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 during operation.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1. 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.

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.

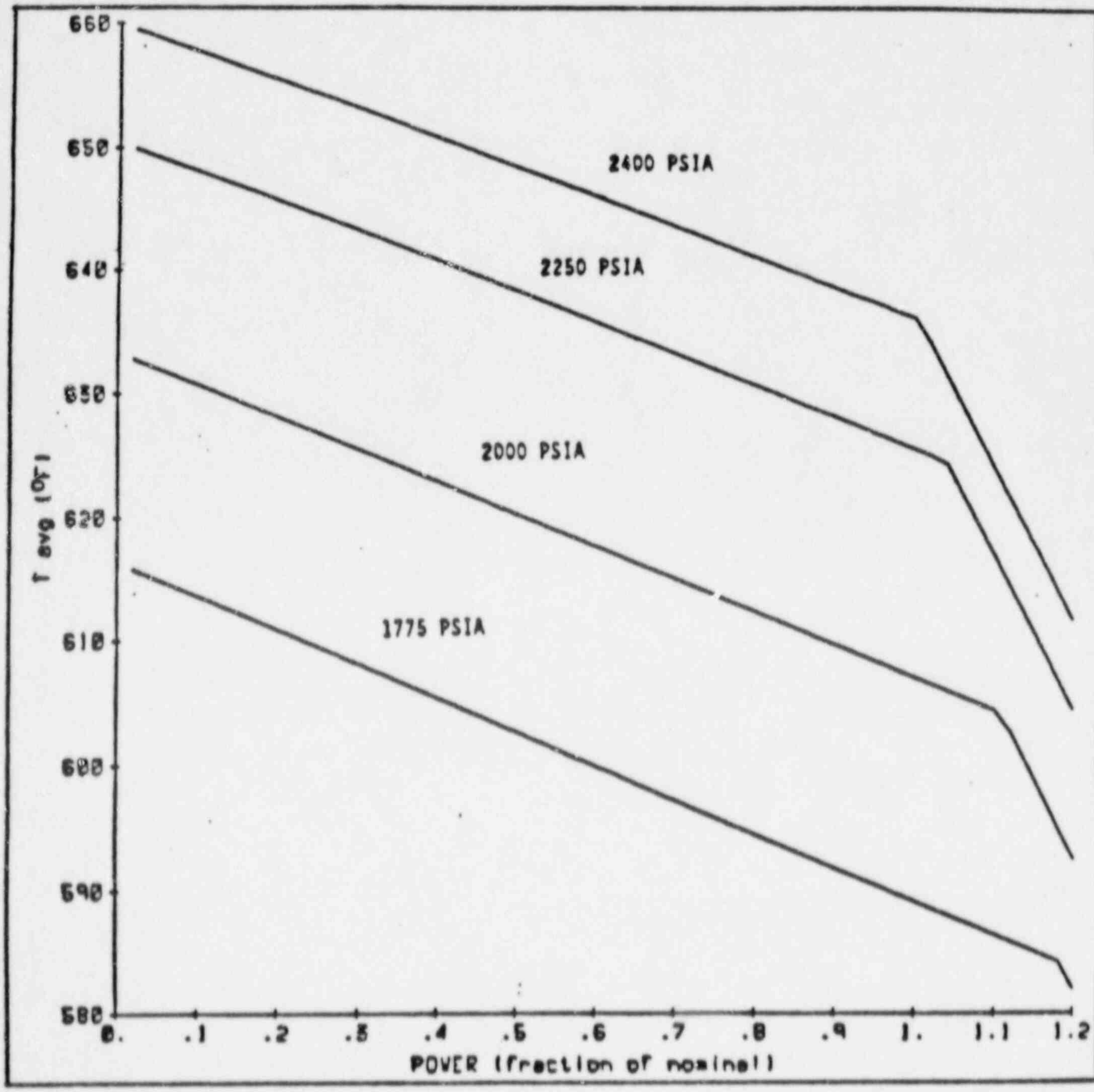
This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative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

Figure 15.2.1-1

REACTOR CORE SAFETY LIMITS
POINT BEACH UNITS 1 AND 2



(3) Low pressurizer pressure - $\geq$1865 psig for operation at 2250 psia primary system pressure
$\geq$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

ΔTo	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T^1	$\leq$	573.9°F
P	=	pressurizer pressure, psig
P^1	=	2235 psig
K_1	$\leq$	1.30
K_2	=	0.0200
K_3	=	0.000791
τ_1	=	25 sec
τ_2	=	3 sec
τ_3	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD

and $f(\Delta I)$ 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, $f(\Delta I) = 0$.

(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.

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

$$\leq \Delta T_o \left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S + 1}\right)\left(\frac{1}{1+\tau_4 S}\right) T - K_6 \left[T \frac{1}{1+\tau_4 S}\right) - T^1\right] - f(\Delta I)\right]$$

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T^1	$\leq$	573.9°F
K_4	$\leq$	1.089 of rated power
K_5	=	0.0262 for increasing T
	=	0.0 for decreasing T
K_6	=	0.00123 for $T \geq T$
	=	0.0 for $T < T^1$
τ_5	=	10 sec
		f (ΔI) as defined in (4) above,
τ_3	=	2 sec for Rosemont or equivalent RTD
		0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
		0 sec for Sostman or equivalent RTD

(6) Undervoltage - $\geq$75 percent of normal voltage

(7) Indicated reactor coolant flow per loop - $\geq$90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point $\geq$55.0 HZ

(b) Low voltage set point $\geq$75 percent of normal voltage.

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

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 ΔT 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

C. MAXIMUM COOLANT ACTIVITY

Specification:

The specific activity of the reactor coolant shall be limited to:

1. Less than or equal to 1.0 microcurie per gram Dose Equivalent I-131.
   a. If the specific activity of the reactor coolant is greater than 1.0 microcuries per gram Dose Equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 15.3.1-5, operation may continue for up to 48 hours. Reactor coolant sampling shall be in accordance with Table 15.4.1-2.
   b. If the specific activity of the reactor coolant is greater than 1.0 microcuries per gram Dose Equivalent I-131 for more than 48 hours during one continuous time interval or exceeds the allowable limit (above and to the right of the line) shown on Figure 15.3.1-5, the reactor will be shut down and the average reactor coolant temperature will be less than 500°F within 6 hours.
2. Less than or equal to $100/\bar{E}$ microcuries per gram.
   a. If the specific activity of the reactor coolant is greater than $100/\bar{E}$ microcuries per gram, the reactor will be shut down and the average reactor coolant temperature will be less than 500°F within 6 hours. Reactor coolant sampling shall be in accordance with Table 15.4.1-2.

Basis:

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 500 gpd in either steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative for Point Beach Nuclear Plant.

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown during Rated Power operation:

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

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

3. Reactor Coolant System raw measured Total Flow Rate
   ≥181,800 gpm. (See Basis).

Basis:

The reactor coolant system total flow rate of 181,800 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimeter at the beginning of each cycle.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.*

Time After Shutdown	Decay Heat % of Rated Power
1 min.	3.6
30 min.	1.55
1 hour	1.25
8 hours	0.7
48 hours	0.4

*Based on ANS 5.1-1979, "Decay Heat Power in Light-Water Reactors"

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safety system components in order to effect repairs.

Failure to complete safety injection system repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition. When the failures involve the residual heat removal system, in order to insure redundant means of decay heat removal, the reactor system may remain in a condition with reactor coolant temperatures between 500 and 350°F so that the reactor coolant loops and associated steam generators may be utilized for redundant decay heat removal. However, when the remaining RHR loop must be relied upon for redundant decay heat removal capability, reactor coolant temperatures shall be maintained between 350°F and 140°F.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.(2)

The operability of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core; (2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, spray additive tank, containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1); (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area greater than 3 ft^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO); and (4) long term subcriticality is maintained following a steamline break assuming ARI-1 and fuel failure is precluded.

The containment cooling function is provided by two independent systems: (a) fan coolers and (b) containment spray which, with sodium hydroxide addition, provides the iodine removal function. During normal power operation, only three of the four fan coolers are required to remove heat lost from equipment and piping within the containment.[3] In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: (1) four fan coolers, (2) two containment spray pumps, (3) two fan coolers plus one containment spray pump.[4] Sodium hydroxide addition via one spray pump reduces airborne iodine activity sufficiently to limit off-site doses to acceptable values. One of the four fan coolers is permitted to be inoperable for up to 48 hours during power operation.

The component cooling system is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. The component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit either following a loss-of-coolant accident, or during normal plant shutdown. If during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected.[5]

A total of six service water pumps are installed, only three of which are required to operate during the injection and recirculation phases of a postulated loss-of-coolant accident,[6] in one unit together with a hot shutdown condition in the other unit.

References

(1) FSAR Section 3.2.1

(2) FSAR Section 6.2

(3) FSAR Section 6.3.2

(4) FSAR Section 6.3

(5) FSAR Section 9.3.2

(6) FSAR Section 9.6.2

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:

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

$F_Q(Z) \leq 5.00 \times K(Z)$ for $P \leq 0.5$

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

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

An upper bound envelope of 2.50 times the normalized peaking factor axial dependence of Figure 15.3.10-3 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 (ΔI) 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 mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

Axial Power Distribution

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the $F_Q(Z)$ upper bound envelope of F_Q^{Limit} times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

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 at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and the thermal power is greater than 50 percent of Rated Power.

FIGURE 15.3.10-1

CONTROL BANK INSERTION LIMITS
POINT BEACH UNITS 1 AND 2

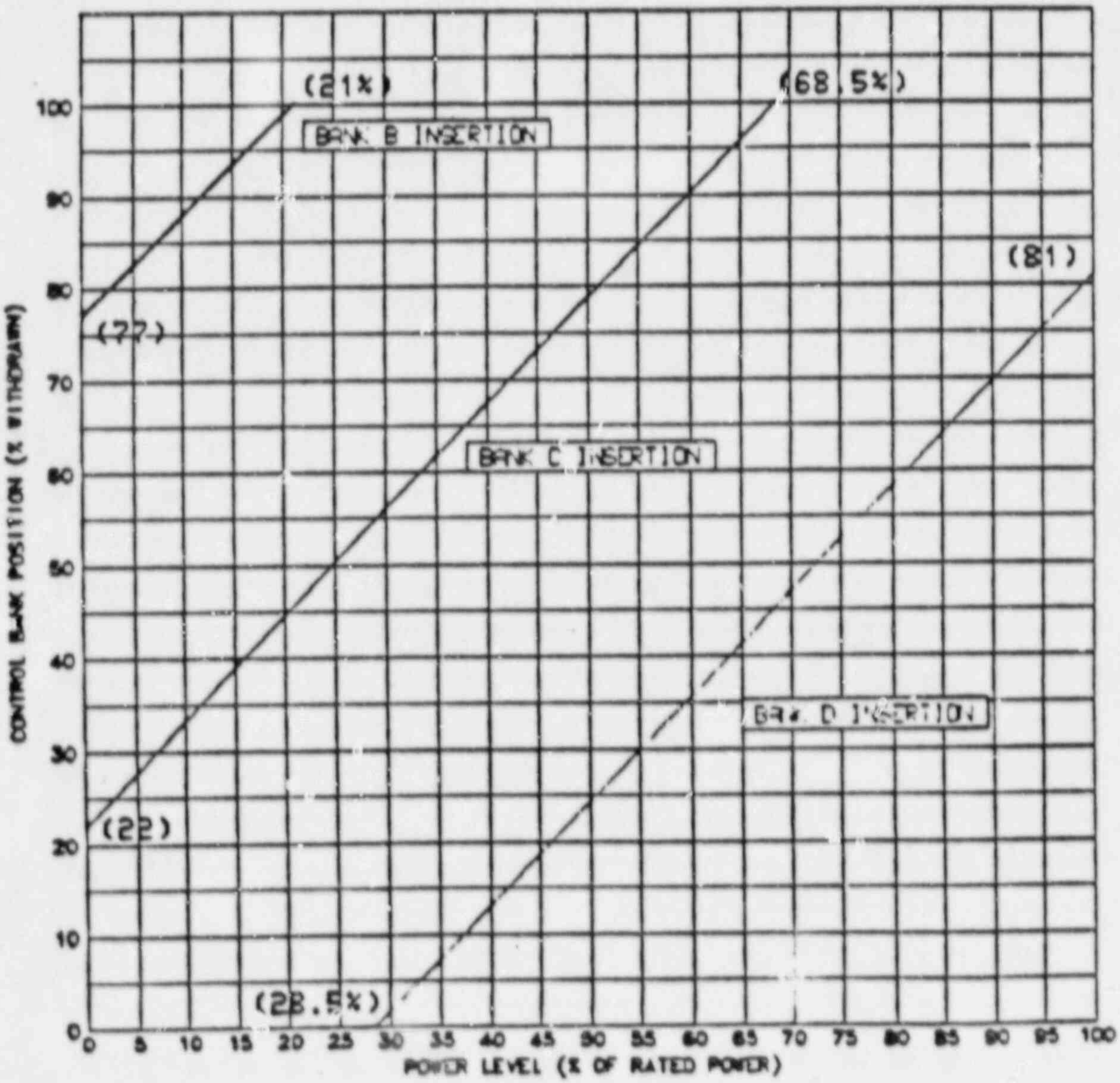


FIGURE 15.3.10-3

POINT BEACH UNITS 1 AND 2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

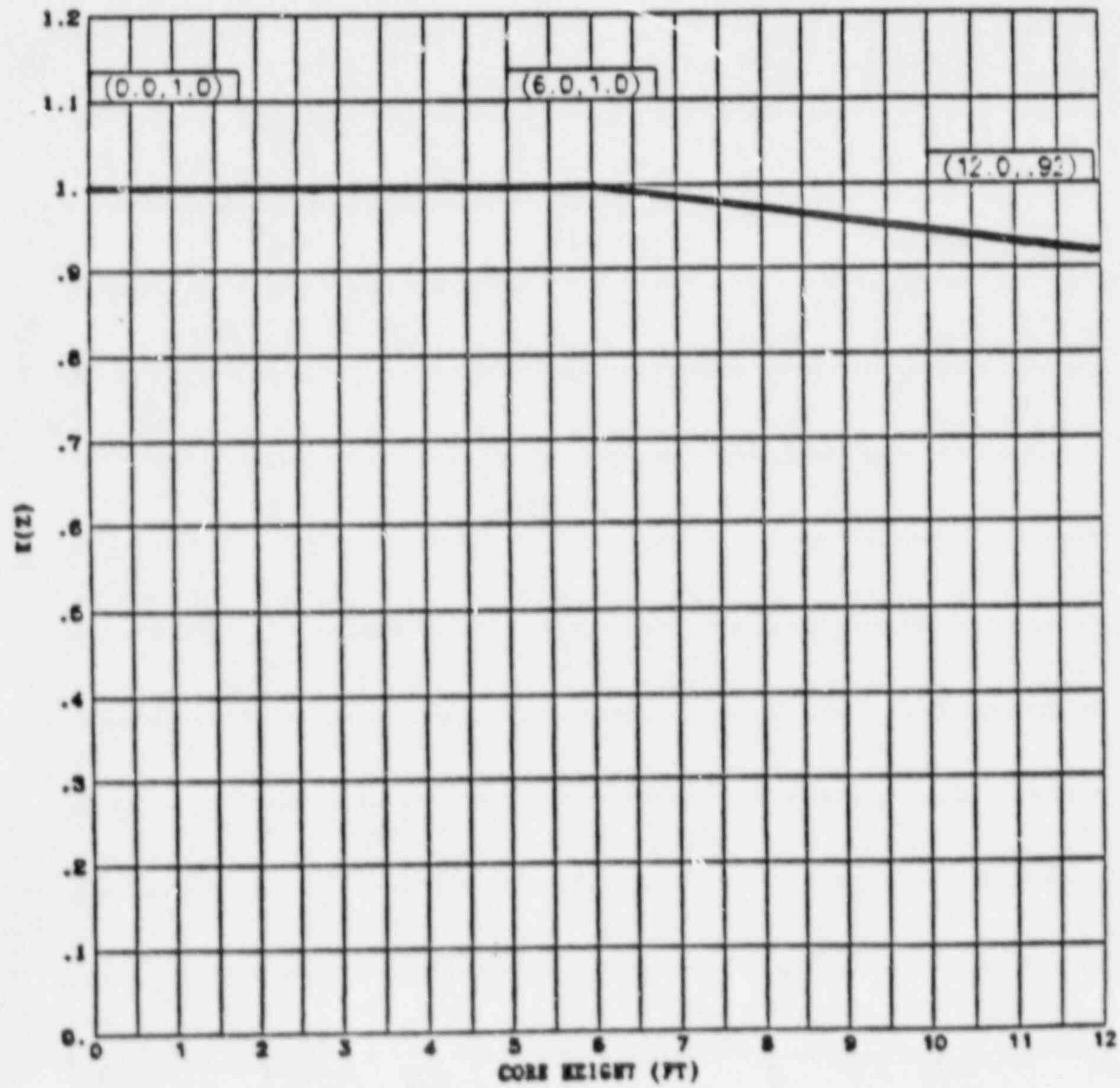
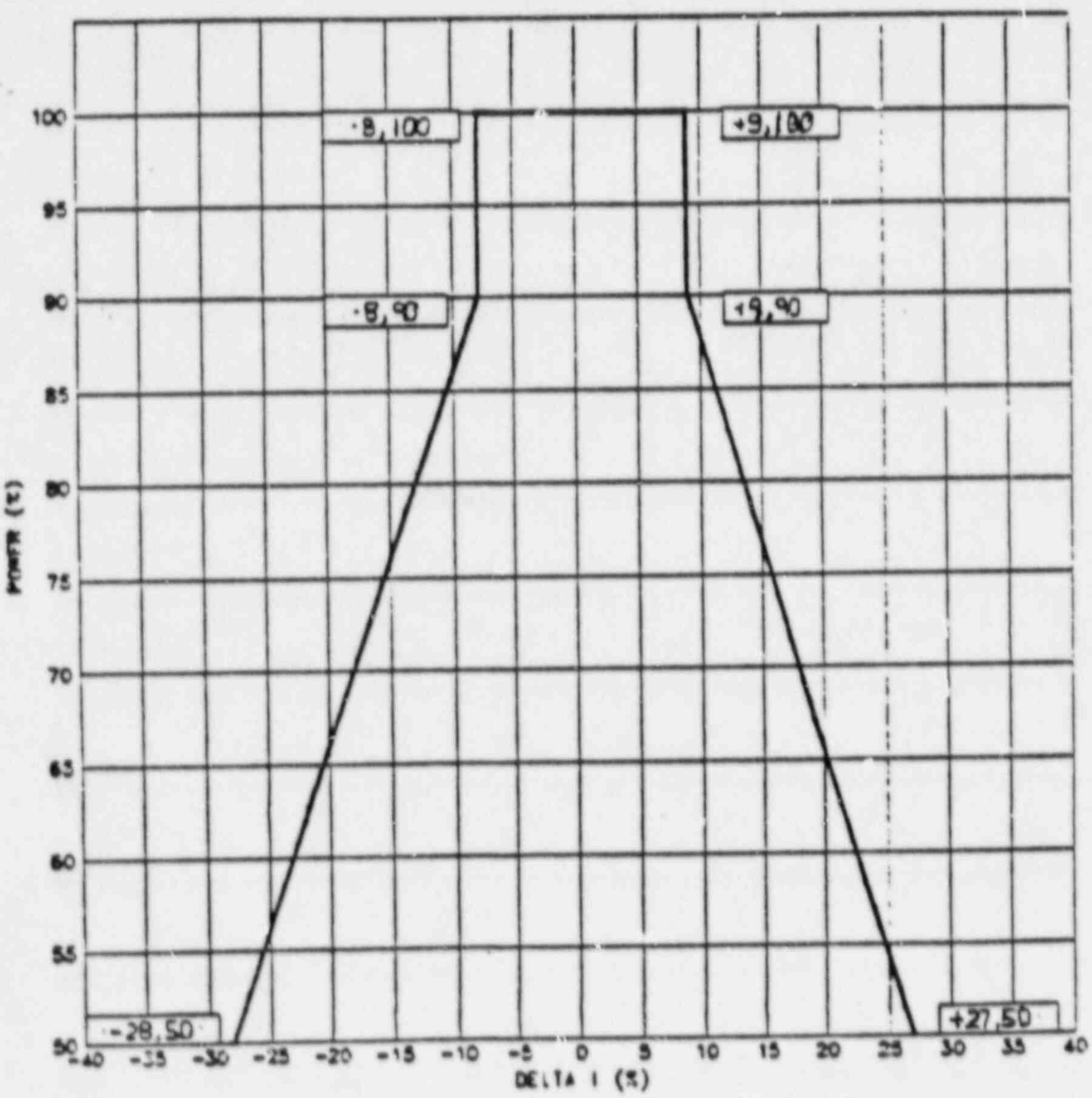


FIGURE 15.3.10-4

FLUX DIFFERENCE

OPERATING ENVELOPE

POINT BEACH UNITS 1 AND 2



15.5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Emergency Core Cooling Systems.

Objective

To define those design features which are essential in providing for safe system operation.

Specifications

A. Reactor Core

1. General

The uranium fuel is in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly nominally contains 179 fuel rods[(1)]. Where safety limits are not violated, limited substitutions of fuel rods by filler rods consisting of Zircaloy 4 or stainless steel, or by vacancies, may be made to replace damaged fuel rods if justified by cycle specific reload analysis.

2. Core

A reactor core is a core loading pattern containing any combination of 14x14 OFA and 14x14 upgraded OFA fuel assemblies. The core may also contain previously depleted 14x14 standard fuel assemblies. The use of previously depleted 14x14 standard fuel assemblies will be justified by a cycle specific reload analysis.

3. Burnable absorber and/or water displacer rods are incorporated for reactivity and/or power distribution control. The burnable absorber rods consist of borated pyrex glass clad with stainless steel[4]. The water displacer rods are empty burnable absorber rods containing no pyrex glass. Another type of burnable absorber may consist of a thin coating of zirconium diboride on the radial surface of selected fuel rod pellets.

4. There are 33 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain a 142-inch length of silver-indium-cadmium alloy clad with the stainless steel.

5. Neutron source assemblies are used to provide a required minimum count rate during startup operations. The core contains at least two such assemblies, each containing four source rodlets comprised of a mixture of antimony and beryllium.

B. <u>Reactor Coolant System</u>

1. The design of the Reactor Coolant System complies with the code requirements[6].

2. All high pressure piping, components of the Reactor Coolant System and their supporting structures are designed to Class I requirements, and have been designed to withstand:

Attachment 3

SIGNIFICANT HAZARDS EVALUATION

As required by 10 CFR 50.91(a), we have evaluated these proposed changes in accordance with the standards specified in 10 CFR 50.92 to determine if the proposed changes constitute a significant hazards consideration. A proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

1. Revised Safety Limits/Transition Core Elimination

1.1 Proposed Change

The first change involve replacement of the two sets of Transition Core DNB Safety Limits (one for optimized fuel assemblies and one for transition cores) with a single revised Reactor Core Safety Limits figure. The revised Safety Limits reflect the proposed upgraded core features including the increase in enthalpy rise hot channel factor (F-delta-H) from 1.58 to 1.70, the increase in core bypass flow due to the elimination of the thimble plugging devices, and certain minor upgrades to the fuel design. Consideration of a transition core has been eliminated, since the transition from standard fuel assemblies (STD) to optimized fuel assemblies (OFA) will be completed for both units by April 1989. The changes in TS 15.2.1.1, the basis of TS 15.2.1.1, and the basis of TS 15.2.3 eliminate the reference to the Transition Core Safety Limits, and eliminate the definition of a transition core. Figures 15.2.1-1 and 15.2.1-2 are replaced with the revised Figure 15.2.1-1. Finally, the change in TS 15.5.3.A.2 and the deletion of the current TS 15.5.3.A.3, TS 15.5.3.A.4, and TS 15.5.3.A.5 revise the description of a reactor core to reflect the elimination of transition core considerations.

1.2 Significant Hazards Evaluation

1.2.1 First Criterion

Replacement of the Transition Core and OFA Core Safety Limits with a single revised Reactor Core Safety Limits figure and the elimination of transition core references will not significantly increase the probability or consequences of an accident previously analyzed.

1.2.1.1 Probability

The revised Safety Limits were developed in the safety analyses to ensure acceptable DNB results for the higher F-delta-H limit and the higher core bypass flow due to thimble plugging device elimination. The limits are used to determine the acceptability of the consequences of certain design-basis events and as such have no effect on the probability of those events occuring. There will therefore be no significant increase in the probability of occurrence of any of the accidents previously analyzed using these Safety Limits.

1.2.1.2 Consequences

The design-basis accidents for which the core Safety Limits are used (Uncontrolled RCCA Withdrawal at Power, Reduction in Feedwater Enthalpy Incident, Excessive Load Increase Incident, and Loss of External Electrical Load) were reanalyzed assuming the revised Reactor Core DNB Safety Limits. The results show that the minimum DNBR value for each event is greater than the Safety Analysis Limit value, derived through the use of the Revised Thermal Design Procedure (RTDP) as described in WCAP-11397, "Revised Thermal Design Procedure." Through the RTDP, a design limit DNBR value is calculated. This is then increased to provide margin to the design limit; the increased value is the safety analysis limit value. Since the analysis of these accidents show that the calculated DNBR is greater than the safety analysis limit value, there will be no significant increase in the consequences of an accident previously evaluated.

Consideration of a transition core has been eliminated. The operational characteristics of the OFA and the upgraded OFA are similar. Consequently, core thermal-hydraulic parameters are not significantly changed in the transition to upgraded OFA fuel features and the revised Reactor Core Safety Limits apply to both fuel types. Since the previously-depleted standard fuel assemblies will be at lower relative powers than the fresh OFA, the limited reinsertion of standard fuel assemblies will be bounded by the revised Reactor Core Safety Limits, and transition core Safety Limits are no longer required. The continued use of previously-depleted standard fuel assemblies will be justified by a cycle-specific reload safety analysis in accordance with 10 CFR 50.59.

1.2.2 Second Criterion

The proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated. This change does not involve significant physical modifications to the Point Beach cores. Upgraded OFA adds the following modifications to the current OFA design and is bounded in the analyses: removable top nozzles, debris filter bottom nozzles, axial blankets, integral fuel burnable absorbers, and extended burnup geometry (shorter nozzles to allow for additional fuel rod growth and longer fuel rod plenums to accommodate additional fission gas production). These modifications are minor from a core parameters standpoint, and are summarized in WCAP-10444-P-A, "VANTAGE 5 Reference Core Report - VANTAGE 5 Fuel Assembly." The NRC generically approved the use of VANTAGE 5 assemblies in a Safety Evaluation Report dated July 1985. As stated above, the operational characteristics of the OFA and the upgraded OFA are similar. Consequently, core thermal-hydraulic parameters are not significantly changed in the transition to upgraded OFA fuel features. In addition to bounding these fuel modifications, the safety analyses bound the use or removal of fuel assembly thimble plugging devices. Removing the plugs has the effect of slightly increasing the core bypass flow. This increase, however, was input to the safety analyses with acceptable results.

The physical modifications associated with this specification change are not significant; therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

1.2.3 Third Criterion

The Core Safety Limits are developed using the WRB-1 DNB correlation and the safety analysis limit DNBR (as described above). The Core Safety Limit curves are designed to ensure that the safety analysis limit DNBR is met. Overtemperature Delta-T and Overpower Delta-T setpoints are then calculated based on the Core Safety Limits, and are used in the analysis of the four events listed above. With these setpoints, the Core Safety Limits, and therefore the safety analysis limit DNBR, will not be violated. Since margin is built into the safety analysis limit DNBR, meeting that limit ensures that sufficient margin exists. In the analyses of the four events assuming the upgraded core features, results show that the Core Safety Limits are not violated for any of these events. The safety analyses show a slight but insignificant reduction in the DNB margin. The DNB safety acceptance criterion is met with margin, and the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that these proposed Technical Specification changes do not involve a significant hazards consideration.

2. OTDT/OPDT SETPOINTS/BOUNDING PRESSURE

2.1 Proposed Change

This change involves a revision to the Overtemperature Delta-T (OTDT) and Overpower Delta-T (OPDT) setpoints to reflect the revised Reactor Core Safety Limits discussed above in change number 1. The setpoints are calculated to ensure the Core Safety Limits are not violated. This calculation was performed at a bounding pressure value to make the OTDT and OPDT setpoints the same for operation at both 2000 and 2250 psia. The changes in TS 15.2.3.1.B.4 and TS 15.2.3.1.B.5 reflect the revised setpoint inputs as a result of the revision in the Reactor Core Safety Limits.

2.2 Significant Hazards Evaluation

2.2.1 First Criterion

Revision of the OTDT and OPDT setpoints to reflect the revised Reactor Core DNB Safety Limits will not significantly increase the probability or consequences of an accident previously analyzed.

2.2.1.1 Probability

The OTDT and OPDT setpoints ensure that the core Safety Limits are not violated. The revised Safety Limits, and therefore the revised setpoints, were developed in the safety analyses to ensure acceptable DNB results for the higher F-delta-H limit and the higher core bypass flow due to thimble plugging device elimination. The limits are used to determine the acceptability of the consequences of certain design-basis events and as such have no effect on the probability of those events occurring. There will therefore be no significant increase in the probability of occurrence of any of the accidents previously analyzed using these Safety Limits.

2.2.1.2 Consequences

The design-basis accidents for which the OTDT and OPDT setpoints are used (Uncontrolled RCCA Withdrawal at Power, Reduction in Feedwater Enthalpy Incident, Excessive Load Increase Incident, and Loss of External Electrical Load) were reanalyzed assuming the revised setpoints based on the revised Reactor Core DNB Safety Limits discussed in change number 1. The results show that the minimum DNBR value for each event is greater than the Safety Analysis Limit value, derived through the use of the Revised Thermal Design Procedure (RTDP) as described in WCAP-11397, "Revised Thermal Design Procedure." Through the RTDP, a design limit DNBR value is calculated. This is then increased to provide margin to the design limit; the increased value is the safety analysis limit value. Since the analysis of these accidents show that the DNB safety acceptance criterion is met, there will be no significant increase in the consequences of an accident previously evaluated.

2.2.2 Second Criterion

The proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated. This change does not involve significant physical modifications to the Point Beach cores. Upgraded OFA adds the following modifications to the current OFA design and is bounded in the analyses: removable top nozzles, debris filter bottom nozzles, axial blankets, integral fuel burnable absorbers, and extended burnup geometry (shorter nozzles to allow for additional fuel rod growth and longer fuel rod plenums to accommodate additional fission gas production). These modifications are minor from a core parameters standpoint, and are summarized in WCAP-10444-P-A, "VANTAGE 5 Reference Core Report - VANTAGE 5 Fuel Assembly." The NRC has generically approved the use of VANTAGE 5 fuel assemblies in a Safety Evaluation Report dated July 1985. As stated above, the operational characteristics of the OFA and the upgraded OFA are similar. Consequently, core thermal-hydraulic parameters are not significantly changed in the transition to upgraded OFA fuel features. In addition to bounding these fuel modifications, the safety analyses bound the use or removal of fuel assembly thimble plugging devices. Removing the plugs has the effect of slightly increasing the core bypass flow. This increase, however, was input to the safety analyses with acceptable results.

The physical modifications associated with this specification change are not significant; therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

2.2.3 Third Criterion

The Core Safety Limits are developed using the WRB-1 DNB correlation and the safety analysis limit DNBR (as described above). The curves are designed to ensure that the safety analysis limit DNBR is met. OTDT and OPDT setpoints are then calculated based on the Core Safety Limits, and used in the analysis of the four events listed above. With these setpoints, the Core

Safety Limits and therefore the safety analysis limit DNBR will not be violated. Since margin is built into the safety analysis limit DNBR, meeting that limit ensures that margin exists. In the analyses of the four events assuming the upgraded core features, results show that the Core Safety Limits are not violated for any of these events. The safety analyses show a slight but insignificant reduction in the DNB margin. Therefore, the OTDT and OPDT setpoints are acceptable, the DNB safety acceptance criterion is met with margin, and the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

3. PRIMARY-TO-SECONDARY STEAM GENERATOR LEAKAGE

3.1 Proposed Change

This change involves a revision in the assumed steady-state primary-to-secondary steam generator leakage rate stated in the basis of TS 15.3.1.C. This will make this steady state value consistent with the primary-to-secondary steam generator leakage rate listed in TS 15.3.1.D.4 as a Limiting Condition for Operation for Leakage of Coolant.

3.2 Significant Hazards Evaluation

3.2.1 First Criterion

The revision to the assumed steady state primary-to-secondary steam generator leakage rate will not significantly increase the probability or consequences of an accident previously analyzed.

3.2.1.1 Probability

The steady state leakage rate is used as an input to the steam generator tube rupture analysis to determine the level of activity to be assumed at the start of the accident. The initial level of activity helps to determine only the consequences of the steam generator tube rupture event and as such has no effect on the probability of that event occurring. There will therefore be no significant increase in the probability of occurrence of any of the accidents previously analyzed using these Safety Limits.

3.2.1.2 Consequences

The design-basis accident for which the steady state primary-to-secondary steam generator leakage is used (Steam Generator Tube Rupture) was reanalyzed assuming the revised leakage rate. The results show that the doses calculated fall within a small fraction of the 10 CFR 100 exposure guidelines. That is, the doses are lower than 30 rem thyroid and 2.5 rem whole body, which are 10 percent of the 10 CFR 100 guideline values of 300 rem thyroid and 25 rem whole body. Since the analysis of this accident using the revised leakage rate shows that the doses calculated are within a small fraction of the exposure guidelines, there will be no significant increase in the consequences of an accident previously evaluated.

3.2.2 Second Criterion

The proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated. This change does not involve physical modifications to the Point Beach cores. It revises a primary-to-secondary leakage rate value in the basis of a specification to be consistent with that of another specification. This revision was input to the safety analysis of the steam generator tube rupture event with acceptable results. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3.2.3 Third Criterion

As stated above, the results of the analysis of steam generator tube rupture using the revised primary-to-secondary steam generator leak._: demonstrate that the doses calculated fall within a small fraction of the 10 CFR 100 exposure guideline values. Falling within a small fraction of the guidelines allows for a significant margin to the guideline values. Therefore, the exposure guidelines are met with margin, and the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

4. TAVG AND PRESSURE REVISION

4.1 Proposed Change

This change involves a revision to the reactor coolant system Tavg and pressure Operational Limitations in TS 15.3.1.G.1 and TS 15.3.1.G.2, respectively. The former is changed to better reflect the value used in the safety analyses. The latter is changed to reflect the location at which the pressure indication is taken. In addition, as described in change number 2, the change also eliminates the footnote to TS 15.3.1.G.2 to reflect the fact that the safety analyses were performed at a bounding pressure value. This eliminates the need to reanalyze any accidents at a future date to accommodate a change in operating pressure.

4.2 Significant Hazards Evaluation

4.2.1 First Criterion

The actual Tavg and pressure at which the plant is operated will not be changed. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. There will therefore be no significant increase in the probability or consequences of any of the accidents previously analyzed.

4.2.2 Second Criterion

This change does not involve any physical modifications to the Point Beach cores; it only involves a revision to the description of the Operational Limitations. Since there will be no physical modifications associated with this specification change, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.2.3 Third Criterion

The actual Tavg and pressure at which the plant is operated will not be changed. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. Therefore, the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

5. RWST DESCRIPTION ADDITION

5.1 Proposed Change

This change involves an addition to the basis of TS 15.3.3. This addition describes the basis of the RWST minimum volume and minimum boron concentration Limiting Conditions for Operation. While this change is not required to support the increased peaking factors and upgraded core features, it does make the basis of the Limiting Conditions for Operation more complete.

5.2 Significant Hazards Evaluation

5.2.1 First Criterion

No plant design or operational parameters will be affected by this change; it is only an addition of a description in the basis. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. There will therefore be no significant increase in the probability or consequences of any of the accidents previously analyzed.

5.2.2 Second Criterion

This changes does not involve any physical modifications to the Point Beach cores; it only involves an addition to the basis of Limiting Conditions for Operation. Since there will be no physical modifications associated with this specification change, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

5.2.3 Third Criterion

Plant design and operational parameters will not be changed by this addition to the basis of Limiting Conditions for Operations. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. Therefore, the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

6. F_Q Value Revision

6.1 Proposed Change

By meeting the F_Q limit it is ensured that cladding integrity and fuel melting at the "hot spot" are maintained within the applicable safety analysis limits in the event of an accident. This change to TS 15.3.10.B.1.a involves an increase in the F_Q limit from 2.21 to 2.50 to increase the flexibility of the reload core design. This evaluation assumes acceptable results from the Upper Plenum Injection (UPI) Large-Break LOCA reanalysis project scheduled for completion in October 1988.

6.2 Significant Hazards Evaluation

6.2.1 First Criterion

Revision of the F_Q limit will not significantly increase the probability or consequences of an accident previously analyzed.

6.2.1.1 Probability

The F_Q limit is used as an input to the Locked Rotor, Rod Ejection, and LOCA safety analysis to help determine the extent of the consequences of these design-basis events and as such has no effect on the probability of those events occurring. There will therefore be no significant increase in the probability of occurrence of any of the accidents previously analyzed using this limit.

6.2.1.2 Consequences

The design-basis accidents for which the F_Q limit is used (Locked Rotor, Rod Ejection, Small-Break LOCA) were reanalyzed assuming the revised limit. The results of the non-LOCA analyses show that all safety criteria are met. The small-break LOCA analysis results demonstrate that the Emergency Core Cooling System (ECCS) satisfies the acceptance criteria of 10 CFR 50.46. Since the analysis of these accidents show acceptable results using the revised F_Q limit, there will be no significant increase in the consequences of an accident previously evaluated. This assumes acceptable results from the UPI project.

6.2.2 Second Criterion

The increase in the F_Q limit to provide additional flexibility in the reload core design process does not involve any physical modifications to the Point Beach cores. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

6.2.3 Third Criterion

The safety analysis limits (maximum RCS pressure, cladding temperature, fuel stored energy, fuel melt, and Zr-water reaction) for the non-LOCA events are set to ensure that cladding integrity and fuel melting at the "hot spot" are minimized. Analyses of the locked rotor and rod ejection events demonstrate that these safety analysis limits are met using the higher F_Q limit. There will therefore be no significant reduction in a margin of safety for these non-LOCA events.

The margin of safety for LOCA events is established in the LOCA analysis acceptance criteria of 10 CFR 50.46. Reanalysis of the small-break LOCA event demonstrates that the emergency core cooling system satisfies the acceptance criteria. There will therefore be no signficant reduction in a margin of safety for a small-break LOCA.

Since the safety analysis limits and acceptance criteria are met in both the non-LOCA and small-break LOCA cases, this Technical Specification change does not involve a significant reduction in a margin of safety. As mentioned above, this conclusion assumes acceptable results from the UPI Large-Break LOCA reanalysis project scheduled for completion in October 1988.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

7. REVISED F-DELTA-H LIMIT

7.1 Proposed Change

This change involves the revision of the F-Delta-H limit stated in TS 15.3.10.B.1.a. The limit is increased from 1.58 to 1.70 for additional flexibility in the reload core design process. An increase in the F-Delta-H limit does not directly affect the system transient response of the non-LOCA events. Rather, the F-Delta-H limit is used in the determination of the DNBR for those events for which DNB is the safety acceptance criterion. DNBR calculations fall into two categories: 1) those events in which F-Delta-H is indirectly accounted for via the core limits, and 2) those events which directly assume the F-Delta-H in the analysis. The former of these categories was addressed in Significant Hazards Evaluation 1 and therefore only the second category will be addressed here. In addition, the effect of the F-Delta-H change on the LOCA analyses must also be addressed. This evaluation assumes acceptable results from the UPI Large-Break LOCA reanalysis project scheduled for completion in October 1988.

7.2 Significant Hazards Evaluation

7.2.1 First Criterion

The increase in F-Delta-H from 1.58 to 1.70 will not significantly increase the probability or consequences of an accident previously analyzed.

7.2.1.1 Probability

In the safety analyses which directly assume the revised F-Delta-H limit, the change in the F-Delta-H limit serves only to change the consequences of the events and has no effect on the probability of occurrence of any of the accidents previously analyzed using the higher F-Delta-H limits.

7.1.2 Consequences

The non-LOCA accident analyses which directly assume the revised F-Delta-H limit (Rod Withdrawal from Subcritical, Dropped Rod, Startup of an Inactive Loop, and Loss of Flow) were reanalyzed assuming the revised limit. The results show that the minimum DNBR value for each event is greater than the safety analysis limit DNBR value, derived through the use of the Revised Thermal Design Procedure (RTDP) as described in WCAP-11397, "Revised Thermal Design Procedure." Through the RTDP, a design limit DNBR value is calculated. This is then increased to provide margin to the design limit; the increased value is the safety analysis limit value. The analyses of these accidents show that the DNB safety acceptance criterion is met. The small-break LOCA analysis results demonstrate that the Emergency Core Cooling System (ECCS) still satisfies the acceptance criteria of 10 CFR 50.46 assuming the revised F-Delta-H limit. Since the safety acceptance criteria are met for all these analyses, there will be no significant increase in the consequences of an accident previously evaluated. This assumes acceptable results from the UPI project.

7.2.2 Second Criterion

The increase in the F-Delta-H limit provides additional flexibility in the reload core design process and does not involve any physical modifications to the Point Beach cores. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

7.2.3 Third Criterion

Analysis of the non-LOCA events which directly assume the revised F-Delta-H limit demonstrates that the safety analysis limit DNBR is met in each case. Since margin is built into the safety analysis limit DNBR, meeting that limit ensures that margin exists. The margin of safety for LOCA events is established in the LOCA analysis acceptance criteria of 10 CFR 50.46. Reanalysis of the small-break LOCA event assuming the revised F-Delta-H limit demonstrates that the emergency core cooling system satisfies the acceptance criteria. There will therefore be no significant reduction in a margin of safety for a small-break LOCA.

The safety analyses show a slight but insignificant reduction in the DNB margin. Since the safety analysis limits and acceptance criteria are met in both the non-LOCA and small-break LOCA cases, this Technical Specification change does not involve a significant reduction in a margin of safety. Again, this conclusion assumes acceptable results from the Upper Plenum Injection Large-Break LOCA reanalysis project scheduled for completion in October 1988. We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

8. <u>POWER DISTRIBUTION DESCRIPTION</u>

8.1 Proposed Change

This change involves an addition to the basis of TS 15.3.10. This addition aids in the clarification of the basis of the Hot Channel Factor Normalized

Operating Envelope and its use in the safety analyses. In addition, the F-Delta-H and F_Q values listed in the basis of TS-15.3.10 are revised to reflect their increased values. These revisions were discussed in Significant Hazards Evaluations 6 and 7, respectively.

8.2 Significant Hazards Evaluation

8.2.1 First Criterion

No plant design or operational parameters will be affected by this change; it is only an addition of a description in the basis. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. There will therefore be no significant increase in the probability or consequences of any of the accidents previously analyzed.

8.2.2 Second Criterion

This change does not involve any physical modifications to the Point Beach cores; it only involves an addition to the basis of Limiting Conditions for Operation. Since there will be no physical modifications associated with this specification change, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

8.2.3 Third Criterion

Plant design and operational parameters will not be changed by this addition to the basis of Limiting Conditions for Operations. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. Therefore, this Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

9. CONTROL BANK INSERTION LIMIT REVISION

9.1 Proposed Change

This change involves a revision to the Control Bank Insertion Limits. Figure 15.3.10-1 of TS 15.3.10 is revised to obtain a wider delta-I band and to ensure that the delta-I band will be conservative for future fuel cycles using upgraded core features. The revised insertion limits require a shallower insertion at any given power level. This increases the allowable power level at the top of the core, which results in a larger allowable positive axial offset. The delta-I band is therefore wider and will bound future cycles. This evaluation assumes acceptable results from the Upper Plenum Injection (UPI) Large-Break LOCA reanalysis project scheduled for completion in October 1988.

9.2 Significant Hazards Evaluation

9.2.1 First Criterion

Revision of the Control Bank Insertion limits will not significantly increase the probability or consequences of an accident previously analyzed.

9.2.1.1 Probability

The Control Bank Insertion Limits are revised to obtain a revised operating delta-I band, which are input to the safety analyses. The limits are used to provide for achieving hot shutdown by reactor trip at any time. Additionally, they help determine the consequences of certain design-basis events. The Control Bank Insertion Limits have no effect on the probability of those events occurring. There will therefore be no significant increase in the probability of occurrence of any of the accidents previously analyzed using these limits.

9.2.1.2 Consequences

The revised Control Bank Insertion Limits can impact the non-LOCA safety analyses in the following areas: shutdown margin, trip reactivity, power distribution limits, ejected and dropped rod worths, post-ejected rod peaking factors, and differential rod worths. The revised limits were encompassed in the nuclear design calculations and in the safety analyses. The results demonstrate that the acceptance criteria for each event were met. The small-break LOCA analysis results demonstrate that the Emergency Core Cooling System (ECCS) satisfies the acceptance criteria of 10 CFR 50.46. Since the analyses of these accidents show that the acceptance criteria are met, there will be no significant increase in the consequences of an accident previously evaluated. This assumes acceptable results from the UPI project.

9.2.2 Second Criterion

The revision to the Control Bank insertion Limits to obtain a wider delta-I band does not involve any physical modifications to the Point Beach cores. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

9.2.3 Third Criterion

Analysis of the non-LOCA events assuming the revised Control Bank Insertion Limits demonstrates that the safety acceptance criteria are met in each case. Since margin is built into these criteria to preclude unacceptable consequences, meeting those criteria ensures that margin exists. The margin of safety for LOCA events is established in the LOCA analysis acceptance criteria of 10 CFR 50.46. Reanalysis of the small-break LOCA event assuming the revised insertion limits demonstrates that the emergency core cooling system satisfies the acceptance criteria. There will therefore be no significant reduction in a margin of safety for a small-break LOCA.

Since the safety analysis limits and acceptance criteria are met in both the non-LOCA and small-break LOCA cases, this Technical Specification change does not involve a significant reduction in a margin of safety. Again, this conclusion assumes acceptable results from the Upper Plenum Injection Large-Break LOCA reanalysis project scheduled for completion in October 1988.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

10. HOT CHANNEL FACTOR NORMALIZED AXIAL OPERATING ENVELOPE REVISION

10.1 Proposed Change

This change involves a revision to the Hot Channel Factor Normalized Axial Operating Envelope (K(z) curve), which is presented in Figure 15.3.10-3 of TS 15.3.10. K(z) is used as a multiplier in the axial height dependent equation for $F_Q(z)$ in TS 15.3.10.B.1.a. The third line segment of this curve is eliminated to increase the flexibility of the reload core design and operation by allowing operation at a higher power level at the top of the core. The remaining two line segments of the K(z) curve remain unchanged.

10.2 Significant Hazards Evaluation

10.2.1 First Criterion

Elimination of the third line segment of the K(z) curve will not significantly increase the probability or consequences of an accident previously analyzed.

10.2.1.1 Probability

The K(z) curve is used as a multiplier for the F_Q limit, which is used as input to the Locked Rotor, Rod Ejection, and LOCA safety analyses to help determine the extent of the consequences of these design-basis events and as such has no effect on the probability of those events occuring. There will therefore be no significant increase in the probability of occurence of any of the accidents previously analyzed using this limit.

10.2.1.2 Consequences

Since the K(z) curve is used in the determination of the axial height dependent F_Q limit, its effects are implicit to the discussion of the F_Q limit in significant hazards evaluation number 6. The conclusions drawn there from the analysis of the F_Q limit are equally applicable to the revised K(z) curve. Therefore, there will be no significant increase in the consequences of an accident previously evaluated.

10.2.2 Second Criterion

The revision to the K(z) curve to provide additional flexibility in the reload core design process does not involve any physical modifications to the Point Beach cores. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

10.2.3 Third Criterion

As stated in section 10.2.1.2, analysis of the revised K(z) curve is implicit to the analysis of the F_Q limit discussed in significant hazards evaluation number 6. The conclusions drawn there from the analysis of the F_Q limit are equally applicable to the revised K(z) curve. Therefore, there will be no significant reduction in a margin of safety as a result of the revision of the K(z) curve.

We conclude by this evaluation that the proposed specification change does not involve a significant hazards consideration.

11. FLUX DIFFERENCE OPERATING ENVELOPE REVISION

11.1 Proposed Change

This change involves a revision to the Flux Difference Operating Envelope (delta-I band), Figure 15.3.10-4 of TS 15.3.10. The revision to this figure reflects the use of upgraded core features and the revised Control Bank Insertion Limits. The ways in which these affect the Flux Difference Operating Envelope are explained in Significant Hazards Evaluation section 9.1.

11.2 Significant Hazards Evaluation

11.2.1 First Criterion

Revision of the Flux Difference Operating Envelope will not significantly increase the probability or consequences of an accident previously analyzed.

11.2.1.1 Probability

The revised Envelope was developed in the safety analyses to accommodate the upgraded core features. Adherence to the Envelope ensures that the power distribution limits assumed in the safety analyses are met. The Envelope is used to limit the consequences of certain design-basis events and as such has no effect on the probabilty of those events occuring. There will therefore be no significant increase in the probability of occurrence of any of the accidents previously analyzed using these Safety Limits.

11.2.1.2 Consequences

Nuclear design calculations were performed assuming the revised Flux Difference Operating Envelope. Results show that adherence to the Envelope during operation ensures that the power distribution limits assumed in the safety analyses are met. Since those power distribution limits have been shown to be acceptable in Significant Hazards Evaluations 6, 7, 9, and 10, there will be no significant increase in the consequences of an accident previously evaluated.

11.2.2 Second Criterion

The revision to the Flux Difference Operating Envelope does not involve any physical modifications to the Point Beach cores. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

11.2.3 Third Criterion

As described above, nuclear design calculations were performed assuming the revised Flux Difference Operating Envelope. Results show that adherence to the Envelope during operation ensures that the power distribution limits assumed in the safety analyses are met. The revision to the Envelope only reflects the revision to the power distribution limits, and in itself cannot involve a significant reduction in a margin of safety. Since the power distribution limits have been shown to be acceptable in Significant Hazards Evaluations 6, 7, 9, and 10, the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

12. IFBA DESCRIPTION ADDITION

12.1 Proposed Change

This change involves the addition of a description to the Reactor Design Features section of the Technical Specifications, TS 15.5.3.A. The description of burnable absorbers is updated to include a description of integral fuel burnable absorbers (IFBAs), one of the upgraded core features. An IFBA is a thin coating of zirconium diboride on the radial surface of selected fuel rod pellets. The use of IFBAs is addressed in WCAP-10444-P-A, "VANTAGE 5 Reference Core Report - VANTAGE 5 Fuel Assembly." The NRC generically approved the use of VANTAGE 5 assemblies in a Safety Evaluation report dated July 1985.

12.2 Significant Hazards Evaluation

12.2.1 First Criterion

The use of IFBAs will not significantly increase the probability or consequences of an accident previously analyzed.

12.2.1.1 Probability

Consideration of IFBAs is included in the nuclear design calculations performed to support the safety analyses. As input to the analyses, the effects of IFBA use are used to help determine the consequences of an accident, and as such have no effect on the probability of that event occuring. There will therefore be no significant increase in the probability of occurence of any of the accidents previously analyzed.

12.2.1.2 Consequences

The use of IFBAs is assumed in the safety analyses, and is reflected in Significant Hazards Evaluations 1, 2, and 9. Since their use is found in those evaluations to be bounded by the input assumed for the safety analyses, the conclusion drawn in those evaluations that there will be no significant increase in the consequences of an accident previously evaluated is applicable here.

12.2.2 Second Criterion

The proposed change does not create the possibility for a new or different kind of accident from any accident previously evaluated. This change does not involve significant physical modifications to the Point Beach cores. The use of IFBAs is minor from a core parameters standpoint, and is summarized in

WCAP-10444-P-A, "VANTAGE 5 Reference Core Report - VANTAGE 4 Fuel Assembly," which the NRC generically approved for use in a Safety Evaluation Report dated July 1985. The physical modifications associated with this specification change are not significant; therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

12.2.3 Third Criterion

As stated above, the use of IFBAs is covered in Significant Hazards Evaluations 1, 2, and 9. Since those evaluations found the use of IFBAs to be acceptable, this Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.

13. WATER DISPLACER/NEUTRON SOURCE DESCRIPTION ADDITION

13.1 Proposed Change

This change involves a revision to TS 15.5.3.A.6 and an addition to TS 15.5.3.A. The revision expands the description of burnable absorber rods to include a description of water displacer rods. These rods, which are essentially burnable absorber rods without the burnable absorber, are used for reactivity and power distribution control. Their use was originally evaluated by Westinghouse in the Reload Safety Evaluation for Point Beach Nuclear Plant, Unit 1 Cycle 14, January 1986. The description of neutron source assemblies is added to complete the description of the reactor core components.

13.2 Significant Hazards Evaluation

13.2.1 First Criterion

No plant design or operational parameters will be affected by this change; it is only an addition to the Reactor Core description in the Design Features section. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. There will therefore be no significant increase in the probability or consequences of any of the accidents previously analyzed.

13.2.2 Second Criterion

This change does not involve any physical modifications to the Point Beach cores; it only involves an addition to the description of the Reactor Core in the Design Features section. Since there will be no physical modifications associated with this specification change, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

13.2.3 Third Criterion

Plant design and operational parameters will not be changed by this addition to the Reactor Core description in the Design Features section. The input to the safety analyses will not be affected, so the results of those analyses will not be affected. Therefore, the Technical Specification change does not involve a significant reduction in a margin of safety.

We conclude by this evaluation that this proposed Technical Specification change does not involve a significant hazards consideration.