ATTACHMENT 1

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

April 15, 1998

Proposed Amendment 152

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

Introduction

In Reference 1, Wisconsin Public Service Corporation(WPSC) docketed changes related to the feed fuel (KEW-19) for the cycle 23 reload. Operation of Cycle 23 is anticipated to begin on/or about November 25, 1998. A Technical Specification amendment was included in the Reference 1 submittal schedule. The revisions in this proposed Technical Specification amendment are considered an integral part of the WPSC fuel and reload change plan for cycle 23. The revisions implement the improvements realized by the new fuel design (e.g., improved Departure from Nucleate Boiling Ratio (DNBR) and thermal performance due to the High Thermal Performance spacer (see Ref.2)) as well as reflect the changing Kewaunee plant conditions (e.g., the reduction in the minimum Reactor Coolant System flow due to increased steam generator tube plugging). The proposed changes in this amendment and the corresponding assessments are applicable to cycle 23 and subsequent cycles of operation that are bounded by the conditions of the assessment.

The proposed Technical Specification changes support cycle 23 fuel and reload changes and ensure consistency in fuel and reload design, safety analyses, and the technical specification operating limits. A description of each of the proposed changes follows:

Description of Proposed Changes to Technical Specification (TS) 2.1, "Safety Limits, Reactor Core" and Technical Specification (TS) 3.10, "Control Rod and Power Distribution Limits"

TS Section 2.1 is being revised as follows:

1) Figure 2.1-1 is revised to reflect the recently approved High Thermal Performance (HTP) Critical Heat Flux (CHF) correlation and corresponding Departure from Nucleate Boiling Ratio (DNBR) limit of 1.14. The figure also reflects changes in peak rod power and minimum reactor coolant flow.

2) Paragraph TS 3.10.b - new hot channel factors were incorporated for the new fuel design and the corresponding increase in peaking factors. The limits for $F^N_Q(Z)$ are specified in TS 3.10.b.1 and the limits for $F^N_{\Delta H}(Z)$ are specified in 3.10.b.2.

3) Paragraph TS 3.10.k - the specification for the maximum Reactor Coolant System (RCS) Inlet Temperature is being replaced with a specification for the maximum Reactor Coolant System (RCS) Average Temperature. The RCS Average Temperature is limited to 568.8°F.

4) Paragraph TS 3.10.l - the statement "During 100% steady-state power operation" is revised in the specification for minimum Reactor Coolant System (RCS) pressure and replaced with "During steady-state power operation."

5) Paragraph TS 3.10.m - the minimum Reactor Coolant Flow is being decreased to 85,500 gallons per minute per loop.

6) Paragraph TS 3.10.n - this specification was changed to reflect the approved Minimum DNBR limit.

7) Figure TS 3.10-1 - the Required Shutdown Reactivity vs. Boron Concentration was revised to reflect the change to an 18 month fuel cycle.

8) Figure TS 3.10-2, the Hot Channel Factor Normalized Operating Envelope was revised to reflect the values used in the safety analyses.

9) The Table of Contents and the Basis sections are being revised accordingly and are being submitted for information.

Safety Evaluation for Proposed Change to Figure TS 2.1-1 and TS B2.1

The Departure from Nucleate Boiling (DNB) correlation and the associated Minimum DNBR limit were changed in this section. Additionally, the minimum reactor coolant flow and corresponding peaking factor limits were changed. The new safety limits curves are a result of changes to DNBR limit, Critical Heat Flux (CHF) correlation, Reactor Coolant System (RCS) flow, peaking factors and fuel design. The new curves will bound operation with Siemens Power Corporation standard or heavy fuel. A safety evaluation for the change to the CHF correlation was previously performed and approved (Reference 2).

Operation within the limits of the curve is not an accident initiator nor will it increase the probability of an accident previously evaluated or introduce a new type of accident.

The change will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. Therefore, the change does not increase the probability of occurrence or increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

Significant Hazards Determination for Proposed Change to Figure TS 2.1-1 and TS B2.1

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The safety limits curves are not accident initiators. Therefore, the change will not increase the probability of an accident previously evaluated.

The proposed changes to the safety limits curves do not alter the plant configuration, operating set points, or overall plant performance. The safety limits curves reflect the changes to the DNBR limit, CHF correlation, RCS flow peaking factors and fuel design. The significant hazards determinations for these parameters are evaluated later in this submittal. Therefore, the change will not increase the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes in the safety limits curves do not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3. Involve a significant reduction in the margin of safety.

   Operation in the acceptable regions (i.e., below and to the left of the safety limit curves) in combination with the reactor protection and engineered safety systems designed into the plant will ensure that the safety limits are not exceeded during normal operation or during anticipated design basis operational transients. The core will be operated in the nucleate boiling heat transfer regime. Departure from nucleate boiling (DNB) will not occur and therefore fuel cladding integrity will be assured.

   The revised safety limit curves have been developed using operating parameters at their bounding values (e.g., rod powers at the peaking factor limits, reactor coolant flow at the minimum operating limit). The revised curves will bound plant operation with Siemens Power Corporation standard or heavy fuel. Therefore, this change will not involve a significant reduction in safety margin.

Safety Evaluation for Proposed Change to TS 3.10.b

The section of the Technical Specification for Power Distribution Limits was revised. Additional core peaking factor limits are incorporated into this section for Siemens Power Corporation heavy fuel. The reason for the change is to provide assurance that the reactor is operated within the assumptions of the transient and safety analyses. The core peaking factor limits were increased to 1.70 $F^{N}_{\Delta H}(Z)$ and 2.35 $F^{N}_{Q}(Z)$ (100% power limits) for Siemens Power Corporation heavy fuel.

Additionally, the format of the specification was changed to be consistent with the standardized Technical Specifications. The corrective actions for F_Q^{EQ} (Z) were moved from 3.10.b.6 to 3.10.b.4.A.

The core peaking factors are not accident initiators. Changing the Technical Specifications to be consistent with the safety analyses will not increase the probability of an accident previously evaluated or introduce a new type of accident.

The design basis safety analyses, the Large and Small Break LOCA accidents (Attachment 4) and the non-LOCA accidents (Attachment 5), have been analyzed and/or evaluated at the revised peaking factors. The re-analysis and evaluation have demonstrated that all safety analysis acceptance criteria are satisfied at the specified peaking factor limits. Therefore, the change will not significantly increase the consequences of an accident previously evaluated.

The change will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. Therefore, the change does not increase the probability of occurrence or increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

Significant Hazards Determination for Proposed Change to TS 3.10.b

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to determine that no significant hazards exist. The proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Peaking factor limits are input assumptions to the safety analyses and are not accident initiators. Therefore, this change will not increase the probability of occurrence of an accident previously evaluated.

The safety analyses input assumptions are designed to bound actual plant operation. Changing the safety analysis input assumption for the increased peaking factor limits does not change the underlying progression of design basis accidents evaluated in the safety analyses. All safety analysis acceptance criteria are satisfied in the increased peaking factor limit conditions (Attachments 4 and 5). Additionally, the radiological consequences are bounded by existing analysis at the increased peaking factor limits (Attachment 6). Therefore, this change will not significantly increase the consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   This change incorporates the safety analyses assumptions for core peaking factor limits for Siemens Power Corporation heavy fuel. The change does not alter plant equipment, set points or plant performance. Therefore, changing the peaking factor limits for analysis purposes will not create a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in the margin of safety.

   Results of the safety analyses performed to support this change are provided in Attachments 4 and 5. The radiological consequences are assessed to be bounded by existing analysis (Attachment 6). As shown, all safety analyses acceptance criteria are satisfied at the changed condition. In addition, the peaking factor limits assumed in the safety analyses are consistent with the proposed revised limits and these revised limits are established to bound actual plant operation. Therefore, this change will not involve a significant reduction in the margin of safety.

Safety Evaluation for Proposed Change to TS 3.10.k

The section of the Technical Specifications for reactor coolant temperature was changed. Current Technical Specifications limit reactor coolant inlet temperature at 100% power. The revised specification limits the maximum RCS average temperature and is consistent with the standardized Technical Specifications. This change provides assurance that the reactor is operated within the assumptions of the transient and safety analyses.

The RCS average temperature is not an accident initiator. Changing the technical specification limit consistent with the safety analyses will not increase the probability of an accident previously evaluated or introduce a new type of accident.

The design basis safety analyses have been re-analyzed and/or evaluated at the RCS average temperature. The re-analysis and evaluation have demonstrated that all safety analyses acceptance criteria are satisfied at the specified temperature. Therefore, the change will not increase the consequences of an accident previously evaluated.

The change will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. Therefore, the change does not increase the probability of occurrence or increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

Significant Hazards Determination for Proposed Change to TS 3.10.k

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The RCS average temperature limit is not an accident initiator. Changing the technical specification limit consistent with the accident analyses will not increase the probability of an accident previously evaluated.

   The proposed change limits the maximum reactor coolant system average temperature to 568.8°F. The design basis safety analyses, the Large and Small Break LOCA accidents (Attachment 4) and the non-LOCA accidents (Attachment 5), have been analyzed and/or evaluated consistent with the revised RCS average temperature. The re-analysis and evaluation have demonstrated that all safety analysis acceptance criteria are satisfied at the specified temperature. Therefore, the change will not increase the consequences of an accident previously evaluated.

The proposed technical specification limit for maximum allowed RCS average temperature was decreased below the analytical limit to account for instrument error.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change is consistent with the safety analyses. All safety analyses acceptance criteria are satisfied at the revised reactor coolant system average temperature. The TS limit will bound actual plant operation. Therefore, there is no significant reduction in the margin of safety.

Safety Evaluation for Proposed Change to TS 3.10.1

This section of the Technical Specifications for reactor coolant pressure was changed. The previous limit specified 100% power steady-state operation. The "100%" power value was removed to provide assurance that the reactor is operated within the assumptions of the transient and safety analyses at all power levels.

This change is administrative and is more restrictive. The limit for RCS pressure is not changed in the analysis or in the Technical Specifications. Therefore, the change will not affect the health and safety of the public.

Significant Hazards Determination for Proposed Change to TS 3.10.1

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The RCS pressure limit is not an accident initiator. By removing the 100% value from the specification, the assumptions in the safety analyses are not changed. Changing the technical specification to remove the 100% power criteria will not increase the probability of an accident previously evaluated.

   The design basis safety analyses have been analyzed and/or evaluated at the specified RCS pressure. The analyses and evaluations have demonstrated that all safety analyses acceptance criterion are satisfied at this pressure. Therefore, the change will not increase the consequences of an accident previously evaluated.

   The proposed technical specification limit for minimum allowed RCS pressure was increased above the analytical limit to account for instrument error.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change is consistent with the safety analyses. All safety analyses acceptance criteria are satisfied at the reactor coolant system pressure. The limit will bound actual plant operation. Therefore, there is no significant reduction in the margin of safety.

Safety Evaluation for Proposed Change to TS 3.10.m

This section of the Technical Specifications for reactor coolant flow was changed. The revised specification reduces the minimum RCS flow limit to accommodate increased steam generator plugging. The reason for this change is to provide assurance that the reactor is safely operated within the assumptions of the transient and safety analyses.

The RCS flow limit is not an accident initiator. Changing the technical specification limit consistent with the safety analyses will not increase the probability of an accident previously evaluated or introduce an accident .

The design basis safety analyses, the Large and Small Break LOCA accidents (Attachment 4) and the non-LOCA accidents (Attachment 5), have been re-analyzed and/or evaluated at the revised RCS flow. The re-analysis and evaluation have demonstrated that all safety analysis acceptance criteria are satisfied at the specified flow. Therefore, the change will not increase the consequences of an accident previously evaluated.

The change will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. Therefore, the change does not increase the probability of occurrence or increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The proposed change does not alter the plant configuration or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

Significant Hazards Determination for Proposed Change to TS 3.10.m

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The RCS flow limit is not an accident initiator. Changing the technical specification limit consistent with the accident analysis will not increase the probability of an accident previously evaluated.

   The proposed change limits the minimum reactor coolant flow. The design basis safety analyses have been analyzed and/or evaluated at the revised RCS flow. The re-analysis and evaluation have demonstrated that all safety analysis acceptance criteria are satisfied at the specified flow. Therefore, the change will not significantly increase the consequences of an accident previously evaluated.

The proposed technical specification limit for minimum allowed RCS flow was increased above the analytical limit to account for instrument error.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not alter the plant configuration or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change is consistent with the safety analyses. All safety analyses acceptance criteria are satisfied at the revised reactor coolant system flow. The limit will bound actual plant operation.

   The change reduces the RCS flow rate limit. Re-analysis of LOCA and non-LOCA transients determined all safety requirements of KNPP accident analyses were still met at the reduced RCS flow rate limit. Therefore, this proposed change does not significantly reduce the margin of safety.

Safety Evaluation for Proposed Change to TS 3.10.n

The minimum DNBR (MDNBR) value was removed from this paragraph of the Technical Specifications. The MDNBR value must be consistent with the approved DNBR correlation. For KNPP, this is currently the HTP CHF correlation. A safety evaluation was previously performed and approved for the minimum DNBR value of 1.14 for the HTP CHF correlation (Reference 2).

Significant Hazards Determination for Proposed Change to TS 3.10.n

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The Departure from Nucleate Boiling Ratio (DNBR) is not an accident initiator. Therefore, the change in the DNBR will not increase the probability of an accident previously evaluated.

   The proposed change to the DNBR value does not change plant configuration, operating set points, or overall plant performance. Therefore, the change will not increase the consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   All safety analyses acceptance criteria are satisfied using the HTP CHF correlation. The DNBR limits assumed in the safety analyses will bound actual plant operation and assures at 95/95 that DNBR will not occur. Therefore, there is no reduction in the margin of safety.

Safety Evaluation for Proposed Change to TS Figure 3.10-1

This figure of the Technical Specifications for Required Shutdown Reactivity vs. Boron Concentration was revised to reflect the values used in the safety analyses. The limits for the shutdown margin were not changed. The boron concentration values were extended to account for the longer fuel cycles.

This change is administrative. The required shutdown reactivity line was extended from 1300 ppm to 2000 ppm. The existing values on the figure are not changed in the analysis or in the Technical Specifications. Therefore, the change will not affect the health and safety of the public.

Significant Hazards Determination for Proposed Change to TS Figure 3.10-1

The proposed change was reviewed in accordance with the provisions of 10CFR50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Required Shutdown Reactivity vs. Boron Concentration was revised to reflect the longer cycle length and the resulting increase in boron concentration.

   The Required Shutdown Reactivity vs. Boron Concentration is not an accident initiator. Extending the boron concentrations to account for longer fuel cycles will not increase 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.

   The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change is consistent with the cycle length and core physics analyses for longer fuel cycles. Operation within the limits specified in the figure will assure all core safety evaluation acceptance criteria are satisfied. The limit will bound actual plant operation. Therefore, there is no reduction in the margin of safety.

Safety Evaluation for Proposed Change to TS Figure 3.10-2

This technical specification figure for Hot Channel Factor Normalized Operating Envelope was revised to reflect the values used in the safety analyses.

Significant Hazards Determination for Proposed Change to TS Figure 3.10-2

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The Hot Channel Factor Normalized Operating Envelope figure was revised to reflect the values used in the safety analyses.

   The Hot Channel Factor Normalized Operating Envelope figure is not an accident initiator. Changing the technical specification figure consistent with the assumptions of the accident analyses will not increase 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.

   The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The proposed change is consistent with the safety analyses. Operation within the limits specified in the figure will assure all safety analyses acceptance criteria are satisfied. The limit will bound actual plant operation. Therefore, there is no reduction in the margin of safety.

Environmental Considerations

This proposed amendment involves a change to the Technical Specifications. It does not modify any facility components located within the restricted area, as defined in 10 CFR 20. WPSC has determined that the proposed amendment involves no significant hazards considerations and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in the individual or cumulative occupational radiation exposure. Accordingly, this proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

ATTACHMENT 2

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

April 15, 1998

Proposed Amendment 152

Strike Out TS Pages:

TS ii
TS B2.1-1
TS B2.1-2
Figure TS 2.1-1
TS 3.10-1 through TS 3.10-10
TS B3.10-1 through TS B3.10-9
Figure TS 3.10-1
Figure TS 3.10-2

Section | Title | Page

BASIS - Safety Limits, Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of rated power, reactor coolant temperature and pressure have been related to DNB through ~~the W-3 & "L" Grid~~ DNB correlation~~s~~. The ~~"L" Grid~~ DNB correlation has been developed to predict the DNB flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ~~heat flux~~ ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to ~~1.30~~. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[†††]

The curves of Figure TS 2.I-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ~~ratio~~ is equal to ~~1.3~~ or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches ~~1.3~~ and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is assured is below these lines.

The curves are based on the ~~following~~ nuclear hot channel factor~~s:~~

$\cancel{F^N_{\Delta H} = 1.55}$ $\qquad$ $\cancel{F^N_Q = 2.51}$

~~and include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:~~

$\cancel{F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1 - P)]}$ ~~where P is the fraction of rated power~~

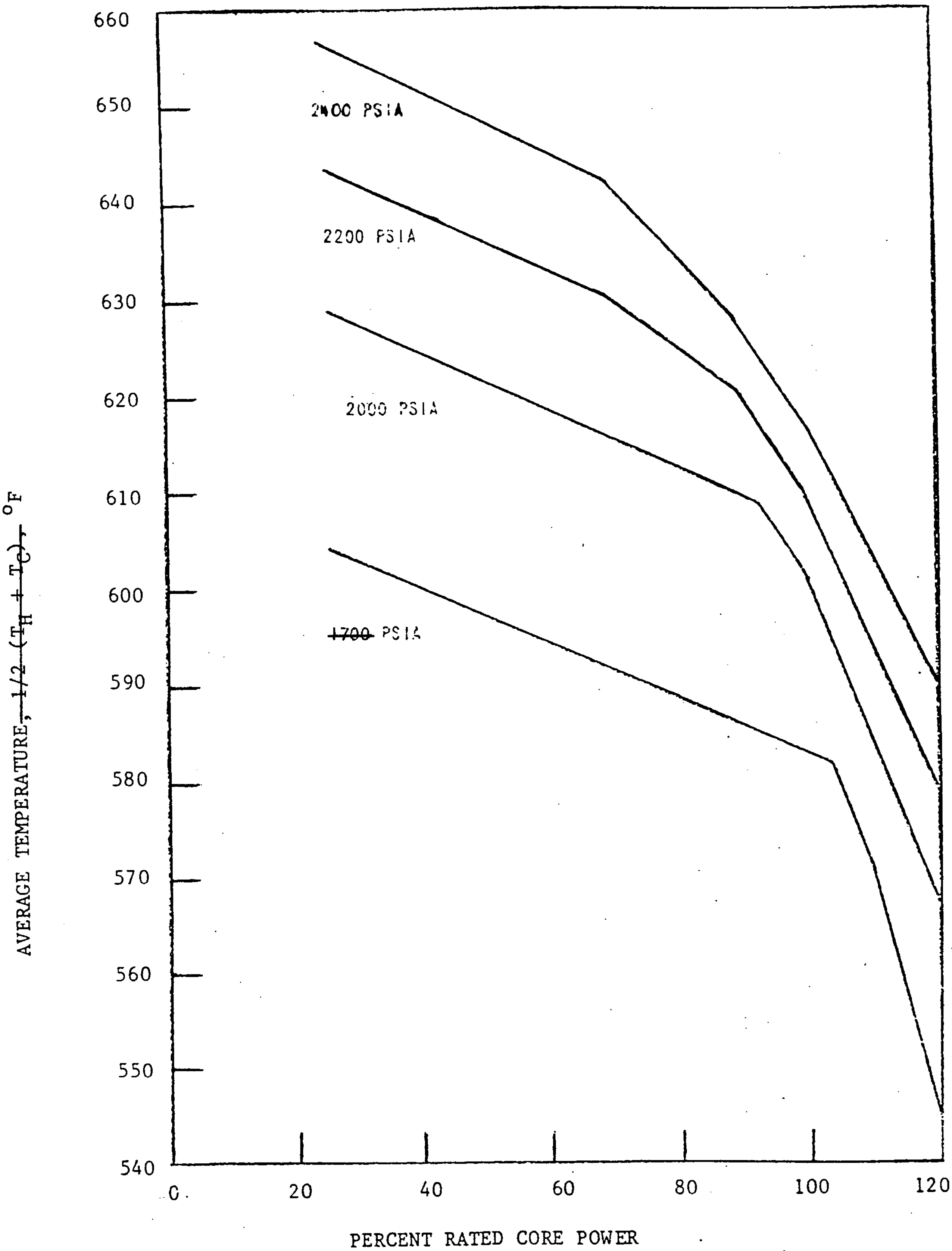
[†††] ~~USAR Section 3.3.3~~

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 insure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR ~~of < 1.30.~~

~~REFERENCES~~

~~(1) WCAP 8092~~



Safety Limits Reactor Core, ~~Thermal and Hydraulic Two Loop Operation, 100%~~ Flow, Minimum DNBR ~~> 1.3~~

FIGURE TS 2.1-1

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the HOT SHUTDOWN margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon or boron.

b. Power Distribution Limits

1. ~~At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:~~

~~A. $F_Q^N(Z)$ Limits~~ for Siemens Power Corporation Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z) \text{ for } P > .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z) \text{ for } P \leq .5$$

where:

P is the fraction of full power at which the core is OPERATING

K(Z) is the function given in Figure TS 3.10-2

Z is the core height location for the F_Q of interest

~~B. $F_{\Delta H}^{N}$ Limits~~ for Siemens Power Corporation Fuel

$$F_{\Delta H}^{N} \times 1.04 \leq 1.55 \, [1 + 0.2(1-P)]$$

where:

P is the fraction of full power at which the core is OPERATING

~~2. If, for any measured hot channel factor, the relationships specified in TS 3.10.b.1 are not true, reactor power shall be reduced by a fractional amount of the design power to a value for which the relationships are true, and the high neutron flux trip setpoint shall be reduced by the same fractional amount. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.~~

3. Following initial loading and at regular effective full-power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied.

4. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core for Siemens Power Corporation fuel:

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.28)/P \times K(Z)$$

where:

P is the fraction of full power at which the core is OPERATING

V(Z) is defined in Figure TS 3.10-6

$F_Q^{EQ}(Z)$ is a measured F_Q distribution obtained during the target flux determination

5. Power distribution maps using the movable detector system shall be made to confirm the relationship of TS 3.10.b.4 according to the following schedules with allowances for a 25% grace period:

   A. During the target flux difference determination or once per effective full-power monthly interval, whichever occurs first.

B. Upon achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.5.A.

C. If a power distribution map indicates an increase in peak pin power, $F_{\Delta H}^{N}$, of 2% or more, due to exposure, when compared to the last power distribution map, either of the following actions shall be taken:

i. $F_{Q}^{EQ}(Z)$ shall be increased by an additional 2% for comparison to the relationship specified in TS 3.10.b.4, OR

ii. $F_{Q}^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every 7 effective full-power days until a power distribution map indicates that the peak pin power, $F_{\Delta H}^{N}$, is not increasing with exposure when compared to the last power distribution map.

~~6. If, for a measured~~ $\cancel{F_{Q}^{EQ}}$~~, the relationships of TS 3.10.b.4 are not satisfied and the relationships of TS 3.10.b.1 are satisfied, within 12 hours take one of the following actions:~~

~~A. Take corrective actions to improve the power distribution and upon achieving equilibrium conditions measure the target flux difference and verify that the relationships specified in TS 3.10.b.4 are satisfied, OR~~

~~B. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the left hand sides of the relationships specified in TS 3.10.b.4 exceed the limits specified in the right hand sides. Reactor power may subsequently be increased provided that a power distribution map verifies that the relationships of TS 3.10.b.4 are satisfied with at least 1% of margin for each percent of power level to be increased.~~

~~7~~. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per full-power month.

~~8~~. The indicated axial flux difference shall be considered outside of the limits of TS 3.10.b.~~9~~ through TS 3.10.b.~~12~~ when more than one of the OPERABLE excore channels are indicating the axial flux difference to be outside a limit.

~~9~~. Except during physics tests, during excore detector calibration and except as modified by TS 3.10.b.~~10~~ through TS 3.10.b.~~12~~, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference.

~~10~~. At a power level > 90% of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band within 15 minutes or reactor power shall be reduced to a level no greater than 90% of rated power.

~~11~~. At power levels > 50% and ≤ 90% of rated power:

A. The indicated axial flux difference may deviate from its ± 5% target band for a maximum of 1 hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by -10% and +10% from the target axial flux difference at 90% rated power and increasing by -1% and +1% from the target axial flux difference for each 2.7% decrease in rated power < 90% and > 50%. If the cumulative time exceeds 1 hour, then the reactor power shall be reduced to ≤ 50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.

If the indicated axial flux difference exceeds the outer envelope defined above, then the reactor power shall be reduced to ≤ 50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.

B. A power increase to a level > 90% of rated power is contingent upon the indicated axial flux difference being within its target band.

~~12~~. At a power level no greater than 50% of rated power:

A. The indicated axial flux difference may deviate from its target band.

B. A power increase to a level > 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than 2 hours (cumulative) of the preceding 24-hour period.

One half of the time the indicated axial flux difference is out of its target band, up to 50% of rated power is to be counted as contributing to the 1 hour cumulative maximum the flux difference may deviate from its target band at a power level ≤ 90% of rated power.

~~13~~. Alarms shall normally be used to indicate nonconformance with the flux difference requirement of TS 3.10.b.~~10~~ or the flux difference time requirement of TS 3.10.b.~~11~~.A. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within 2 hours:

   A. Eliminate the tilt.

   B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

   A. Restrict maximum core power level by 2% of rated values for every 1% of indicated power tilt ratio > 1.0.

   B. If the tilt condition is not eliminated within 12 hours, the reactor shall be brought to a minimum load condition (≤ 30 Mwe).

4. If the indicated quadrant tilt is > 1.09 and there is no simultaneous indication of rod misalignment, the reactor shall immediately be brought to a no load condition (≤ 5% reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. The control banks shall be limited in physical insertion; insertion limits are shown in Figure TS 3.10-3. If any one of the control bank insertion limits shown in Figure TS 3.10-3 is not met:

   A. Within 1 hour, initiate boration to restore control bank insertion to within the limits of Figure TS 3.10-3, and

   B. Restore control bank insertion to within the limits of Figure TS 3.10-3 within 2 hours of exceeding the insertion limits.

   C. If any one of the conditions of TS 3.10.d.2.A or TS 3.10.d.2.B cannot be met, then within 1 hour action shall be initiated to

      - Achieve HOT STANDBY within 6 hours
      - Achieve HOT SHUTDOWN within the following 6 hours

3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure TS 3.10-1 must be maintained except for the Low Power Physics Test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In TS 3.10.e.1 and TS 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage. Rod misalignment limitations do not apply during physics testing.

1. When reactor power is ≥ 85% of rating, the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 12 steps when reactor power is ≥ 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and TS 3.10.b applied. If peaking factors are not determined within 4 hours, the reactor power shall be reduced to < 85% of rating.

2. When reactor power is < 85% but ≥ 50% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is < 85% but ≥ 50%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and TS 3.10.b applied. If the peaking factors are not determined within 4 hours, the reactor power shall be reduced to < 50% of rating.

3. And, in addition to TS 3.10.e.1 and TS 3.10.e.2, if the misaligned rod cluster control assembly is not realigned within 8 hours, the rod shall be declared inoperable.

f. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service, then:

A. For operation between 50% and 100% of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) at least once per 8 hours, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.

B. During operation < 50% of rating, no special monitoring is required.

2. Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3. If a rod cluster control assembly having a rod position indicator channel out of service is found to be misaligned from TS 3.10.f.1.A, then TS 3.10.e will be applied.

g. Inoperable Rod Limitations

1. An inoperable rod is a rod which does not trip or which is declared inoperable under TS 3.10.e or TS 3.10.h.

2. Not more than one inoperable full length rod shall be allowed at any time.

3. If reactor operation is continued with one inoperable full length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is made OPERABLE earlier. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

h. Rod Drop Time

At OPERATING temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If drop time is > 1.8 seconds, the rod shall be declared inoperable.

i. Rod Position Deviation Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged at least once per 8 hours after a load change > 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change > 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. During steady-state ~~100%~~ power operation, ~~T_{inlet}~~ shall be maintained < ~~535.5~~°F, except as provided by TS 3.10.n.

l. During steady-state ~~100%~~ power operation, Reactor Coolant System pressure shall be maintained > 2205 psig, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant flow rate shall be ≥ ~~89,000~~ gallons per minute average per loop. If reactor coolant flow rate is < ~~89,000~~ gallons per minute per loop, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in 2 hours or less to within limits or reduce power to < 5% of thermal rated power within an additional 6 hours. Following analysis, thermal power may be raised not to exceed a level analyzed to maintain a ~~minimum~~ DNBR ~~of 1.30~~.

BASIS

Shutdown Reactivity (TS 3.10.a)

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. To maintain the required trip reactivity, the rod insertion limits of Figure TS 3.10-3 must be observed. In addition, for HOT SHUTDOWN conditions, the shutdown margin of Figure TS 3.10-1 must be provided for protection against the steam line break accident ~~which requires more shutdown reactivity at end of core life (due to a more negative moderator temperature coefficient at end-of-life boron concentrations)~~.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. ~~The available control rod reactivity or excess beyond needs decreases with decreasing boron concentration, because the negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low.~~

The exception to the rod insertion limits in TS 3.10.d.3 is to allow the measurement of the worth of all rods ~~less the worth of the worst case of an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest, such as end-of-life cooldown or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These~~ measurements ~~will~~ augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Power Distribution Control (TS 3.10.b)

Criteria

Criteria have been chosen for Condition I and II events as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed the value assumed in the accident analyses.~~[1][2]~~ The peak linear power density is chosen to ensure peak clad temperature during a postulated large break loss-of-coolant accident is < the 2200°F limit. Second, the minimum DNBR in the core must not be ~~< 1.30~~ in normal operation or during Condition I or II transient events.~~[3]~~

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

$F_Q^{EQ}(Z)$ is the measured F_Q^N distribution obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for F_Q^N defined by TS 3.10.b.1 has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures remain < the 2200°F limit.

The $F_Q^N(Z)$ limits of TS 3.10.b.1~~.A~~ are derived from the LOCA analyses ~~in footnote~~ ~~[4]~~.

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

~~[1] USAR Section 4.3~~

~~[2] USAR Section 14~~

~~[3] USAR Section 4.3~~

~~[4] M.S. Stricker, "Kewaunee High Burnup Safety Analysis: Limiting Break LOCA and Radiological Consequences," ZN-NF-84-31 Rev. 1, Exxon Nuclear Company, October 1984.~~

In TS 3.10.b.1 ~~and TS 3.10.b.4~~ F_Q^N is arbitrarily limited for $P \leq 0.5$ (except for Low Power Physics Tests).

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are 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 horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The $F_{\Delta H}^N$ limit is determined from safety analyses of the limiting DNBR transient events. In these analyses, the important operational parameters are selected to minimize DNBR~~; T_{inlet} is 4° above nominal, RCS pressure is 30 psi below nominal, and RCS flow is assumed to be at the minimum design flow of 89,000 gpm average per loop~~.

The results of the safety analyses must demonstrate that minimum DNBR ~~≥ 1.30~~ for a fuel rod operating at the $F_{\Delta H}^N$ limit.

~~In the specified limit of $F_{\Delta H}^N$, there is an 8% allowance for design protection uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. When a measurement of $F_{\Delta H}^N$ is taken, measurement error must be allowed for and 4% is the appropriate allowance, as specified in TS 3.10.b.1. The logic behind the larger design uncertainty is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q^N; (b) the operator has a direct influence on F_Q^N through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^N$; 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^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.~~

The use of $F_{\Delta H}^N$ in TS 3.10.b.5 is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

No penalty for rod bow effects need be included in TS 3.10.b.1 for Siemens Power Corporation fuel~~(5).~~

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full-power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on 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 identifies operational anomalies which would otherwise affect these bases.

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥ 85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as shown in Figure TS 3.10-3.

3. The control bank insertion limits are not violated, except as allowed by TS 3.10.d.2.

4. Axial power distribution control specifications 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 specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.~~(6)~~

~~(5)~~N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.

~~(6)~~XN-NF-77-57 Exxon Nuclear Power Distribution Control for Pressurized Water Reactor, Phase II, January 1978.

Conformance with TS 3.10.b.~~9~~ through TS 3.10.b.~~12~~ ensures the F_Q^N upper bound envelope is not exceeded and xenon distributions will not develop which at a later time would cause greater local power peaking.

At the beginning of cycle, power escalation may proceed without the constraints of TS 3.10.b.5 since the startup test program provides adequate surveillance to ensure peaking factor limits. Target flux difference surveillance is initiated after achieving equilibrium conditions for sustained operation.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is determined from the nuclear instrumentation. This value, divided by the fraction of full power at which the core was OPERATING is the full-power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full-power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviations of ± 5% flux difference are permitted from the indicated reference value. Figure TS 3.10-5 shows a typical construction of target flux difference band ~~at~~ BOL and Figure TS 3.10-4 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of 1 hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band.

The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10% increment in peaking factor for flux difference in the range +10% to -10% from the target flux increasing by ± 1% from the target axial flux difference for each 2.7% decrease in rated power < 90% and > 50%. Therefore, while the deviation exists the power level is limited to 90% or lower depending on the indicated flux difference without additional core monitoring. If, for any reason, flux difference is not controlled within the ± 5% band for as long a period as 1 hour, then xenon distributions may be significantly changed and operation at 50% is required to protect against potentially more severe consequences of some accidents unless incore monitoring is initiated.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full-power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR ~~of 1.30~~ by an automatic protection system. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant Power Tilt Limits (TS 3.10.c)

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The 2% tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This symmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level. This monitoring is required by TS 4.1.

The 2 hour time interval in TS 3.10.c is considered ample to identify a dropped or misaligned rod. If the tilt condition cannot be eliminated within the 2 hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core ~~physics~~ map using the movable detector system. For a tilt ratio > 1.02 but ≤ 1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of 2% for each 1% of indicated tilt is required. ~~Physics~~ measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment. If a tilt ratio of > 1.02 but ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to ≤ 50%.

If a misaligned rod has caused a tilt ratio > 1.09, the core power shall be reduced by 2% of rated value for every I% of indicated power tilt ratio > 1.0. If after 8 hours the rod has not been realigned, the rod shall be declared inoperable in accordance with TS 3.IO.e, and action shall be taken in accordance with TS 3.10.g. If the tilt condition cannot be eliminated after 12 hours, the reactor shall be brought to a minimum load condition; i.e., electric power ≤ 30 MW. If the cause of the tilt condition has been identified and is in the process of being corrected, the generator may remain connected to the grid.

If the tilt ratio is > 1.09, and it is not due to a misaligned rod, the reactor shall be brought to a no load condition (i.e., reactor power ≤ 5%) for investigation by flux mapping. Although the reactor may be maintained critical for flux mapping, the generator must be disconnected from the grid since the cause of the tilt condition is not known, or it cannot be readily corrected.

Rod Insertion Limits (TS 3.10.d)

The allowed completion time of 2 hours for restoring the control banks to within the insertion limits provides an acceptable time for evaluation and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

Operation beyond the ~~LCO~~ limits is allowed for a short-time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss-of-flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The time limits of 6 hours to achieve HOT STANDBY and an additional 6 hours to achieve HOT SHUTDOWN allow for a safe and orderly shutdown sequence and are consistent with most the remainder of the Technical Specifications.

Rod Misalignment Limitations (TS 3.10.e)

During normal power operation it is desirable to maintain the rods in alignment with their respective banks to provide consistency with the assumption of the safety analyses, to maintain symmetric neutron flux and power distribution profiles, to provide assurance that peaking factors are within acceptable limits and to assure adequate shutdown margin.

Analyses have been performed which indicate that the above objectives will be met if the rods are aligned within the limits of TS 3.10.e. A relaxation in those limits for power levels < 85% is allowable because of the increased margin in peaking factors and available shutdown margin obtained while OPERATING at lower power levels. This increased flexibility is desirable to account for the nonlinearity inherent in the rod position indication system and for the effects of temperature and power as seen on the rod position indication system.

Rod position measurement is performed through the effects of the rod drive shaft metal on the output voltage of a series of vertically stacked coils located above the head of the reactor pressure vessel. The rod position can be determined by the analog individual rod position indicators, the plant process computer which receives a voltage input from the conditioning module, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

The plant process computer converts the output voltage signal from each IRPI conditioning module to an equivalent position (in steps) through a curve fitting process, which may include the latest actual voltage-to-position rod calibration curve.

The rod position as determined by any of these methods can then be compared to the bank demand position which is indicated on the group step counters to determine the existence and magnitude of a rod misalignment. This comparison is performed automatically by the plant process computer. The rod deviation monitor on the annunciator panel is activated (or reactivated) if the two position signals for any rod as detected by the process computer deviate by more than a predetermined value. The value of this setpoint is set to warn the operator when the Technical Specification limits are exceeded.

The rod position indicator system is calibrated once per REFUELING cycle and forms the basis of the correlation of rod position vs. voltage. This calibration is typically performed at HOT SHUTDOWN conditions prior to initial operations for that cycle. Upon reaching full-power conditions and verifying that the rods are aligned with their respective banks, the rod position indication may be adjusted to compensate for the effects of the power ascension. After this adjustment is performed, the calibration of the rod position indicator channel is checked at an intermediate and low level to confirm that the calibration is not adversely affected by the adjustment.

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

Core ~~Inlet~~ Temperature (TS 3.10.k)

The core ~~inlet~~ temperature limit is consistent with the safety analysis.

Reactor Coolant System Pressure (TS 3.10.l)

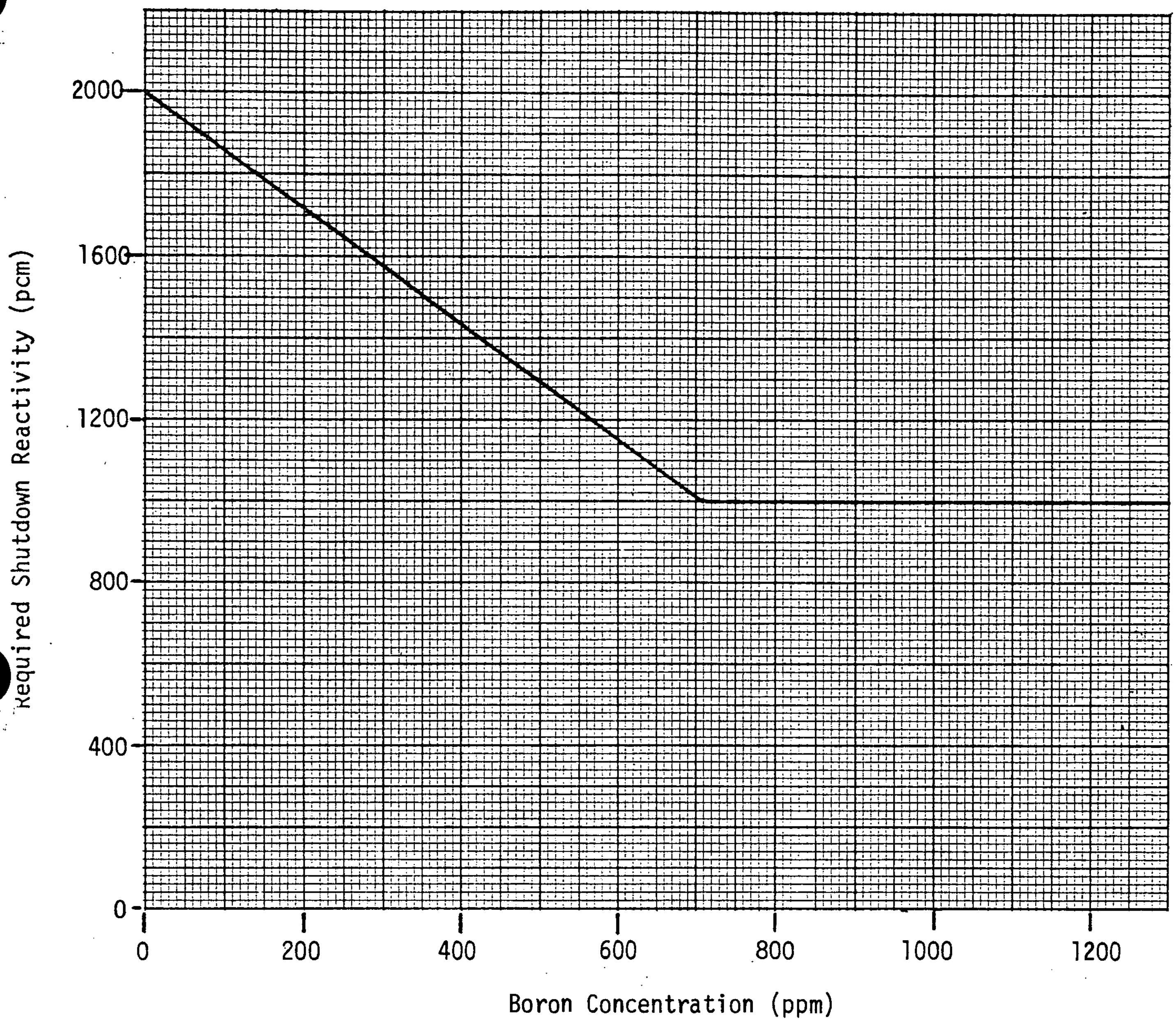
The Reactor Coolant System pressure limit is consistent with the safety analysis.

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant flow is consistent with the safety analysis.

DNB Parameters (TS 3.10.n)

~~The DNB related accident analyses assumed as initial conditions that the T_{inlet} was 4°F above nominal design or T_{avg} was 4°F above nominal design. The Reactor Coolant System pressure was assumed to be 30 psi below nominal design.~~



REQUIRED SHUTDOWN REACTIVITY VS.
REACTOR BORON CONCENTRATION

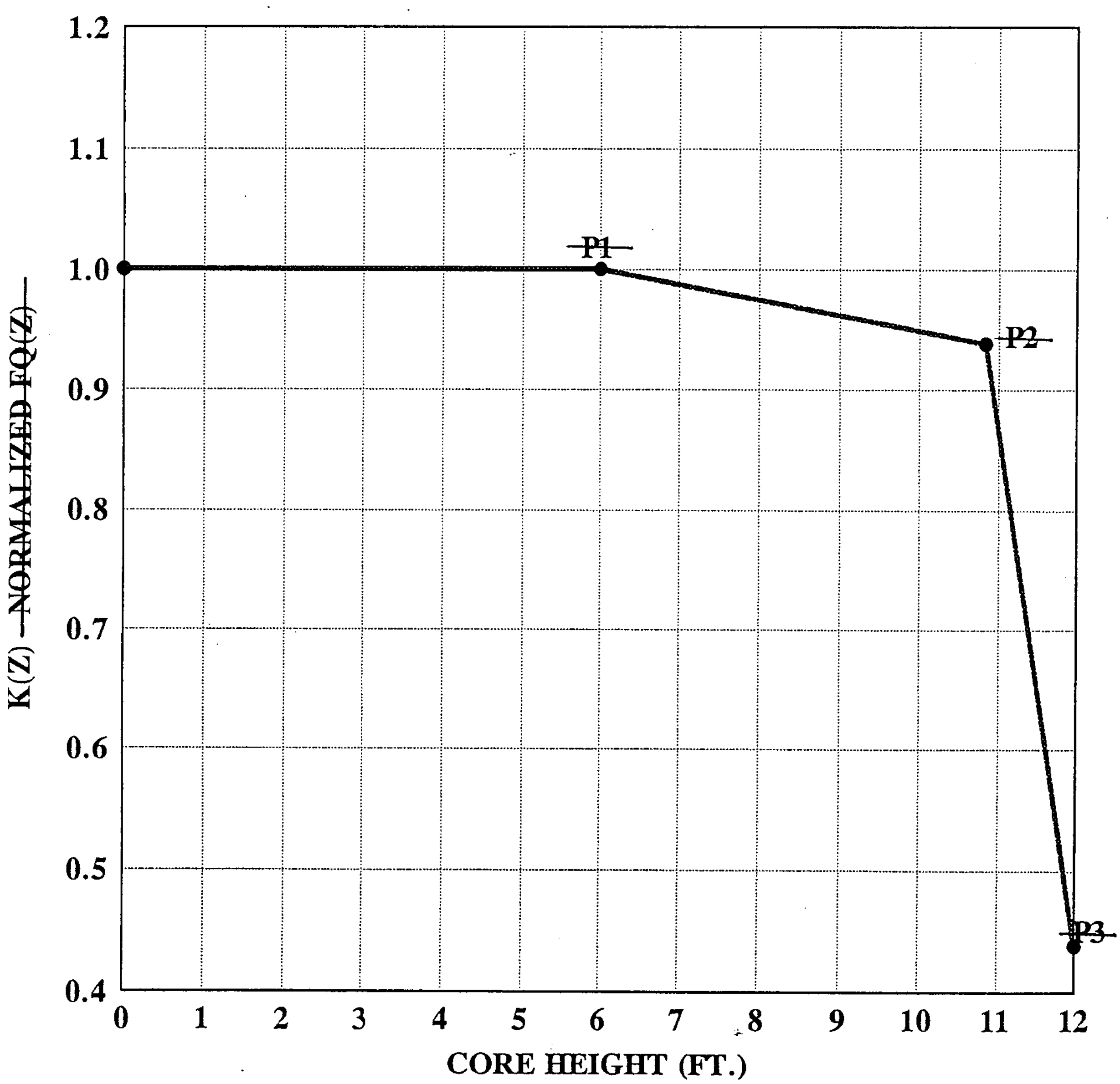
FIGURE TS3.10-1

Amendment No. ~~9~~
~~3/22/76~~

FIGURE TS 3.10-2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

~~Siemens Power Corporation Fuel K(Z) Coordinates~~
~~P1 (6, 1.0)~~
~~P2 (10.84, 0.938)~~
~~P3 (12, 0.438)~~
~~Normalized to 2.28~~



ATTACHMENT 3

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

April 15, 1998

<u>Proposed Amendment 152</u>

Affected TS Pages:

TS ii
TS B2.1-1
TS B2.1-2
Figure TS 2.1-1
TS 3.10-1 through TS 3.10-10
TS B3.10-1 through TS B3.10-9
Figure TS 3.10-1
Figure TS 3.10-2

Section	Title	Page

BASIS - Safety Limits, Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of rated power, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure TS 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is assured is below these lines.

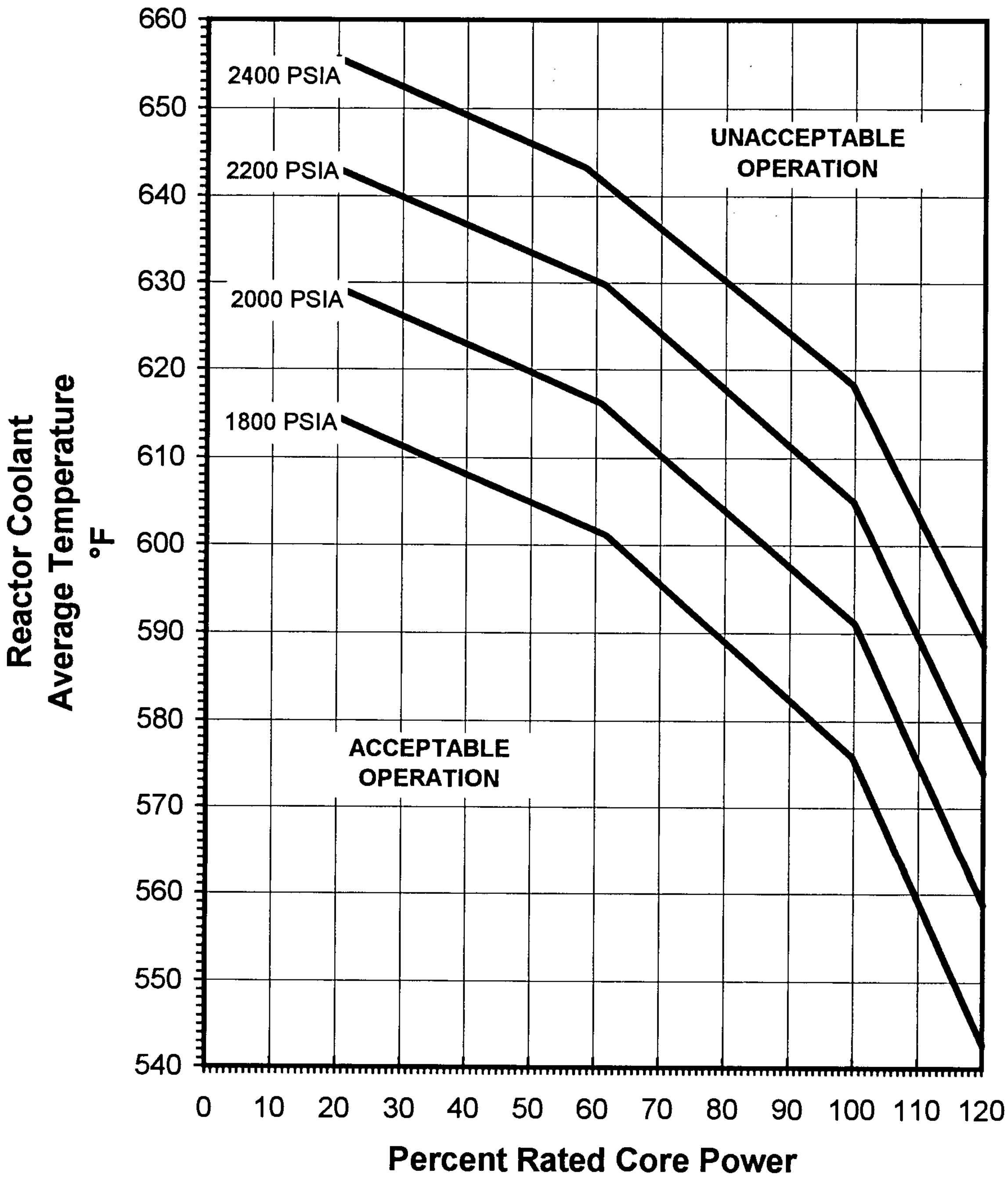
The curves are based on the nuclear hot channel factor limits of TS 3.10.b.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 insure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

FIGURE TS 2.1-1

Safety Limits Reactor Core, Minimum Coolant System Flow (TS 3.10.m), Minimum DNBR



3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the HOT SHUTDOWN margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon or boron.

b. Power Distribution Limits

1. When the reactor is critical, except during Low Power Physics Tests, $F_Q^N(Z)$ shall be limited by the following relationships for Siemens Power Corporation heavy fuel:

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.35)/P \times K(Z) \text{ for } P > .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.70) \times K(Z) \text{ for } P \leq .5$$

and for Siemens Power Corporation standard fuel:

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z) \text{ for } P > .5$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z) \text{ for } P \leq .5$$

where:

P is the fraction of full power at which the core is OPERATING

K(z) is the function given in Figure TS 3.10-2

Z is the core height location for the F_Q of interest

A. If the $F_Q^N(Z)$ relationships specified in TS 3.10.b.1 cannot be met, initiate the following actions:

- Reduce thermal power at least 1% for each 1% $F_Q^N(Z)$ exceeds limit within 15 minutes, and

- Reduce Power Range Neutron Flux-High Trip setpoints at least 1% for each 1% $F_Q^N(Z)$ exceeds limit within 72 hours, and

- Reduce Overpower Delta T Trip Setpoints at least 1% for each 1% $F_Q^N(Z)$ exceeds limit within 72 hours.

B. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by TS 3.10.b.1.A; THERMAL POWER may then be increased provided $F_Q^N(Z)$ is demonstrated through incore mapping to be within its limits.

2. When the reactor is critical, except during Low Power Physics Tests, $F_{\Delta H}^{N}(Z)$ shall be limited by the following relationships for Siemens Power Corporation heavy fuel:

$$F_{\Delta H}^{N}(Z) \times 1.04 \leq 1.70 [1 + 0.2(1-P)]$$

and for Siemens Power Corporation standard fuel:

$$F_{\Delta H}^{N}(Z) \times 1.04 \leq 1.55 [1 + 0.2(1-P)]$$

where:

P is the fraction of full power at which the core is OPERATING

A. If the $F_{\Delta H}^{N}(Z)$ relationships specified in TS 3.10.b.2 cannot be met, initiate the following actions:

- Reduce thermal power at least 1% for each 1% $F_{\Delta H}^{N}(Z)$ exceeds limit within 15 minutes, and

- Reduce Power Range Neutron Flux-High Trip setpoints at least 1% for each 1% $F_{\Delta H}^{N}(Z)$ exceeds limit within 72 hours, and

- Reduce Overpower Delta T Trip Setpoints at least 1% for each 1% $F_{\Delta H}^{N}(Z)$ exceeds limit within 72 hours.

B. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by TS 3.10.b.2.A; THERMAL POWER may then be increased provided $F_{\Delta H}^{N}(Z)$ is demonstrated through incore mapping to be within its limits.

3. Following initial loading and at regular effective full-power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 and TS 3.10.b.2 are satisfied.

4. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core for Siemens Power Corporation heavy fuel:

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.35)/P \times K(Z)$$

and for Siemens Power Corporation standard fuel:

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.28)/P \times K(Z)$$

where:

P is the fraction of full power at which the core is OPERATING

V(Z) is defined in Figure TS 3.10-6

$F_Q^{EQ}(Z)$ is a measured F_Q distribution obtained during the target flux determination

A. If, for a measured F_Q^{EQ}, the relationships of TS 3.10.b.4 are not satisfied and the relationships of TS 3.10.b.1 are satisfied, within 12 hours take one of the following actions:

i. Take corrective actions to improve the power distribution and upon achieving equilibrium conditions measure the target flux difference and verify that the relationships specified in TS 3.10.b.4 are satisfied, OR

ii. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the left hand side of the relationship specified in TS 3.10.b.4 exceeds the limit specified in the right hand side. Reactor power may subsequently be increased provided that a power distribution map verifies that the relationship of TS 3.10.b.4 is satisfied with at least 1% of margin for each percent of power level to be increased.

5. Power distribution maps using the movable detector system shall be made to confirm the relationship of TS 3.10.b.4 according to the following schedules with allowances for a 25% grace period:

A. During the target flux difference determination or once per effective full-power monthly interval, whichever occurs first.

B. Upon achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.5.A.

C. If a power distribution map indicates an increase in peak pin power, $F_{\Delta H}^{N}$, of 2% or more, due to exposure, when compared to the last power distribution map, either of the following actions shall be taken:

i. $F_{Q}^{EQ}(Z)$ shall be increased by an additional 2% for comparison to the relationship specified in TS 3.10.b.4, OR

ii. $F_{Q}^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every 7 effective full-power days until a power distribution map indicates that the peak pin power, $F_{\Delta H}^{N}$, is not increasing with exposure when compared to the last power distribution map.

6. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per full-power month with allowances for a 25% grace period.

7. The indicated axial flux difference shall be considered outside of the limits of TS 3.10.b.8 through TS 3.10.b.11 when more than one of the OPERABLE excore channels are indicating the axial flux difference to be outside a limit.

8. Except during physics tests, during excore detector calibration and except as modified by TS 3.10.b.9 through TS 3.10.b.11, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference.

9. At a power level > 90% of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band within 15 minutes or reactor power shall be reduced to a level no greater than 90% of rated power.

10. At power levels > 50% and ≤ 90% of rated power:

A. The indicated axial flux difference may deviate from its ± 5% target band for a maximum of 1 hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope bounded by -10% and +10% from the target axial flux difference at 90% rated power and increasing by -1% and +1% from the target axial flux difference for each 2.7% decrease in rated power < 90% and > 50%. If the cumulative time exceeds 1 hour, then the reactor power shall be reduced to ≤ 50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.

If the indicated axial flux difference exceeds the outer envelope defined above, then the reactor power shall be reduced to ≤ 50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.

B. A power increase to a level > 90% of rated power is contingent upon the indicated axial flux difference being within its target band.

11. At a power level no greater than 50% of rated power:

A. The indicated axial flux difference may deviate from its target band.

B. A power increase to a level > 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than 2 hours (cumulative) of the preceding 24-hour period.

One half of the time the indicated axial flux difference is out of its target band, up to 50% of rated power is to be counted as contributing to the 1 hour cumulative maximum the flux difference may deviate from its target band at a power level ≤ 90% of rated power.

12. Alarms shall normally be used to indicate nonconformance with the flux difference requirement of TS 3.10.b.9 or the flux difference time requirement of TS 3.10.b.10.A. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within 2 hours:

   A. Eliminate the tilt.

   B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

   A. Restrict maximum core power level by 2% of rated values for every 1% of indicated power tilt ratio > 1.0.

   B. If the tilt condition is not eliminated within 12 hours, the reactor shall be brought to a minimum load condition (≤ 30 Mwe).

4. If the indicated quadrant tilt is > 1.09 and there is no simultaneous indication of rod misalignment, the reactor shall immediately be brought to a no load condition (≤ 5% reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. The control banks shall be limited in physical insertion; insertion limits are shown in Figure TS 3.10-3. If any one of the control bank insertion limits shown in Figure TS 3.10-3 is not met:

   A. Within 1 hour, initiate boration to restore control bank insertion to within the limits of Figure TS 3.10-3, and

   B. Restore control bank insertion to within the limits of Figure TS 3.10-3 within 2 hours of exceeding the insertion limits.

C. If any one of the conditions of TS 3.10.d.2.A or TS 3.10.d.2.B cannot be met, then within 1 hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours.
- Achieve HOT SHUTDOWN within the following 6 hours.

3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure TS 3.10-1 must be maintained except for the Low Power Physics Test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In TS 3.10.e.1 and TS 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage. Rod misalignment limitations do not apply during physics testing.

1. When reactor power is ≥ 85% of rating, the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± I2 steps when reactor power is ≥ 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and TS 3.10.b applied. If peaking factors are not determined within 4 hours, the reactor power shall be reduced to < 85% of rating.

2. When reactor power is < 85% but ≥ 50% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is < 85% but ≥ 50%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and TS 3.10.b applied. If the peaking factors are not determined within 4 hours, the reactor power shall be reduced to < 50% of rating.

3. And, in addition to TS 3.10.e.1 and TS 3.10.e.2, if the misaligned rod cluster control assembly is not realigned within 8 hours, the rod shall be declared inoperable.

f. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service, then:

   A. For operation between 50% and 100% of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) at least once per 8 hours, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.

   B. During operation < 50% of rating, no special monitoring is required.

2. Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3. If a rod cluster control assembly having a rod position indicator channel out of service is found to be misaligned from TS 3.10.f.1.A, then TS 3.IO.e will be applied.

g. Inoperable Rod Limitations

1. An inoperable rod is a rod which does not trip or which is declared inoperable under TS 3.10.e or TS 3.10.h.

2. Not more than one inoperable full length rod shall be allowed at any time.

3. If reactor operation is continued with one inoperable full length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is made OPERABLE earlier. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

h. Rod Drop Time

At OPERATING temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If drop time is > 1.8 seconds, the rod shall be declared inoperable.

i. Rod Position Deviation Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged at least once per 8 hours after a load change > 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change > 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained < 568.8°F, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained > 2205 psig, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant flow rate shall be ≥ 85,500 gallons per minute average per loop. If reactor coolant flow rate is < 85,500 gallons per minute per loop, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in 2 hours or less to within limits or reduce power to < 5% of thermal rated power within an additional 6 hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

BASIS

Shutdown Reactivity (TS 3.10.a)

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. To maintain the required trip reactivity, the rod insertion limits of Figure TS 3.10-3 must be observed. In addition, for HOT SHUTDOWN conditions, the shutdown margin of Figure TS 3.10-1 must be provided for protection against the steam line break accident.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident.

The exception to the rod insertion limits in TS 3.10.d.3 is to allow the measurement of the worth of all rods. This measurement is a part of the Reactor Physics Test Program performed at the startup of each cycle. Rod worth measurements augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Power Distribution Control (TS 3.10.b)

Criteria

Criteria have been chosen for Condition I and II events as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed the value assumed in the accident analyses. The peak linear power density is chosen to ensure peak clad temperature during a postulated large break loss-of-coolant accident is less than the 2200°F limit. Second, the minimum DNBR in the core must not be less than the DNBR limit in normal operation or during Condition I or II transient events.

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

$F_Q^{EQ}(Z)$ is the measured $F_Q^N(Z)$ obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for $F_Q^N(Z)$ defined by TS 3.10.b.1 has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures remain < the 2200°F limit.

The $F_Q^N(Z)$ limits of TS 3.10.b.1 are derived from the LOCA analyses. The LOCA analyses are performed for Siemens Power Corporation heavy fuel and for Siemens Power Corporation standard fuel.

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent (5%) is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

In TS 3.10.b.1, $F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for Low Power Physics Tests).

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are 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 horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The $F_{\Delta H}^N$ limit is determined from safety analyses of the limiting DNBR transient events. The safety analyses are performed for Siemens Power Corporation heavy fuel and for Siemens Power Corporation standard fuel. In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is less than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^N$ limit.

The use of $F_{\Delta H}^N$ in TS 3.10.b.5.C is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

No penalty for rod bow effects need be included in TS 3.10.b.1 and TS 3.10.b.2 for Siemens Power Corporation fuel.[1]

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on 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 identifies operational anomalies which would otherwise affect these bases.

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥ 85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as shown in Figure TS 3.10-3.

3. The control bank insertion limits are not violated, except as allowed by TS 3.10.d.2.

4. Axial power distribution control specifications 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 specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.[2]

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[1] N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.

[2] XN-NF-77-57 Exxon Nuclear Power Distribution Control for Pressurized Water Reactor, Phase II, January 1978.

Conformance with TS 3.10.b.8 through TS 3.10.b.11 ensures the F_Q^N upper bound envelope is not exceeded and xenon distributions will not develop which at a later time would cause greater local power peaking.

At the beginning of cycle, power escalation may proceed without the constraints of TS 3.10.b.5 since the startup test program provides adequate surveillance to ensure peaking factor limits. Target flux difference surveillance is initiated after achieving equilibrium conditions for sustained operation.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is determined from the nuclear instrumentation. This value, divided by the fraction of full power at which the core was OPERATING is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviations of ± 5% flux difference are permitted from the indicated reference value. Figure TS 3.10-5 shows a typical construction of target flux difference band near BOL and Figure TS 3.10-4 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of 1 hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band.

The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10% increment in peaking factor for flux difference in the range +10% to -10% from the target flux increasing by ± 1% from the target axial flux difference for each 2.7% decrease in rated power < 90% and > 50%. Therefore, while the deviation exists the power level is limited to 90% or lower depending on the indicated flux difference without additional core monitoring. If, for any reason, flux difference is not controlled within the ± 5% band for as long a period as I hour, then xenon distributions may be significantly changed and operation at 50% is required to protect against potentially more severe consequences of some accidents unless incore monitoring is initiated.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit by an automatic protection system. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

<u>Quadrant Power Tilt Limits (TS 3.10.c)</u>

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The 2% tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This symmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level. This monitoring is required by TS 4.1.

The 2 hour time interval in TS 3.10.c is considered ample to identify a dropped or misaligned rod. If the tilt condition cannot be eliminated within the 2 hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core power distribution map using the movable detector system. For a tilt ratio > 1.02 but ≤ 1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of 2% for each 1% of indicated tilt is required. Power distribution measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment. If a tilt ratio of > 1.02 but ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to ≤ 50%.

If a misaligned rod has caused a tilt ratio > 1.09, the core power shall be reduced by 2% of rated value for every 1% of indicated power tilt ratio > 1.0. If after 8 hours the rod has not been realigned, the rod shall be declared inoperable in accordance with TS 3.10.e, and action shall be taken in accordance with TS 3.10.g. If the tilt condition cannot be eliminated after 12 hours, the reactor shall be brought to a minimum load condition; i.e., electric power ≤ 30 MW. If the cause of the tilt condition has been identified and is in the process of being corrected, the generator may remain connected to the grid.

If the tilt ratio is > 1.09, and it is not due to a misaligned rod, the reactor shall be brought to a no load condition (i.e., reactor power ≤ 5%) for investigation by flux mapping. Although the reactor may be maintained critical for flux mapping, the generator must be disconnected from the grid since the cause of the tilt condition is not known, or it cannot be readily corrected.

Rod Insertion Limits (TS 3.10.d)

The allowed completion time of 2 hours for restoring the control banks to within the insertion limits provides an acceptable time for evaluation and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

Operation beyond the rod insertion limits is allowed for a short-time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss-of-flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The time limits of 6 hours to achieve HOT STANDBY and an additional 6 hours to achieve HOT SHUTDOWN allow for a safe and orderly shutdown sequence and are consistent with most of the remainder of the Technical Specifications.

Rod Misalignment Limitations (TS 3.10.e)

During normal power operation it is desirable to maintain the rods in alignment with their respective banks to provide consistency with the assumption of the safety analyses, to maintain symmetric neutron flux and power distribution profiles, to provide assurance that peaking factors are within acceptable limits and to assure adequate shutdown margin.

Analyses have been performed which indicate that the above objectives will be met if the rods are aligned within the limits of TS 3.10.e. A relaxation in those limits for power levels < 85% is allowable because of the increased margin in peaking factors and available shutdown margin obtained while OPERATING at lower power levels. This increased flexibility is desirable to account for the nonlinearity inherent in the rod position indication system and for the effects of temperature and power as seen on the rod position indication system.

Rod position measurement is performed through the effects of the rod drive shaft metal on the output voltage of a series of vertically stacked coils located above the head of the reactor pressure vessel. The rod position can be determined by the analog individual rod position indicators (IRPI), the plant process computer which receives a voltage input from the conditioning module, or through the conditioning module output voltage via a correlation of rod position vs. voltage.

The plant process computer converts the output voltage signal from each IRPI conditioning module to an equivalent position (in steps) through a curve fitting process, which may include the latest actual voltage-to-position rod calibration curve.

The rod position as determined by any of these methods can then be compared to the bank demand position which is indicated on the group step counters to determine the existence and magnitude of a rod misalignment. This comparison is performed automatically by the plant process computer. The rod deviation monitor on the annunciator panel is activated (or reactivated) if the two position signals for any rod as detected by the process computer deviate by more than a predetermined value. The value of this setpoint is set to warn the operator when the Technical Specification limits are exceeded.

The rod position indicator system is calibrated once per REFUELING cycle and forms the basis of the correlation of rod position vs. voltage. This calibration is typically performed at HOT SHUTDOWN conditions prior to initial operations for that cycle. Upon reaching full power conditions and verifying that the rods are aligned with their respective banks, the rod position indication may be adjusted to compensate for the effects of the power ascension. After this adjustment is performed, the calibration of the rod position indicator channel is checked at an intermediate and low level to confirm that the calibration is not adversely affected by the adjustment.

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

Core Average Temperature (TS 3.10.k)

The core average temperature limit is consistent with the safety analysis.

Reactor Coolant System Pressure (TS 3.10.l)

The reactor coolant system pressure limit is consistent with the safety analysis.

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant flow limit is consistent with the safety analysis.

DNBR Parameters (TS 3.10.n)

The DNBR related safety analyses make assumptions on reactor temperature, pressure, and flow. In the event one of these parameters does not meet the TS 3.10.k, TS 3.10.l or TS 3.10.m limits, an analysis can be performed to determine a power level at which the MDNBR limit is satisfied.

FIGURE TS 3.10-1

Required Shutdown Reactivity
vs.
Reactor Boron Concentration

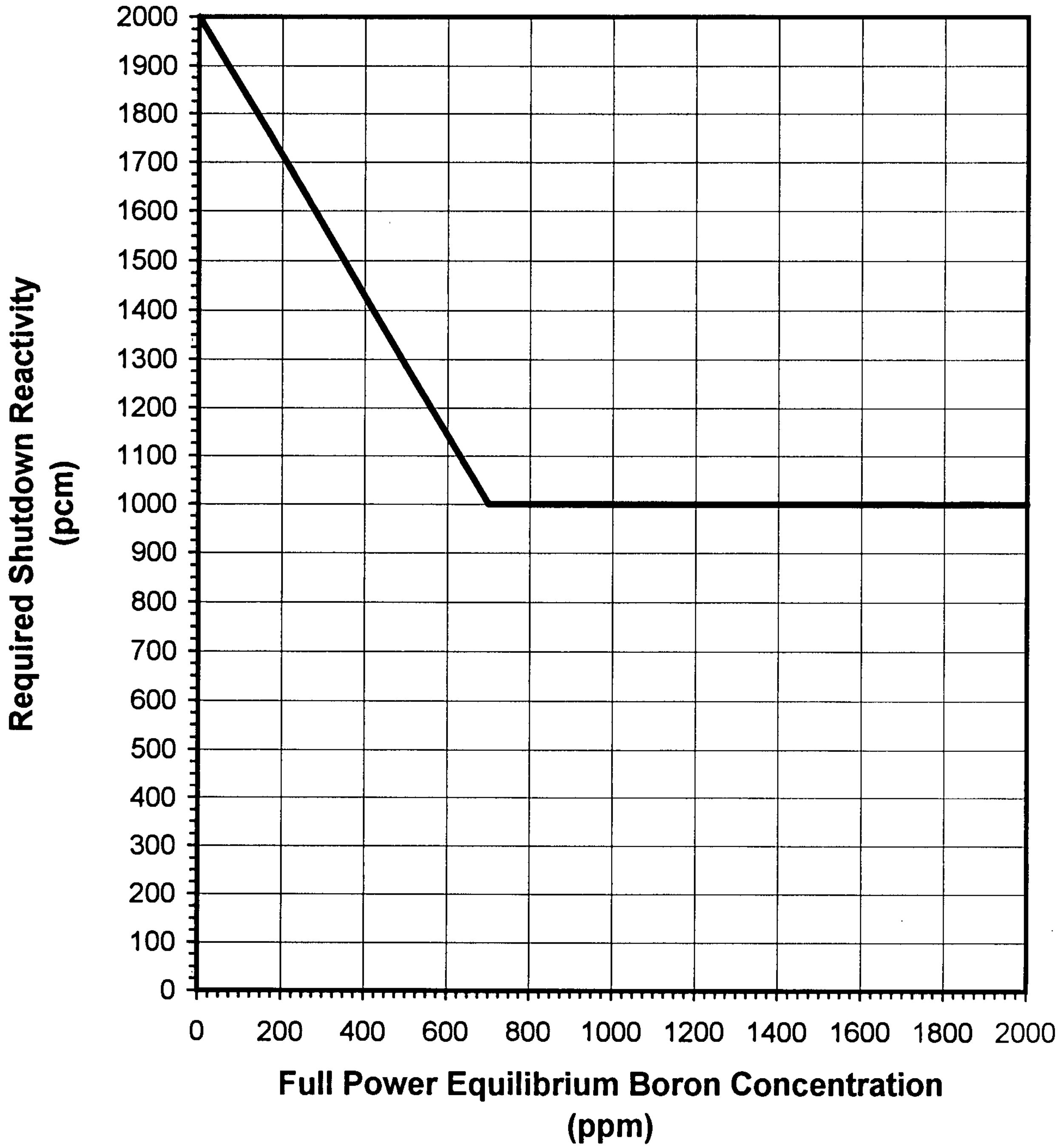
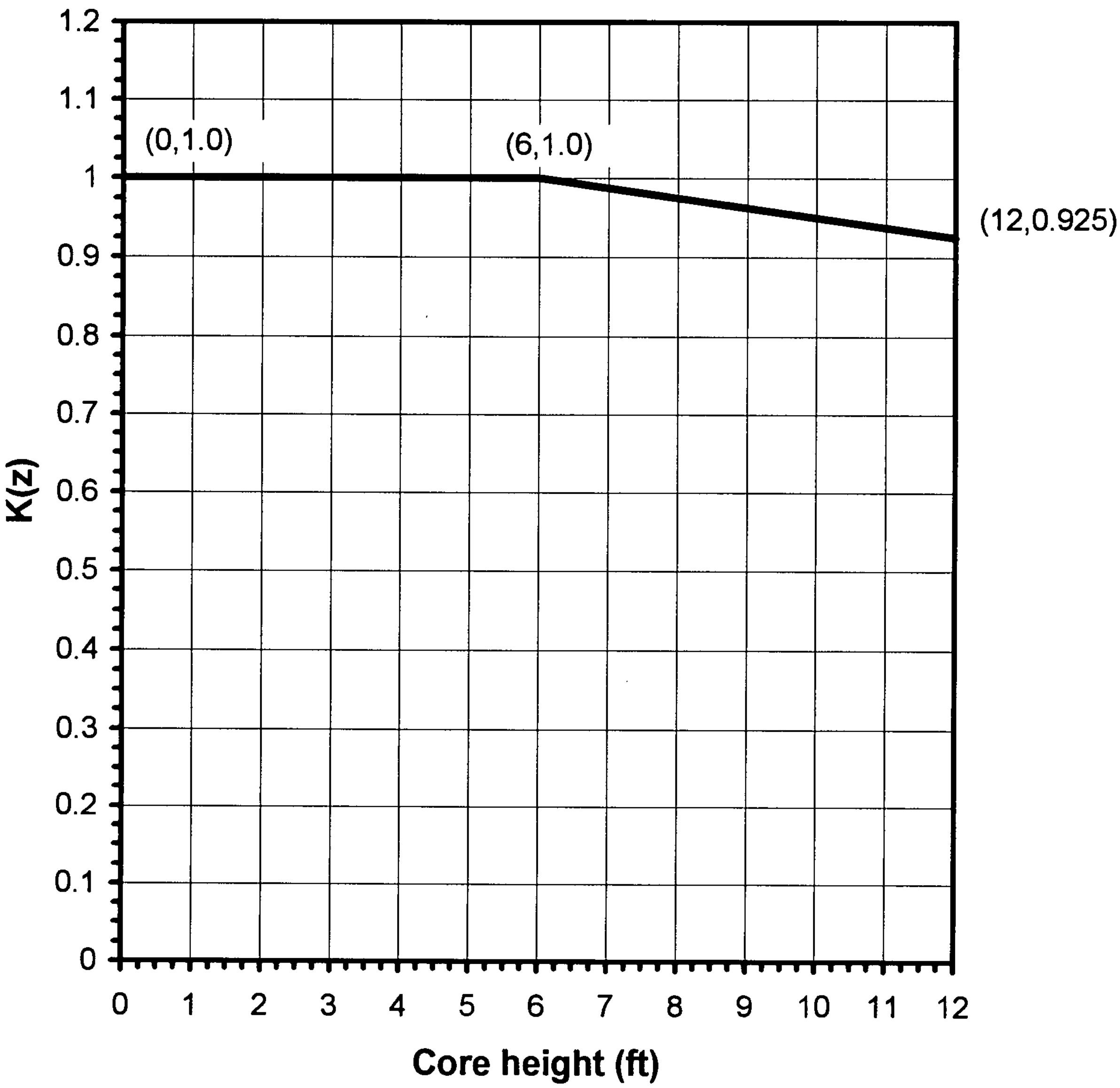


FIGURE TS 3.10-2

Hot Channel Factor
Normalized Operating Envelope



ATTACHMENT 4

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

April 15, 1998

Proposed Amendment 152

LOCA Safety Analyses at
Increased Peaking Factor Limits

LOCA SAFETY ANALYSES AT INCREASED PEAKING FACTOR LIMITS

1.0 Introduction

This report presents the results of an assessment of the impact of increased peaking factor limits on the Chapter 14 USAR Loss of Coolant Accident (LOCA) transient events for the Kewaunee Nuclear Power Plant (KNPP). The assessment is considered applicable for Cycle 23 and subsequent cycles of operation which are bounded by the conditions of the assessment.

This safety assessment will include an analysis and/or evaluation of the USAR Section 14.3.1 Small Break LOCA (SBLOCA) and the USAR Section 14.3.2 Large Break LOCA (LBLOCA) events.

2.0 Methods and Assumptions

Following are the major input assumptions for the SBLOCA and LBLOCA safety analyses. A complete set of input assumptions is detailed in the Cycle 23 Safety Analysis Input Document.

RCS Flow	83,400 gpm/loop
Core Bypass Flow	7.0%
$F^N_{\Delta H}$ (Z)	1.70*
F^N_Q (Z) (SBLOCA)	2.50
F^N_Q (Z) (LBLOCA)	2.35*
Power Level (100%)	1650 MWth
SI Injection Temperature	80 °F*
SI Delay Time	
High Head Injection	20 seconds*
Low Head Injection	25 seconds*
Core Physics Parameters	Consistent with Cycle 23* Reload Safety Evaluation
Set points	Consistent With Current Plant Set points
MDNBR Correlation/ Limit	Not Applicable
Fuel Design	SPC Heavy* (Transition Cycles With Non-Feed Regions of SPC Standard Fuel Were Evaluated)

* These input assumptions were changed from the previous evaluation.

The SBLOCA event was analyzed by Westinghouse using their approved NOTRUMP methodology. The LBLOCA event was analyzed by Westinghouse using their approved SECY UPI methodology. All SBLOCA and LBLOCA steady state initial operating assumptions were consistent with these methodologies.

3.0 <u>Results</u>

<u>SBLOCA</u>

An evaluation of the existing SBLOCA analysis of record was performed by Westinghouse in Reference 5.1. The existing SBLOCA analysis had a peak clad temperature (PCT) of 1053°F. In order to align the SBLOCA analysis of record with the assumptions in Section 2.0 of this attachment, Westinghouse evaluated the following changes:

1) An RCS flow reduction from 85,000 gpm/loop to 83,400 gpm/loop;
2) A steam generator tube plugging increase from 25% to 30%; and
3) A fuel design change from SPC Standard to SPC Heavy.

The effects of PCT changes due to Evaluation Model Updates and the steam generator tube plugging increase (which resulted in the RCS flow reduction) had been performed in Reference 5.2 and resulted in a PCT of 1041°F.

As shown in Reference 5.1, the introduction of SPC Heavy fuel resulted in a PCT benefit of 109°F. The conclusion of the evaluation was therefore that the KNPP SBLOCA licensing basis PCT should remain at 1041°F until the SPC Standard fuel assemblies have burned through two cycles of operation. After that point the KNPP SBLOCA licensing basis PCT can be reduced by 109°F to a value of 932°F. Both of these values are well below the PCT regulatory limit (acceptance criteria) of 2200°F.

LBLOCA

A LBLOCA analysis was performed using the assumptions of Section 2.0 of this attachment. The analysis is documented in Reference 5.3. The Reference 5.3 results compare to the LBLOCA acceptance criteria as follows:

Acceptance Criteria	Reference 5.3 Value	Acceptance Criteria Limit (Upper Bound)
PCT (°F)	1872 °F	2200°F
Maximum Local Zr/H_2O Reaction	3.3%	17%
Total Corewide Zr/H_2O Reaction	0.0033%	1%

All of the values meet the acceptance criteria. Since the LBLOCA analysis was performed with a full core of SPC Heavy fuel, sensitivities were run to determine whether or not a transition core penalty was required. Reference 5.3 shows that a transition core penalty is not required when non-feed SPC Standard fuel is in the core with SPC Heavy fuel. Therefore, the PCT value of 1872°F also applies to the transition cores.

4.0 Conclusions

The USAR Chapter 14 SBLOCA and LBLOCA design basis accidents have been reanalyzed and/or evaluated at the increased peaking factor limits (at 100% power) of 1.70 $F^N_{\Delta H}$ (Z) (SBLOCA and LBLOCA), 2.35 F^N_Q (Z) (LBLOCA), and 2.50 F^N_Q (Z) (SBLOCA).

All safety analysis acceptance criteria have been shown to be adequately satisfied under the revised plant condition assumptions. Therefore, the changes being assessed do not create an unreviewed safety question. The revised safety analyses will be incorporated into an update to the USAR Chapter 14.

5.0 References

5.1 "Safety Assessment for Transition to Siemens 14x14 Heavy Fuel," Westinghouse letter WPS-97-503 from J.A. Bugica, Jr. to D. Wanner, dated February 10, 1997.

5.2 "Safety Assessment for Increased SGTP to 30%," Westinghouse letter WPS-96-521 from J.A. Bugica, Jr. to D. Wanner, dated November 4, 1996.

5.3 Westinghouse Calculation Note SEC-LIS-5058-C8, Rev. 0, "Final UPI SECY WCOBRA/TRAC Siemens Heavy Fuel Calculations (with Revision 12)," dated April 15, 1998.

ATTACHMENT 5

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

April 15, 1998

<u>Proposed Amendment 152</u>

Non-LOCA Safety Analyses at
Increased Peaking Factor Limits

NON-LOCA SAFETY ANALYSES AT INCREASED PEAKING FACTOR LIMITS

1.0 Introduction

This attachment presents the results of an assessment of the impact of increased peaking factor limits on the Chapter 14 USAR non-LOCA transient events for the Kewaunee Nuclear Power Plant (KNPP). The assessment is considered applicable for Cycle 23 and subsequent cycles of operation which are bounded by the conditions of the assessment.

This safety assessment will include an analysis and/or evaluation of the following USAR events:

USAR SECTION	TRANSIENT EVENT
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition
14.1.2	Uncontrolled RCCA Withdrawal at Power
14.1.3	RCC Assembly Misalignment a) Statically Misaligned Full-Length Assemblies b) Dropped Full-Length Assembly Bank c) Dropped Full-Length Assemblies
14.1.4	Chemical and Volume Control System Malfunction a) Refueling b) Startup c) Power Operation (Manual Reactor Mode) d) Power Operation (Automatic Reactor Mode)
14.1.5	Startup of an Inactive Coolant Loop
14.1.6	Excessive Heat Removal Due to Feedwater System Malfunctions a) Manual Reactor Mode b) Automatic Reactor Mode c) No Load Initial Condition
14.1.7	Excessive Load Increase Incident a) Automatic Reactor Control (BOL & EOL) b) Manual Reactor Control (BOL & EOL)
14.1.8	Loss of Reactor Coolant Flow a) Loss of Two Pumps b) Under Frequency c) Locked Rotor
14.1.9	Loss of External Electrical Load a) Manual Reactor Control (BOL) b) Auto Reactor Control (BOL) c) Manual Reactor Control (EOL) d) Auto Reactor Control (EOL)
14.1.10	Loss of Normal Feedwater

USAR SECTION	TRANSIENT EVENT
14.1.11	Loss of A-C Power to the Plant Auxiliaries
14.2.1	Fuel Handling Accident
14.2.4	Steam Generator Tube Rupture
14.2.5	Rupture of a Steam Pipe (Main Steam Line Break)
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCC Assembly Ejection)
14.3.2	Post-LOCA Long Term Core Cooling and Subcriticality Requirements

2.0 Methods and Assumptions

The key input assumptions for the full power non-LOCA safety analyses are shown below. A complete set of input assumptions will be provided in the Cycle 23 Safety Analysis Input Document.

Reactor Coolant System Flow	83,500 gpm/loop
Reactor Coolant System Pressure	2185 psig*
Core Bypass Flow	7.0%
Steam Generator Tube Plugging	30%
$F^N_{\Delta H}$ (Z)	1.70*
F^N_Q (Z)	2.50*
Power Level (102%)	1683 Mwth
Core Average Temperature	573.1°F
Core Physics Parameters	Consistent with Cycle 23* Reload Safety Evaluation
Set points	Consistent with Current Plant Set points
MDNBR Correlation/Limit	High Thermal Performance (HTP)/1.14*
Fuel Design	Siemens Power Corporation (SPC)* Heavy
MSIV	5 seconds*
Engineered Safeguards	Plant Specific Curves*

* These input assumptions were changed from the previous evaluation.

Kewaunee is transitioning to a full core of SPC heavy fuel with the HTP spacer design. The SPC standard fuel will continue to have full power $F^N_{\Delta H}(Z)$ and F^N_Q (Z) limits of 1.55 and 2.28, respectively.

Rod drop and rod misalignment are analyzed in steady state conditions. KNPP will continue to have administrative restrictions on automatic rod control when rods are positioned at or below 215 steps, thus eliminating the need for analysis of the dropped rod accident in auto control.

Reactor protection system positive flux rate, negative flux rate, and reactor coolant pump underfrequency trip functions are not taken credit for in these safety analyses.

3.0 Results

Results of the non-LOCA safety analyses are presented in Tables 1 and 2. As shown, all safety analysis acceptance criteria are adequately satisfied at the revised plant conditions.

4.0 Conclusions

USAR Chapter 14 non-LOCA design basis accidents have been re-analyzed and/or evaluated at increased peaking factor limits and at revised plant conditions.

All safety analysis acceptance criteria have been shown to be adequately satisfied under the revised assumptions. The revised safety analyses will be incorporated into an update to the USAR Chapter 14.

TABLE 1

NON-LOCA SAFETY ANALYSIS

CONDITION II AND CONDITION III1 EVENTS

		Calculated Value/Acceptance Criteria		
USAR Section	Transient	MDNBR	Reactor Coolant System Pressure (psia)	Main Steam System Pressure (psia)
14.1.1	Uncontrolled Rod Withdrawal from Subcritical	3.218/1.14	2324/2750	1133/1210
14.1.2	Uncontrolled Rod Withdrawal at Power			
	Fast Rate Full Power	1.547/1.14	2249/2750	938/1210
	Slow Rate Full Power	1.362/1.14	2309/2750	946/1210
	Fast Rate Intermediate Power	2.039/1.14	2314/2750	953/1210
	Slow Rate Intermediate Power	1.169/1.14	2350/2750	1182/1210
14.1.3	Control Rod Misalignment FΔH=2.02	1.142/1.14	2200/2750	751/1210
14.1.4	Chemical and Volume Control System Malfunction	1.347/1.14	2321/2750	952/1210
14.1.5	Startup of Inactive Loop	5.878/1.14	2313/2750	1153/1210
14.1.6	Feedwater System Malfunction			
	BOC Manual Control	1.681/1.14	2200/2750	751/1210
	EOC Auto Control	1.647/1.14	2200/2750	751/1210
	Feedwater Reg Valve Failure	1.681/1.14	2215/2750	1061/1210
14.1.7	Excessive Load Increase			
	BOC Manual Control	1.681/1.14	2200/2750	751/1210
	BOC Auto Control	1.430/1.14	2200/2750	751/1210
	EOC Manual Control	1.478/1.14	2200/2750	751/1210
	EOC Auto Control	1.438/1.14	2200/2750	751/1210
14.1.8	Loss of Flow			
	2/2 Pump Trip	1.314/1.14	2291/2750	906/1210
	Underfrequency Trip	1.248/1.14	2316/2750	878/1210
14.1.9	Loss of Load			
	BOC Manual Control	1.681/1.14	2501/2750	1182/1210
	BOC Auto Control	1.681/1.14	2470/2750	1182/1210
	EOC Manual Control	1.681/1.14	2477/2750	1181/1210
	EOC Auto Control	1.681/1.14	2377/2750	1198/1210
14.1.10	Loss of Feedwater	1.681/1.14	2500/2750	1165/1210

TABLE 2

NON-LOCA SAFETY ANALYSIS

CONDITION IV EVENTS

USAR Section	Transient	Calculated Value/ Acceptance Criteria: MDNBR	Containment Pressure (psia)
14.2.5	Main Steam Line Break (Core Response, FΔH = 5.00) Upstream Flow Restrictor Downstream Flow Restrictor	 1.451/1.45 3.106/1.45	 --- ---
	Main Steam Line Break (Containment Response) SLB14MYY0	---	60.5/60.7

USAR Section	Transient	Calculated Value/Acceptance Criteria: Max Clad Temp. (°F)	Max Fuel Centerline Temp. (°F)	Max Energy Deposition (cal/grm)	RCS Pressure (psia)	MSS Pressure (psia)
14.2.6	Control Rod Ejection					
	BOC Full Power	2040/2700	4598/4700	182/200	2274/2750	863/1210
	BOC Zero Power	2555/2700	3924/4700	174/200	2306/2750	1028/1210
	EOC Full Power	2019/2700	4591/4700	181/200	2288/2750	864/1210
	EOC Zero Power	2688/2700	4031/4700	182/200	2277/2750	1022/1210

USAR Section	Transient	Calculated Value/Acceptance Criteria: % Fuel Rods < DNB Limit *	Max Clad Temp. (°F)	RCS Pressure (psia)	MSS Pressure (psia)
14.1.8	Locked Rotor	Reload Dependent Calculation/40	1507/2700	2365/2750	1044/1210

*% Fuel Rods with $F^N_{\Delta H}(Z) \geq 1.513$

ATTACHMENT 6

Letter from C.R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

April 15, 1998

Proposed Amendment 152

Radiological Assessment at
Increased Peaking Factor Limits

RADIOLOGICAL ASSESSMENT AT INCREASED PEAKING FACTOR LIMITS

WPSC performed a review of the design transients and accidents to assess the impact of the increased peaking factors on the radiological consequences of these events. Operation at higher peaking factors results in an increase in the limiting fuel assembly total activity and the associated gap source term. Three events were identified as being impacted by the proposed change and include the fuel handling accident in containment, the fuel handling accident outside containment, and fuel damage from a turbine missile. Each are discussed below.

Fuel Handling Accident In Containment (FHAIC)

Kewaunee Technical Specification Amendment 132 permitted operation with the containment airlock doors open during fuel handling activities. On April 7, 1998, WPSC provided an evaluation of the fuel handling accident in containment to support continued NRC review of Amendment 132. For expediency, the analysis was performed with core peaking factors consistent with those proposed for this amendment. Based upon the analysis, WPSC concluded that radiological consequences of a fuel handling accident in containment were acceptable. This analysis is still under NRC review.

Fuel Handling Accident Outside Containment (FHAOC)

The analysis for a FHAIC assumes release of the limiting fuel assembly gap activity without credit for holdup or filtration. Kewaunee Technical Specifications require periodic surveillance and operation of the Spent Fuel Pool Sweep and Exhaust System which includes charcoal filtration during refueling operations. This ventilation system supplies air across the spent fuel pool to be exhausted through the charcoal filters. With this additional protection, WPSC has concluded that any postulated release and consequential doses from a fuel handling accident outside containment are bounded by the results of the FHAIC.

Turbine Missile Damage to Spent Fuel

The Kewaunee USAR identifies the potential for a high trajectory turbine missile to damage fuel assemblies stored in the spent fuel pool. Because of the loss of energy in perforating through intervening walls and barriers and the travel distance after penetration, the probability of low trajectory missiles striking the spent fuel pool is negligible. Although acknowledged as low probability, the high trajectory analysis identifies the potential for 12 assemblies to be impacted by a turbine missile with the subsequent release of the assemblies' gap activity.

Since initial licensing in 1973, additional NRC guidance has been developed for assessing the potential for, and consequences of, turbine missiles including NUREG-0800 and R.G. 1.115. This guidance states that the risk from a high trajectory missile is insignificant unless the vulnerable target area is on the order of 10^4 square feet or more. The Kewaunee spent fuel pool surface is approximately 10^3 or an order of magnitude below the guidance value.

Additionally, more detailed probabilistic studies have been completed by the turbine generator manufacturer on the likelihood of a turbine missile. This information was reviewed by the NRC as part of Technical Specification Amendment 121 establishing the frequency for turbine control and stop valve testing and established a performance requirement of 1E-05/year as the probability of a turbine missile ejection. This is also consistent with the NRC guidance for an unfavorably oriented turbine-generator.

In conclusion, the probability of a turbine missile impacting the spent fuel is sufficiently low that this event and the associated radiological consequences are no longer required to be evaluated as design basis for the Kewaunee Plant. The evaluation for eliminating this accident from the design basis is forthcoming.

Large Break Loss of Coolant Accident (LB-LOCA)

The Updated Safety Analysis Report discusses the postulated doses from the design basis accident. Per NRC Reg. Guide 1.4, it is assumed that 100 percent of the noble gases and 50 percent of the iodines in the core's fission product inventory will be released to the Reactor Containment Vessel.

The changes to the peaking factors will not increase the total inventory of noble gases or iodines in the reactor inventory. Therefore, WPSC has concluded that any postulated release and consequential doses from a Large Break-Loss of Coolant Accident are bounded by the previous analysis results.

Pending NRC approval of the recently transmitted analysis for a FHAIC, WPSC concludes that the proposed changes do not significantly increase the radiological consequences of previously evaluated accidents.