent over thermal design flow ~~(178,000 gpm)~~. ~~The reactor coolant system total flow rate for Unit 2 at rated power is 174,000 gpm. This is based on an assumed measurement uncertainty of 2.1 percent over a thermal design flow of 170,400 gpm. However, Unit 2 is analyzed to support operation with a reactor coolant system total flow rate limit of 169,500 gpm. This is based on an assumed measurement uncertainty of 2.1 percent over a thermal design flow of 166,000 gpm. If the Unit 2 RCS raw measured total flow rate is less than 174,000 gpm but greater than or equal to 169,500 gpm, operation is limited to less than or equal to 98% rated power as described in the note to Specification 15.3.1.G.3.b.~~ The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle.

~~* For Unit 2: If the Reactor Coolant System raw measured total flow rate is <174,000 gpm but ≥169,500 gpm, Unit 2 shall be limited to ≤ 98% rated power.~~

## 15.3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, AND CONTAINMENT SPRAY

Applicablity:

Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray.

Objective:

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident.

Specification:

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made critical, except for low temperature physics tests, unless the following conditions associated with that reactor are met:

   a. The refueling water tank contains not less than 275,000 gal. of water with a boron concentration ~~of at least 2000 ppm~~ within the limits specified in the COLR.

   b. Each accumulator is pressurized to at least 700 psig and contains at least 1100 $ft^3$ but no more than 1136 $ft^3$ of water with a boron concentration ~~of at least 2000 ppm~~ within the limits specified in the COLR. Neither accumulator may be isolated.

   c. Two safety injection pumps are operable.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

## 15.3.8 REFUELING

Applicability:

Applies to operating limitations during refueling operations.

Objective:

To ensure that no incident could occur during refueling operations that would affect public health and safety.

Specifications:

During refueling operations:

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).
2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously.
3. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.
4. At least one residual heat removal loop shall be in operation. However, if refueling operations are affected by the residual heat removal loop flow, the operating residual heat removal loop may be removed from operation for up to one hour per eight hour period.
5. During reactor vessel head removal and while loading and unloading fuel from the reactor, ~~a minimum~~ boron concentration ~~of 1800 ppm shall be maintained~~ in the primary coolant system shall be maintained within the limits specified in the COLR .
6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

## 15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and to core power distribution limits.

Objective

To insure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical rod cluster control assembly (RCCA) ejection, and (3) an acceptable core power distribution during power operation.

Specification

A. Bank Insertion Limits

1. ~~When the reactor is critical, except for physics tests and control rod exercises, the shutdown banks shall be fully withdrawn.~~[1] The shutdown bank shall be within the limits specified in the COLR.
2. ~~When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. Exceptions to the insertion limit are permitted for physics tests and control rod exercises.~~(1) The control banks shall be within the limits specified in the COLR.
3. The shutdown margin shall ~~exceed the applicable value as shown in Figure 15.3.10-2 under all steady state operating conditions from 350°F to full power. An exception to the stuck RCCA component of the shutdown margin requirement is permitted for physics tests.~~ be within the limits specified in the COLR.
4. ~~Except for physics tests a shutdown margin of at least 1% Δk/k shall be maintained when the reactor coolant temperature is less than 350°F.~~

~~5~~4. During any approach to criticality, except for physics tests, the critical rod position shall not be lower than the insertion limit for zero power. That is, if the control rods were withdrawn in normal sequence with no other reactivity change, the reactor would not be critical until the control banks were above the insertion limit.

---

~~[1] Fully withdrawn is defined as a bank demand position equal to or greater than 225 steps. This definition is applicable to shutdown and control banks.~~

B. Power Distribution Limits

1. a. ~~Except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:~~ The height dependent heat flux hot channel factor ($F_Q$) and the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) shall be within the limits specified in the COLR.

$$\cancel{F_Q(Z) \leq \frac{(2.50)}{P} \times K(Z) \qquad \text{for } P>0.5}$$

$$\cancel{F_Q(Z) \leq 5.00 \times K(Z) \qquad \text{for } P \leq 0.5}$$

$$\cancel{F^N_{\Delta H} \leq 1.70 \times [1 + 0.3\ (1-P)]}$$

~~Where P is the fraction of full power at which the core is operating, K(Z) is the function in Figure 15.3.10-3 and Z is the core height location of $F_Q$.~~

b. Following a refueling shutdown prior to exceeding 90 percent of rated power and at effective full power monthly intervals thereafter, power distribution maps using the moveable incore detector system shall be made to confirm that the hot channel factor limits are satisfied. The measured hot channel factors shall be increased in the following way:

(1) The measurement of total peaking factor, $F_Q^{MEAS}$, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

(2) The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error.

c. If a measured hot channel factor exceeds the full power limit of Specification 15.3.10.B.1.a, the reactor power and power range high setpoints shall be reduced until those limits are met. If subsequent flux mapping cannot, within 24 hours, demonstrate that the full power hot channel factor limits are met, the overpower and overtemperature ΔT trip setpoints shall be similarly reduced and reactor power limited such that Specification 15.3.10.B.1.a above is met.

2. a. The indicated axial flux difference (AFD) shall be maintained within the ~~allowed operational space defined by Figure 15.3.10-4~~ limits specified in the COLR except during physics tests. The physics test exemption applies provided that the thermal power is less than or equal to 85% of Rated Power and the limits of Specification 15.3.10.B.1.a are satisfied. During suspension of the specification, the thermal power shall be determined to be less than or equal to 85% of rated thermal power at least once per hour. In addition, the surveillance requirements of 15.3.10.B.1.b shall be performed at least once per 12 hours.

   b. If the indicated AFD deviates from the ~~Figure 15.3.10-4 limits~~ requirements of 15.3.10.B.2.a, the AFD shall be restored to within the ~~Figure 15.3.10-4 limits~~ requirements of 15.3.10.B.2.a within 15 minutes. If this cannot be accomplished, then reactor power shall be reduced until the AFD is within the envelope or the power level is less than 50 percent of Rated Power. Normally the rate of power reduction is 15% per hour. Once AFD has been returned to and maintained within the operating envelope, power level is no longer restricted. If it is necessary to reduce power to 50%, the Power Range Neutron Flux-High Trip Setpoints shall be reduced to less than or equal to 55 percent within the next 4 hours.

   c. A power increase to a level greater than 50 percent of Rated Power is contingent upon the indicated AFD being within the ~~Figure 15.3.10-4 limits~~ requirements of 15.3.10.B.2.a.

   d. Alarms shall normally be used to indicate non-conformance with the flux difference requirements of 15.3.10.B.2.a and 15.3.10.B.2.b. If the alarms are totally out of service, the AFD shall be noted and conformance with the limits assessed every hour for the first 24 hours, and half-hourly thereafter.

   e. The indicated AFD shall be considered outside of its limits when at least 2 operable excore channels are indicating the AFD to be outside the limits.

Basis

Insertion Limits and Shutdown Margin

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration.

~~During power operation, the shutdown banks are fully withdrawn. Fully withdrawn is defined as a bank demand position equal to or greater than 225 steps.~~ The shutdown bank insertion limits are given in the COLR. Evaluation has shown that positioning control rods at 225 steps, or greater, has a negligible effect on core power distributions and peaking factors. Due to the low reactivity worth in this region of the core and the fact that, at 225 steps, control rods are only inserted one step into the active fuel region of the core, positioning rods at this position or higher has minimal effect. This position is varied, based on a predetermined schedule, in order to minimize wear of the guide cards in the guide tubes of the RCCA's.

The control rod insertion limits specified in the COLR provide for achieving hot shutdown by reactor trip at any time and assume the highest worth control rod remains fully withdrawn. A 10% margin in reactivity worth of the control rods is included to assure meeting the assumptions used in the accident analysis. So a reactor trip occurring during power operation will put the reactor into the hot shutdown condition. In addition, the insertion limits provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection and provide for acceptable nuclear peaking factors. The specified control rod insertion limits take into account the effects of fuel densification. The rods are withdrawn in the sequence of A, B, C, D with overlap between banks. The overlap between successive control banks is provided to compensate for the low differential rod worth near the top and bottom of the core.

When the insertion limits specified in the COLR are observed ~~and the control rod banks are above the solid lines shown on Figure 15.3.10-1~~, the shutdown requirement is met. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident.

~~Figure 15.3.10-2~~ As shown in the COLR, ~~shows~~ the shutdown margin is equivalent to 2.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses assume 1% or greater reactivity shutdown margin. Shutdown margin calculations include the effects of axial power distribution. One may assume no change in core poisoning due to xenon, samarium or soluble boron.

Power Distribution

Design criteria have been chosen which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must not be less than the limit DNBR in normal operation or in short-term transients.

In addition to the above, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss-of-coolant accident analysis based upon the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss-of-coolant accident. To aid in specifying the limits on power distribution, the following hot channel factors are defined:

> $F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the 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. Imposed limits pertain to the maximum $F_Q(Z)$ in the core.
>
> $F_Q^E$, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically, the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a fuel rod to the average fuel rod power. Imposed limits pertain to the maximum $F^N_{\Delta H}$ in the core, that is the fuel rod with the highest integrated power. It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat flux is obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the norizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided the following conditions are observed, the hot channel factor limits will be met:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 24 steps from the bank demand position, when the bank demand position is between 30 steps and 215 steps. A misalignment of 36 steps is allowed when the bank position is less than or equal to 30 steps, or, when the bank position is greater than or equal to 215 steps, due to the small worth and consequential effects of an individual rod misalignment.
2. Control rod banks are sequenced with overlapping banks as described in ~~Figure 15.3.10-1~~ the COLR.
3. The full-length control bank insertion limits are not violated.
4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation of $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. ~~In Specification 15.3.10.B.1.a, $F_Q$ is arbitrarily limited for $p \leq 0.5$ (except for low power physics tests).~~

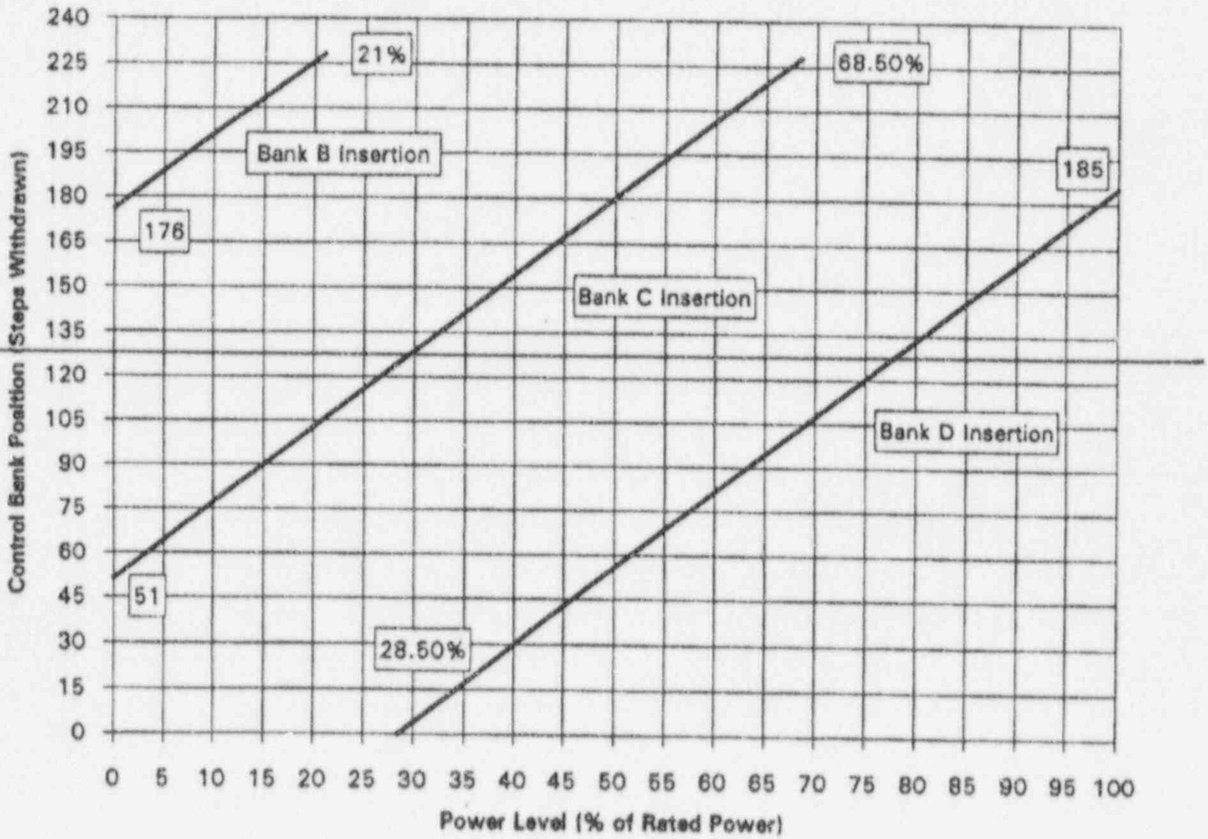
An upper bound envelope ~~of 2.50 times the normalized peaking factor axial dependence of Figure 15.3.10-3~~ (specified in the COLR) consistent with the Technical Specifications on power distribution control as given in Section 15.3.10 was used in the large and small break LOCA analyses. The envelope was determined based on allowable power density distributions at full power restricted to axial flux difference ($\Delta I$) values consistent with those in Specification 15.3.10.B.2. The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.70/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting $F_Q$, (b) while the operator has a direct influence on $F_Q$ through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in $F_Q$ by tighter axial control; but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based upon measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mappinj provides additional assurance that the nuclear design bases remain inviolate and identify operational

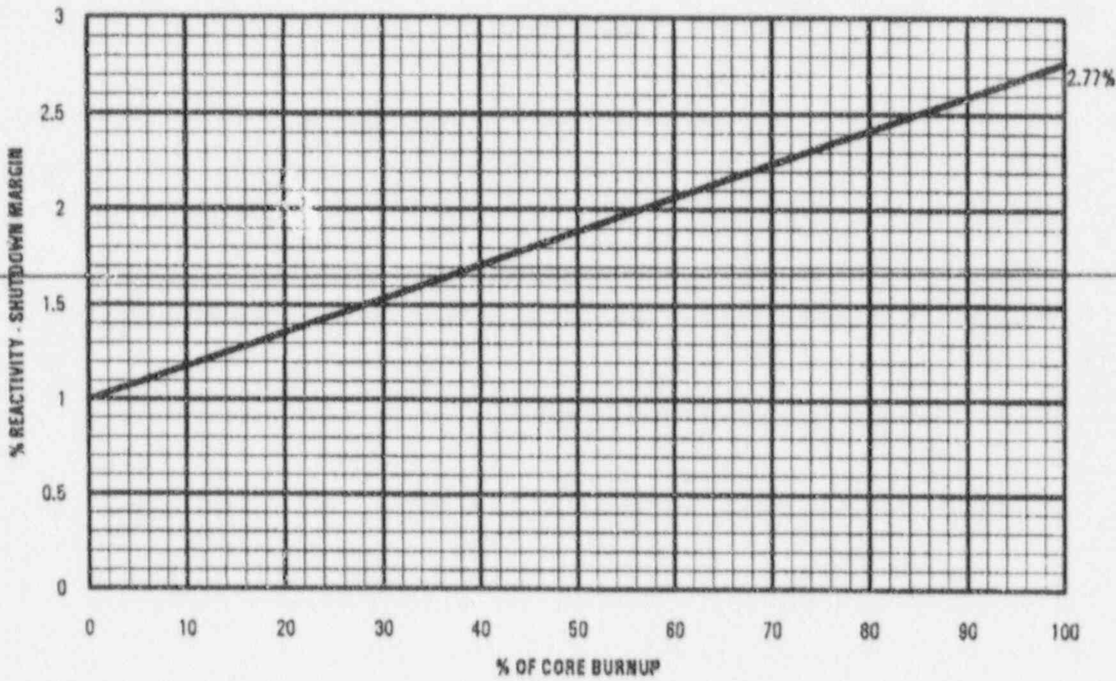
~~FIGURE 15.3.10-1~~

~~CONTROL BANK INSERTION LIMITS~~

~~POINT BEACH UNITS 1 AND 2~~




~~Note: The "fully withdrawn" parking position range can be used without violating this Figure.~~



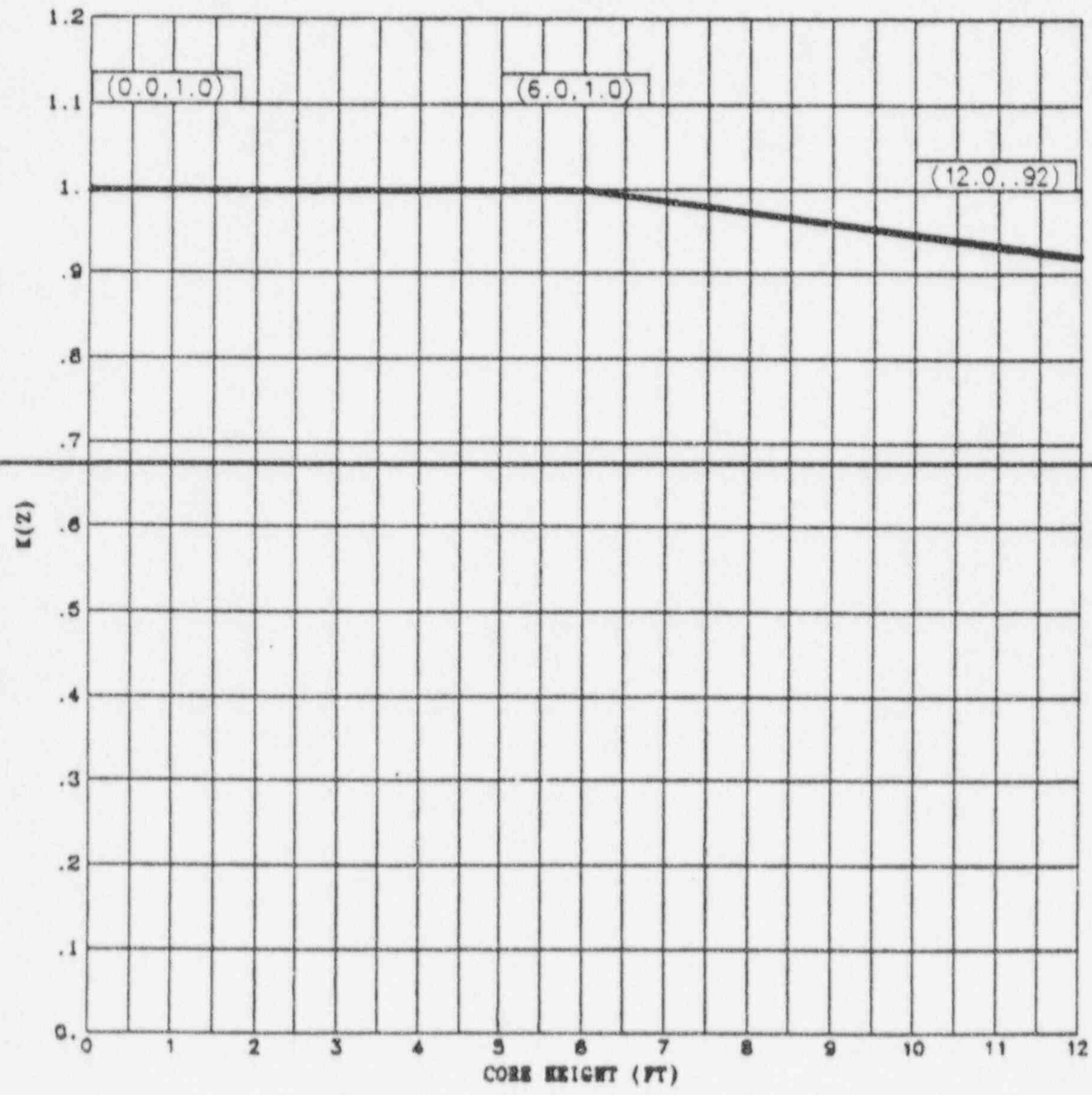

~~FIGURE 15.3.10-2~~

~~REQUIRED SHUTDOWN MARGIN~~

~~FIGURE 15.3.10-3~~

~~POINT BEACH UNITS 1 AND 2~~

~~HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE~~

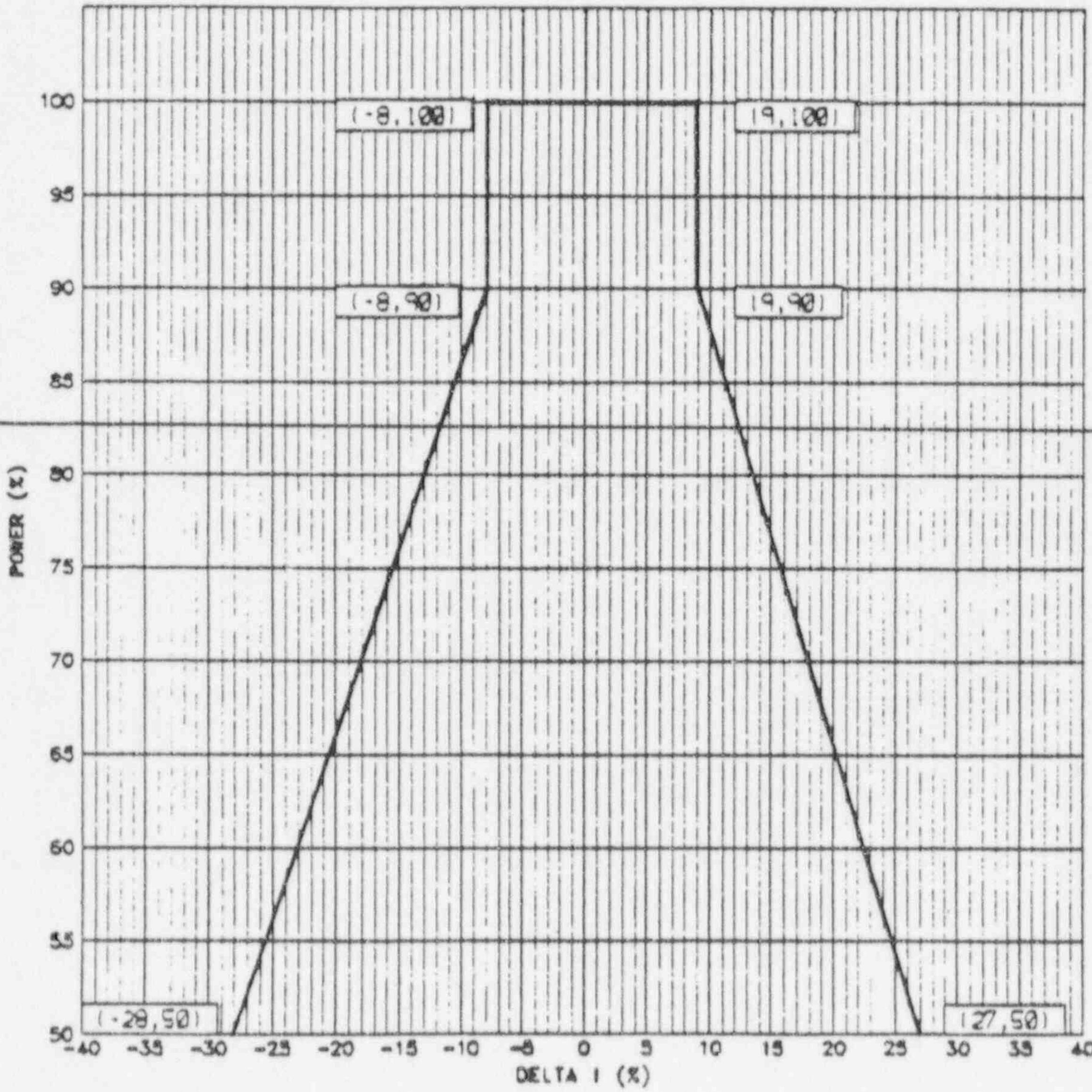


~~FIGURE 15.3.10-4~~

~~FLUX DIFFERENCE~~

~~OPERATING ENVELOPE~~

~~POINT BEACH UNITS 1 AND 2~~



D. Core Operating Limits Report (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following Specifications:

| | |
|---|---|
| 15.3.10.A.3 | Shutdown Margin |
| 15.3.1.F.1 | Moderator Temperature Coefficient |
| 15.3.10.A.1 | Shutdown Bank Insertion Limit |
| 15.3.10.A.2 | Control Bank Insertion Limits |
| 15.3.10.B.1.a | Height Dependent Heat Flux Hot Channel Factor ($F_Q$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) |
| 15.3.10.B.2.a | Axial Flux Difference |
| 15.2.3.1.B(4) | Overtemperature ΔT Setpoint |
| 15.2.3.1.B(5) | Overpower ΔT Setpoint |
| 15.3.1.G | RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits |
| 15.3.3.A.1.b | Accumulator Boron Concentration |
| 15.3.3.A.1.a | Refueling Water Storage Tank (RWST) Boron Concentration |
| 15.3.8.5 | Refueling Boron Concentration |

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The NRC-approved methodologies are listed below for each of the core operating limits:

| COLR Section | Parameter | NRC Approved Methodology |
|---|---|---|
| 2.1 | Shutdown Margin | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.2 | Moderator Temperature Coefficient | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.3 | Shutdown Bank Insertion Limit | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.4 | Control Bank Insertion Limits | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.5 | Height Dependent Heat Flux Hot Channel Factor ($F_Q$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985<br><br>Thadani to Johnson, "Acceptance for Reference of Licensing Topical Report, WCAP-10924, 'Westinghouse Large Break LOCA Best Estimate Methodology,' Addendum 4, 'Model Revision'," February 1991 |
| 2.6 | Axial Flux Difference | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.7 | Overtemperature ΔT Setpoint | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.8 | Overpower ΔT Setpoint | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.9 | RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985<br><br>WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 |
| 2.10 | Accumulator Boron Concentration | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.11 | Refueling Water Storage Tank (RWST) Boron Concentration | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.12 | Refueling Boron Concentration | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |

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

4. The COLR, including any mid-cycle revisions or supplements, shall be provided to the NRC upon issuance for each reload cycle.

15.6.9.2 Unique Reporting Requirements

A. Integrated Leak Rate Test

Each integrated leak test shall be the subject of a summary technical report, including results of the local leak rate tests and isolation valve leak rate tests since the last report. The report shall include analysis and interpretations of the results which demonstrate compliance with specified leak rate limits.

B. Poison Assembly Removal From Spent Fuel Storage Racks

Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

C. Overpressure Mitigating System Operation

In the event the overpressure mitigating system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

# Point Beach Nuclear Plant

# Core Operating Limits Report

Unit 1, Cycle 23
Unit 2, Cycle 22



Note: This report is not part of the PBNP Technical Specifications. This report is referenced in the PBNP Technical Specifications.

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## 1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Point Beach Nuclear Plant has been prepared in accordance with the requirements of Technical Specification (TS) 15.6.9.1.D.

A cross-reference between the COLR sections and the PBNP Technical Specifications affected by this report is given below:

| COLR Section | PBNP TS | Description |
|---|---|---|
| 2.1 | 15.3.10.A.3 | Shutdown Margin |
| 2.2 | 15.3.1.F.1 | Moderator Temperature Coefficient |
| 2.3 | 15.3.10.A.1 | Shutdown Bank Insertion Limit |
| 2.4 | 15.3.10.A.2 | Control Bank Insertion Limits |
| 2.5 | 15.3.10.B.1.a | Height Dependent Heat Flux Hot Channel Factor ($F_Q$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) |
| 2.6 | 15.3.10.B.2.a | Axial Flux Difference |
| 2.7 | 15.2.3.1.B(4) | Overtemperature $\Delta$T Setpoint |
| 2.8 | 15.2.3.1.B(5) | Overpower $\Delta$T Setpoint |
| 2.9 | 15.3.1.G | RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits |
| 2.10 | 15.3.3.A.1.b | Accumulator Boron Concentration |
| 2.11 | 15.3.3.A.1.a | Refueling Water Storage Tank (RWST) Boron Concentration |
| 2.12 | 15.3.8.5 | Refueling Boron Concentration |
| Figure 1 | Figure 15.3.10-2 | Required Shutdown Margin |
| Figure 2 | Figure 15.3.10-1 | Control Bank Insertion Limits |
| Figure 3 | Figure 15.3.10-3 | Hot Channel Factor Normalized Operating Envelope |
| Figure 4 | Figure 15.3.10-4 | Flux Difference Operating Envelope |

## 2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC approved methodologies specified in Technical Specification 15.6.9.1.D.

### 2.1 Shutdown Margin (TS 15.3.10.A.3)

2.1.1 The shutdown margin shall exceed the applicable value as shown in Figure 1 under all steady-state operating conditions from 350°F to full power. An exception to the stuck RCCA component of the shutdown margin requirement is permitted for physics tests.

2.1.2 Except for physics tests, a shutdown margin of at least 1% $\Delta k/k$ shall be maintained when the reactor coolant temperature is less than 350°F.

### 2.2 Moderator Temperature Coefficient (TS 15.3.1.F.1)

2.2.1 Except during low-power physics tests, the reactor shall not be made critical when the moderator temperature coefficient is more positive than 5 pcm/°F.

2.2.2 Reactor power shall not exceed 70 percent of Rated Power if the moderator temperature coefficient is positive.

### 2.3 Shutdown Bank Insertion Limit (TS 15.3.10.A.1)

2.3.1 When the reactor is critical, except for physics tests and control rod exercises, the shutdown banks shall be fully withdrawn. Fully withdrawn is defined as a bank demand position equal to or greater than 225 steps.

### 2.4 Control Bank Insertion Limits (TS 15.3.10.A.2)

2.4.1 When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 2. Exceptions to the insertion limit are permitted for physics tests and control rod exercises. Fully withdrawn is defined as a bank demand position equal to or greater than 225 steps.

**2.5 Height Dependent Heat Flux Hot Channel Factor ($F_Q$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) (TS 15.3.10.B.1.a)**

2.5.1 Except during low power physics tests, the Height Dependent Heat Flux Hot Channel Factor must meet the following limits:

$$F_Q(Z) \leq \frac{(2.50)}{P} \times K(Z) \qquad \text{for } P>0.5$$

$$F_Q(Z) \leq 5.00 \times K(Z) \qquad \text{for } P\leq 0.5$$

where: P is the fraction of full power at which the core is operating,

K(Z) is the function in Figure 3, and

Z is the core height location of $F_Q$.

2.5.2 Except during low power physics tests, the Nuclear Enthalpy Rise Hot Channel Factor must meet the following limit:

$$F^N_{\Delta H} < 1.70 \times \left[1 + 0.3(1-P)\right]$$

where: P is the fraction of full power at which the core is operating.

**2.6 Axial Flux Difference (TS 15.3.10.B.2.a)**

2.6.1 The indicated axial flux difference shall be maintained within the allowed operational space defined by ure 4 except during physics tests.

### 2.7 Overtemperature ΔT Setpoint (TS 15.2.3.1.B(4))

Overtemperature ΔT setpoint parameter values:

| | | |
|---|---|---|
| T′ | ≤ | 573.9°F (Unit 1) |
| | ≤ | 570.0°F (Unit 2) |
| P′ | = | 2235 psig |
| $K_1$ | < | 1.30 |
| $K_2$ | = | 0.0200 |
| $K_3$ | = | 0.000791 |
| $\tau_1$ | = | 25 sec |
| $\tau_2$ | = | 3 sec |
| $\tau_3$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |
| $\tau_4$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |

f(ΔI) is an even 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, where $q_t$ and $q_b$ are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within -17, +5 percent, f(ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power for Unit 1, or by an equivalent of 3.1 percent of rated power for Unit 2.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

**2.8 Overpower ΔT Setpoint (TS 15.2.3.1.B(5))**

Overpower ΔT setpoint parameter values:

| | | |
|---|---|---|
| T′ | ≤ | 573.9°F (Unit 1) |
| | ≤ | 570.0°F (Unit 2) |
| $K_4$ | ≤ | 1.089 of rated power |
| $K_5$ | = | 0.0262 for increasing T |
| | = | 0.0 for decreasing T |
| $K_6$ | = | 0.00123 for T ≥ T′ |
| | = | 0.0 for T < T′ |
| $\tau_5$ | = | 10 sec |
| $\tau_3$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |
| $\tau_4$ | = | 2 sec for Rosemont or equivalent RTD |
| | = | 0 sec for Sostman or equivalent RTD |

**2.9 RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits (TS 15.3.1.G)**

2.9.1 $T_{avg}$ shall be maintained below 578°F

2.9.2 Reactor Coolant System (RCS) pressurizer pressure shall be maintained:

a. Unit 1: ≥ 1955 psig during operation at 2000 psia.
b. Unit 2: ≥ 1955 psig during operation at 2000 psia.

2.9.3 Reactor Coolant System raw measured Total Flow Rate shall be maintained:

a. Unit 1: ≥ 181,800 gpm
b. Unit 2: ≥ 174,000 gpm

The RCS flow rate limit for Unit 2 at rated power is 174,000 gpm. However, Unit 2 is analyzed to support operation with a reactor coolant system total flow rate limit of 169,500 gpm. This is based on an assumed measurement uncertainty of 2.1 percent over a thermal design flow of 166,000 gpm. If the Unit 2 RCS raw measured total flow rate is less than 174,000 gpm but greater than or equal to 169,500 gpm, operation is limited to less than or equal to 98% rated power.

**2.10 Accumulator Boron Concentration (TS 15.3.3.A.1.b)**

2.10.1 The accumulator boron concentration shall be at least 2000 ppm.

**2.11 Refueling Water Storage Tank (RWST) Boron Concentration (TS 15.3.3.A.1.a)**

2.11.1 The refueling water storage tank boron concentration shall be at least 2000 ppm.

**2.12 Refueling Boron Concentration (TS 15.3.8.5)**

2.12.1 During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 1800 ppm shall be maintained in the primary coolant system.

Figure 1: Required Shutdown Margin

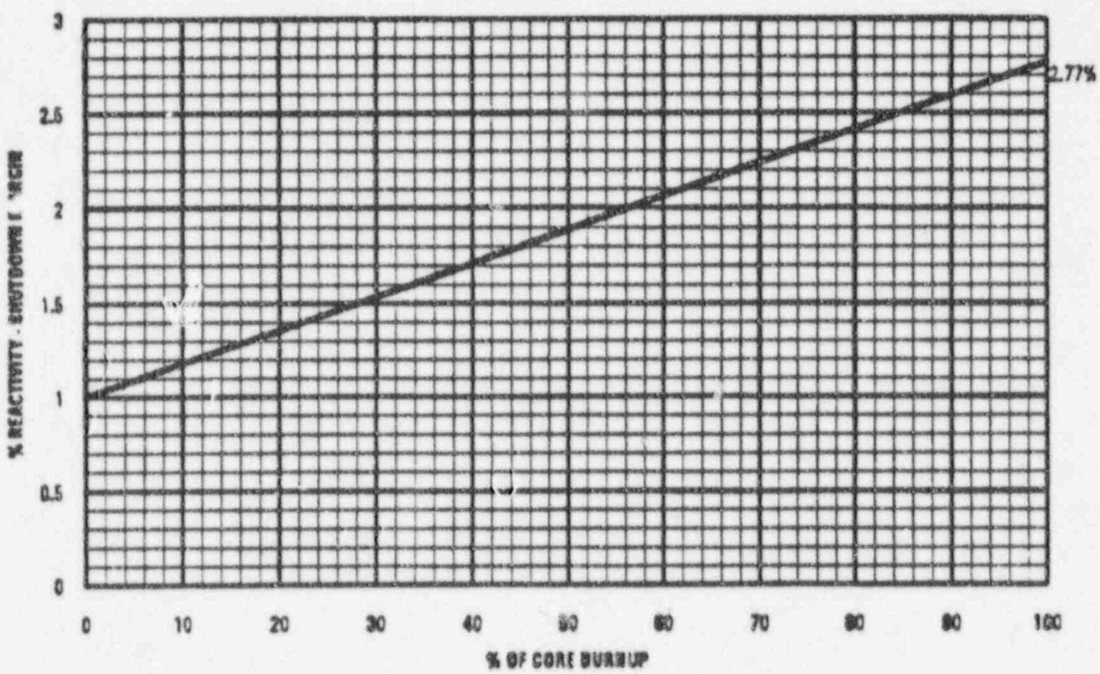
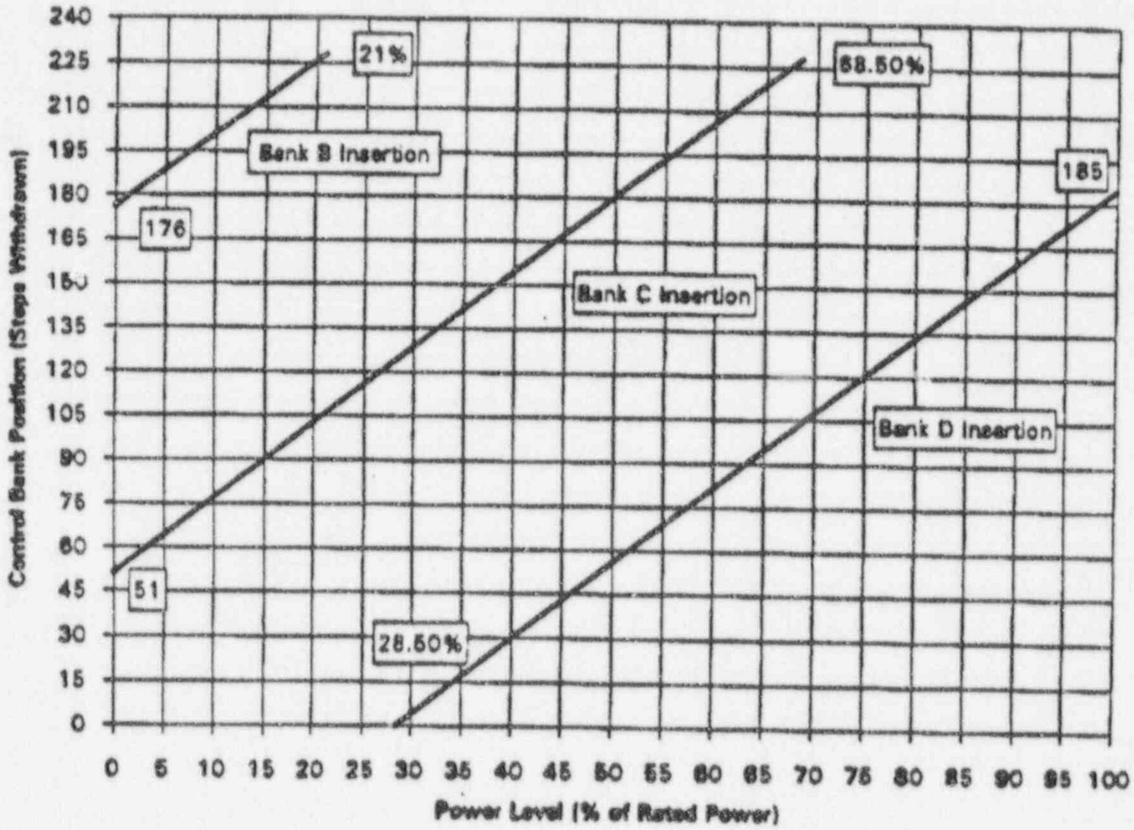


Figure 2: Control Bank Insertion Limits




Note: The "fully withdrawn" parking position range can be used without violating this Figure.

Figure 3: Hot Channel Factor Normalized Operating Envelope

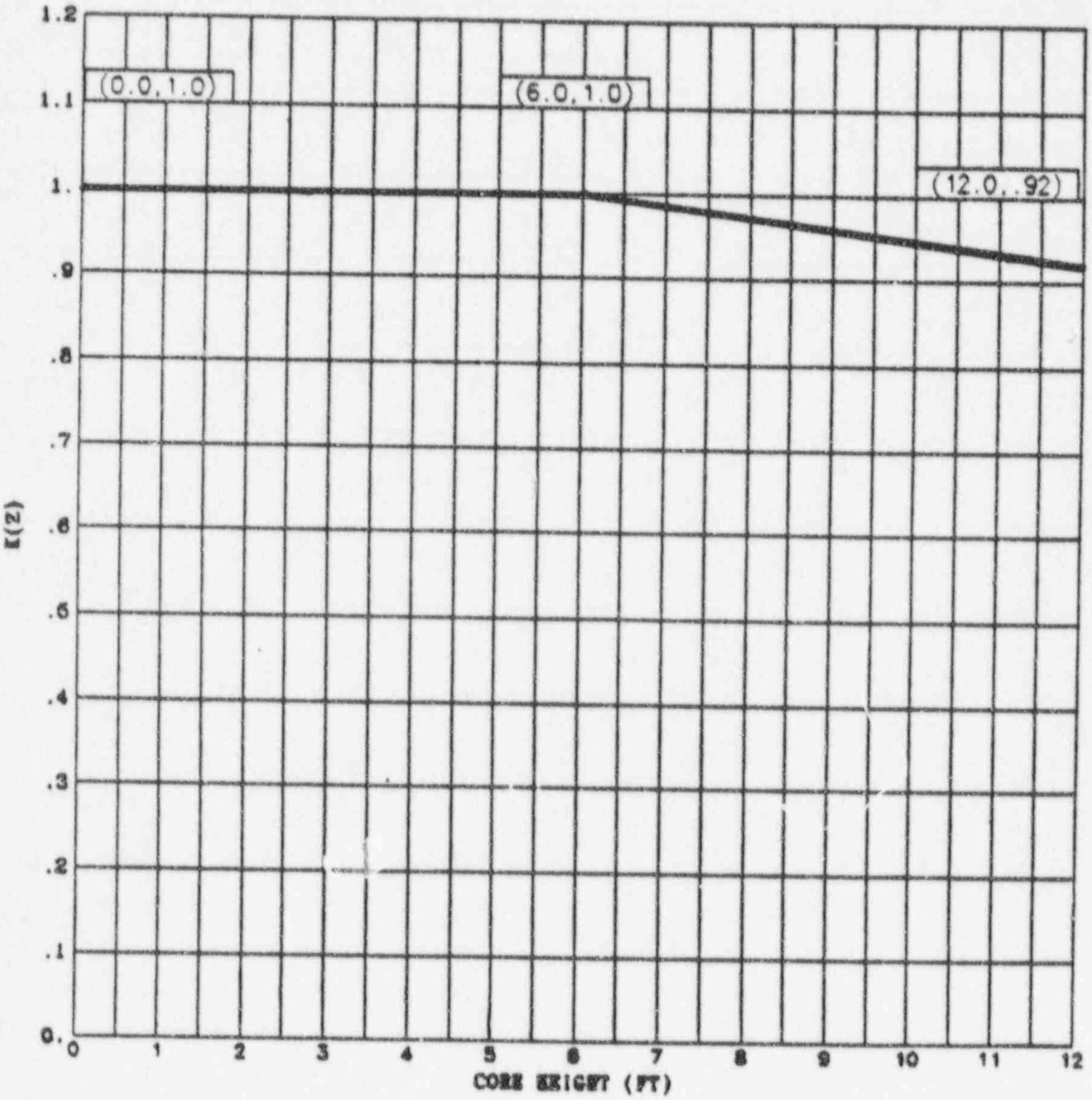


Figure 4: Flux Difference Operating Env 'ope

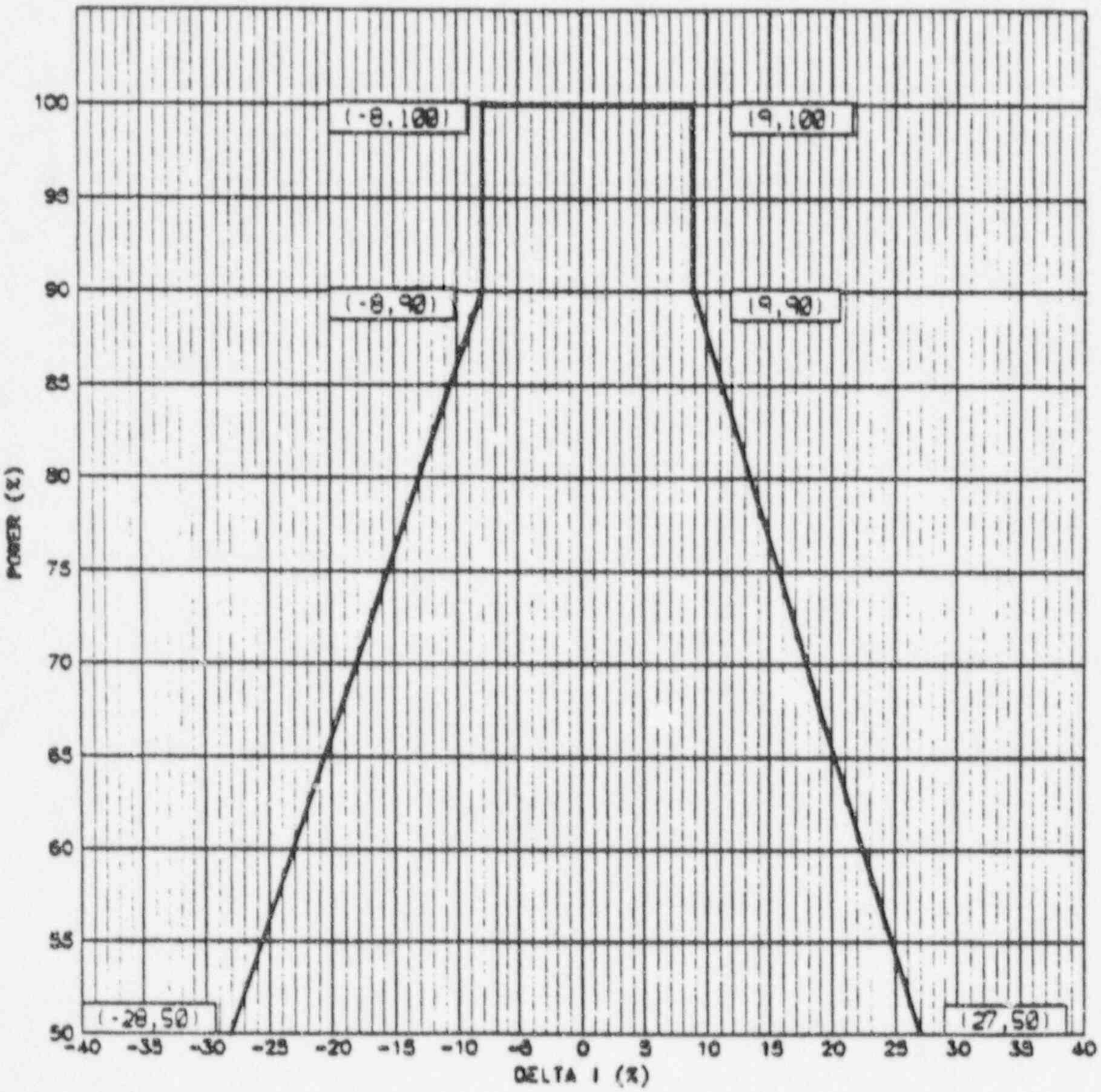


Table 1: NRC Approved Methodologies for COLR Parameters

| COLR Section | Parameter | NRC Approved Methodology |
|---|---|---|
| 2.1 | Shutdown Margin | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.2 | Moderator Temperature Coefficient | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.3 | Shutdown Bank Insertion Limit | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.4 | Control Bank Insertion Limits | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.5 | Height Dependent Heat Flux Hot Channel Factor ($F_Q$) and Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985<br><br>Thadani to Johnson, "Acceptance for Reference of Licensing Topical Report, WCAP-10924, 'Westinghouse Large Break LOCA Best Estimate Methodology,' Addendum 4, 'Model Revisions'," February 1991 |
| 2.6 | Axial Flux Difference | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.7 | Overtemperature ΔT Setpoint | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.8 | Overpower ΔT Setpoint | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
| 2.9 | RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985<br><br>WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 |
| 2.10 | Accumulator Boron Concentration | WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 |
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