Exhibit A

Prairie Island Nuclear Generating Plant
License Amendment Request Dated January 13, 1986

Evaluation of Proposed Changes to the
Technical Specifications Appendix A of
Operating License DPR-42 and DPR-60

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90 the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following change to Appendix A, Technical Specifications:

1. Reference WRB-1 DNB Correlation

Proposed Change

Add reference to the WRB-1 DNB correlation for Westinghouse fuel and identify that the W-3 DNB correlation will be used for Exxon fuel.

Pages affected: TS.2.1-1, 2, 3, TS.2.3-6, TS.3.10-8, TS.3.10-13, Figure TS.2.1-1 (The bases on existing page TS.2.1-3 will be moved to proposed page TS.2.1-2. Therefore, there is no page TS.2.1-3 in Exhibit C.)

Reason For Change

In our April 19, 1985 submittal revising Topical Report NSPNAD-8102 Rev 3, we requested approval of methodology changes which would allow the WRB-1 correlation to be used for Westinghouse fuel. The proposed changes are consistent with the methodology changes.

The reference to Westinghouse fuel does not include the Westinghouse Standard fuel (LOPAR, Low Parasitic), it only refers to the Optimized Fuel Assemblies (OFA). At this time, we have no plans to reuse old Westinghouse standard fuel. Prior to using these assemblies, we would complete the necessary analyses. If the analyses support a 50.59 type evaluation, we would use the old Westinghouse standard fuel and update the Technical Specification Bases in the next Technical Specification change to be submitted. If the review cannot support a 50.59 evaluation, an amendment would be proposed prior to use of this fuel.

Significant Hazards Evaluation

The proposed changes are related to methodology changes currently under review by the Staff. While these changes may change the consequences of a previously-analyzed accident or may change a safety margin, the results are clearly within all acceptable criteria.

For these reasons, operation of the Prairie Island Nuclear Generating

Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

2. Temperature Coefficient Changes

Proposed Change

Delete reference to the "moderator temperature coefficient" in Specification 3.1.F.1.

In place of the existing restriction on isothermal temperature coefficient, require the isothermal temperature coefficient to be below 5 pcm/^{o}F when below 70% power and negative above 70% power.

Change the associated bases.

Replace the existing action statement, specification 3.1.F.3, with the Standard Technical Specification action statements, specification 3.1.1.3. Except the requirement to submit a special report in 10 days has been changed to allow 30 days to submit the report.

Pages affected: TS.3.1-17 and 18

Reason For Change

The existing specifications refer to moderator temperature coefficient for one fuel type and isothermal temperature coefficient for another. This change will simplify the specifications by only referring to one temperature coefficient for both fuel types; the isothermal temperature coefficient.

Fuel economics are significantly affected by the isothermal temperature coefficient limitations. In order to operate the plant more economically a larger (more positive) temperature coefficient is specified.

The existing action statement for the temperature coefficients is unclear and is being replaced with the applicable Standard Technical Specification action statement.

Thirty days to submit the special report is requested for two reasons. 10CFR50.73 allows 30 days for all Reportable Event reporting. Secondly, the isothermal temperature coefficient is measured at the beginning of the cycle during physics testing. This is a very busy time for the engineers that would write this report.

Significant Hazards Evaluation

Exhibit H contains the Safety Evaluation for the Isothermal Temperature Coefficient changes.

These temperature coefficient changes have also been found to be conservative for Unit 2 Cycle 10 (current cycle in operation).

The addition of the Standard Technical S cifications action statement constitutes an additional limitation not presently in the specifications.

While the temperature coefficient revision may change the consequences of a previously-analyzed accident or may change a safety margin, the results are clearly within all acceptable criteria.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

3. Accumulator Specification Change

Proposed Change

Change the volume requirement for the accumulators from "between 1250 and 1282.9" to "1270 $\pm$20" in specification 3.3.A.1.b.(2).

Pages Affected: TS.3.3-1

Reason For Change

This change is being made in an effort to make the accumulator volume setpoint easier for operators to use without significantly changing accumulator volumes. The recent LOCA analyses have used a nominal water volume of 1266.5 cubic feet (Exhibit E, Table 14.6-2).

The 20 cubic foot plus or minus tolerance will provide more allowance for drift of the level instruments as well as more operational flexibility.

Significant Hazards Evaluation

Westinghouse uses the nominal accumulator level as an input for their LOCA analyses. The nominal value proposed is 3.5 cubic feet (approximately a 0.3%) larger than the nominal level. The maximum water volume will increase with this change from 1282.9 cubic feet to 1290. The associated decrease in gas pressure is not large enough to

affect the accumulator performance. These minor changes proposed in Exhibit B and discussed above will have no significant effect of the safety of the plant.

While this change may result in some change in the consequences of a previously-analyzed accident, the results would be clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

4. Peaking Factor Changes

Proposed Change

Change the peaking factors limits in Section 3.10 as shown below:

	Old Value	New Value
FQ	2.32	2.30
FdH	1.55	1.60
FdH equation	1+0.2(1-P)	1+0.3(1-P)

Note: The old FQ for Westinghouse fuel was 2.21.

Revise the peaking factor equations on page TS.2.1-2.

Delete definitions of $BU(E_j)$ and E_j in the specification and references to them in the bases.

Increase the required high neutron flux trip setpoint reduction from 1% to 3.33% for each percent that measured FdH exceeds the limit.

Revise the Bases of Section 3.10 as necessary for the peaking factor changes. The bases have also been edited to remove some outdated information, e.g. deletion of the definition of the term $F_Q(z)$, deletion of an description of the FdH uncertainties and deletion of a discussion of rod bow.

Delete Figure TS.3.10-7, renumber the next sequentially numbered curve and delete references to the curve.

Pages affected: TS-x, TS.2.1-2, TS.3.10-1, 2, 9, 10, 11, 13, 17, and Figures TS.3.10-7 and 8

Reason For Change

The proposed peaking factors and equations will provide the plant with more operating flexibility than previously existed. Both FQ and FdH are being changed in order to obtain a more optimum relationship between the two limits. The FdH equation change will provide a less restrictive limit at reduced powers.

The normalized exposure dependent function, $BU(E_j)$ was used for Exxon Nuclear fuel. Since this value of this function is one for all burnup values up to 55 GWD/MTU, it can be deleted.

The increase in the setpoint reduction for FdH corrects a error in our specifications. If the FdH limit were exceeded by 1%, for example, power would need to be decreased by 3.33%, not the 1% currently in our specification in order to comply with the limit. Similarly, the high neutron flux trip setpoint should be reduced 3.33% for each 1% that the FdH limit is exceeded.

The bases are being revised in accordance with these changes and also being simplified by removing outdated and inaccurate information. On page TS.3.10-10, the reference to "experimental error" has been changed to "measurement error"; a more applicable term. The rod bow bases are being deleted since the rod bow effects are explicitly calculated in our analyses.

Significant Hazards Evaluation

Supporting these changes are Exhibits E and H.

The transient analyses have been done using an FQ of 2.32 and FdH of 1.60 (See Exhibit H, Table 4.2). The change in the FdH equation was also evaluated (Exhibit H, Section 3.6) and found to conform to acceptance criteria. These evaluations have been performed using NSP methods which include methodology changes submitted for NRC review on April 19, 1985. The effect of new upper internals was evaluated and found to have only a minor effect on the results shown in Exhibit H and will meet all acceptance criteria. These new peaking factor limits were also evaluated for Unit 2 Cycle 10 (currently operating Unit 2 cycle) and found to be acceptable.

A revised large break LOCA analysis using the new peaking factors has been performed for Westinghouse fuel including assessments of penalties associated with Upper Plenum Injection and with transition core hydraulic mismatch applicable to a core with Westinghouse and Exxon fuel types. This analysis is contained in Exhibit E. This analysis assumed an FQ of 2.30, FdH of 1.60, the new upper internals and the removal of thimble plugs. This analysis will bound the first cycle of Westinghouse fuel operation for both units, up to a maximum peak pin exposure of 22,000 MWD/MTU. This analysis will only be needed for one cycle operation. Approval of the new UPI (Upper Plenum Injection) Model is expected for subsequent cycles (refer to our letter dated 3/11/85 titled: Upper Plenum Injection LOCA Model

Development).

The fuel parameters used as input for the LOCA analysis were generated using the Revised Thermal PAD Model. Due to the use of the Revised Thermal PAD Model, Westinghouse has evaluated the effect of burnup on peak cladding temperatures (PCT) predicted for the LOCA through the maximum burnup level of cycle 11 using the currently approved LOCA models (1981 EM) as required Reference 18 of Exhibit E. At a peak pin burnup of 22,000 MWD/MTU (maximum burnup for either Unit during cycle 11), the burnup evaluation predicted a PCT of 1934°F compared with a PCT of 2098°F for the beginning-of-life (maximum densification) case, demonstrating that the time of maximum densification remains limiting in terms of peak clad temperature.

A revised large break LOCA analysis has been performed for the worst case break (CD = 0.4) for Exxon fuel. This analysis (submitted November 4, 1985) assumed an FQ of 2.32, FdH of 1.60, thimble plug removal and the new upper internals.

Subsequent to performing the Exxon fuel analysis, a more conservative examination of the effect of thimble plug removal was found to yield a slightly higher peak clad temperature. The results of this examination applied to the Exxon fuel continues to show a substantial margin to the the 2200° F Appendix K acceptance criteria and remains bounded in terms of clad temperature by the limiting case of the Cycle 11 transition core.

These peaking factor limits are conservative for Unit 2 Cycle 10 (current cycle in operation). The use of the new upper internals and thimble plug removal assumption is conservative. The smaller water volume in the new upper internals makes the new internals the limiting upper internals (See 11/4/85 submittal). The removal of thimble plugs decreases the active core flow making it the limiting case.

While this change may result in some change in the consequences of a previously-analyzed accident or may change in some way a safety margin, the results are clearly within all acceptable criteria.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

5. Deletion of the third line segment of the K(z) curve

Proposed Change

On Figure TS.3.10-5, delete the third line segment and extend the second line segment to the 12 foot level.

Change the associated Bases.

Pages affected: TS.3.10-9, Figure TS.3.10-5

Reason For Change

This change will allow more F_Q flexibility in the top of the core.

Significant Hazards Evaluation

This change is supported by recent small break LOCA analyses performed by Westinghouse on Westinghouse and Exxon fuel assemblies. The Westinghouse fuel analysis is contained in Exhibit F and the Exxon fuel analysis is contained in Exhibit G. Both analyses show that the small break LOCA peak cladding temperature is approximately 1000°F and support the removal of the third line segment of the K(z) curve.

While this revision may change the consequences of a previously-analyzed accident or may change in some way a safety margin, the results are clearly within all acceptable criteria.

For these reasons, operation of the Prairie Island Nuclear Generating Plant in accordance with this proposed change will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

Exhibit B

Prairie Island Nuclear Generating Plant
License Amendment Request Dated January 13, 1986

Proposed Changes Marked Up
on Existing Technical Specification Pages

Exhibit B consists of the existing Technical Specification pages with the proposed changes written on those pages. Existing pages affected by this change are listed below:

TS-x
TS.2.1-1
TS.2.1-2
TS.2.1-3 (this page will be deleted)
TS.2.3-6
Figure TS.2.1-1
TS.3.1-17
TS.3.1-18
TS.3.3-1
TS.3.10-1
TS.3.10-2
TS.3.10-8
TS.3.10-9
TS.3.10-10
TS.3.10-11
TS.3.10-13
TS.3.10-17
Figure TS.3.10-5
Figure TS.3.10-7 (this page will be deleted)
Figure TS.3.10-8 (this page will be renumbered to TS.3.10-7)

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, reactor coolant system pressure and coolant temperature during operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure TS.2.1-1. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

Basis

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; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation[5]. The W-3 DNB correlation ~~has~~ have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, 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 DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. ~~A DNB ratio of 1.30~~ correspond~~s~~ 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. ~~(1)~~

The W-3 DNB correlation is used for Exxon Nuclear fuel. The WRB-1 DNB correlation is used for Westinghouse fuel

and WRB1

for the Exxon Nuclear fuel and to 1.17 for the Westinghouse fuel

These limits

The solid curves of Figure TS 2.1-1 represent the loci of points of thermal power, coolant pressure, and coolant average temperature for which either the coolant enthalpy at the core exit is limiting or the DNB ratio is limiting. For the 1685 psig and 1985 psig curves, the coolant average enthalpy at the core exit is equal to saturated water enthalpy below power levels of 91% and 74% respectively. For the 2235 psig and 2385 psig curves, the coolant average temperature at the core exit is equal to 650°F below power levels of 64% and 73% respectively. For all four curves, the DNBR is ~~equal to 1.3~~ limiting at higher power levels. The area of safe operation is below these curves.

The plant conditions required to violate the limits in the lower power range are precluded by the self-actuated safety valves on the steam generators. The highest nominal setting of the steam generator safety valves is 1129 psig (saturation temperature 560°F). At zero power the difference between primary coolant and secondary coolant is zero and at full power it is 50°F. The reactor conditions at which steam generator safety valves open is shown as a dashed line on Figure TS.2.1-1.

Except for special tests, power operation with only one loop or with natural circulation is not allowed. Safety limits for such special tests will be determined as a part of the test procedure.

Conservative for

The curves are ~~based on~~ conservative for the following nuclear hot channel factors ~~(2)~~:

$$F^{N}_{\Delta H} = \cancel{1.58}\ 1.60 \left[1 + \cancel{0.2}\ 0.3(1-P)\right] \quad ; \text{ and } F^{N}_{\cancel{q}\ Q} = \cancel{2.71}\ 2.30$$

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10. ~~Local peaking factors due to fuel densification and included in the hot channel factors, (4)~~.

This combination of hot channel factors is higher than that 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 covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified by Figure TS.3.10-1 assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30~~[3]~~.

for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

~~References:~~

~~(1) FSAR, Section 3.2.2~~
~~(2) FSAR, Section 3.2.1~~
~~(3) FSAR, Section 14.1~~
~~(4) WCAP 8091~~

The other reactor trips specified in A.3. above provide additional protection. The trip initiated by steam/feedwater flow mismatch in coincidence with low steam generator water level is designed for protection from a sudden loss of the reactor's heat sink. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of load transient.

add space between "high" and "and"

change "e" to "i"

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the power range nuclear flux highand low trip to assure that the criteria are met for rod ejection from partial power.

initiated

The negative power range rate trip provides protection satisfying all IEEE criteria to assure that minimum DNBR is maintained ~~above~~ 1.30 for all multiple control rod drop accidents. Analysis indicates (Section 14.1.3) that in the case of a single rod drop, a return to full power will be ~~indicated~~ by the automatic reactor control system in response to a continued full power turbine load demand and it will not result in a DNBR of less than 1.30. Thus, automatic protection for a single rod drop is not required. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

References:

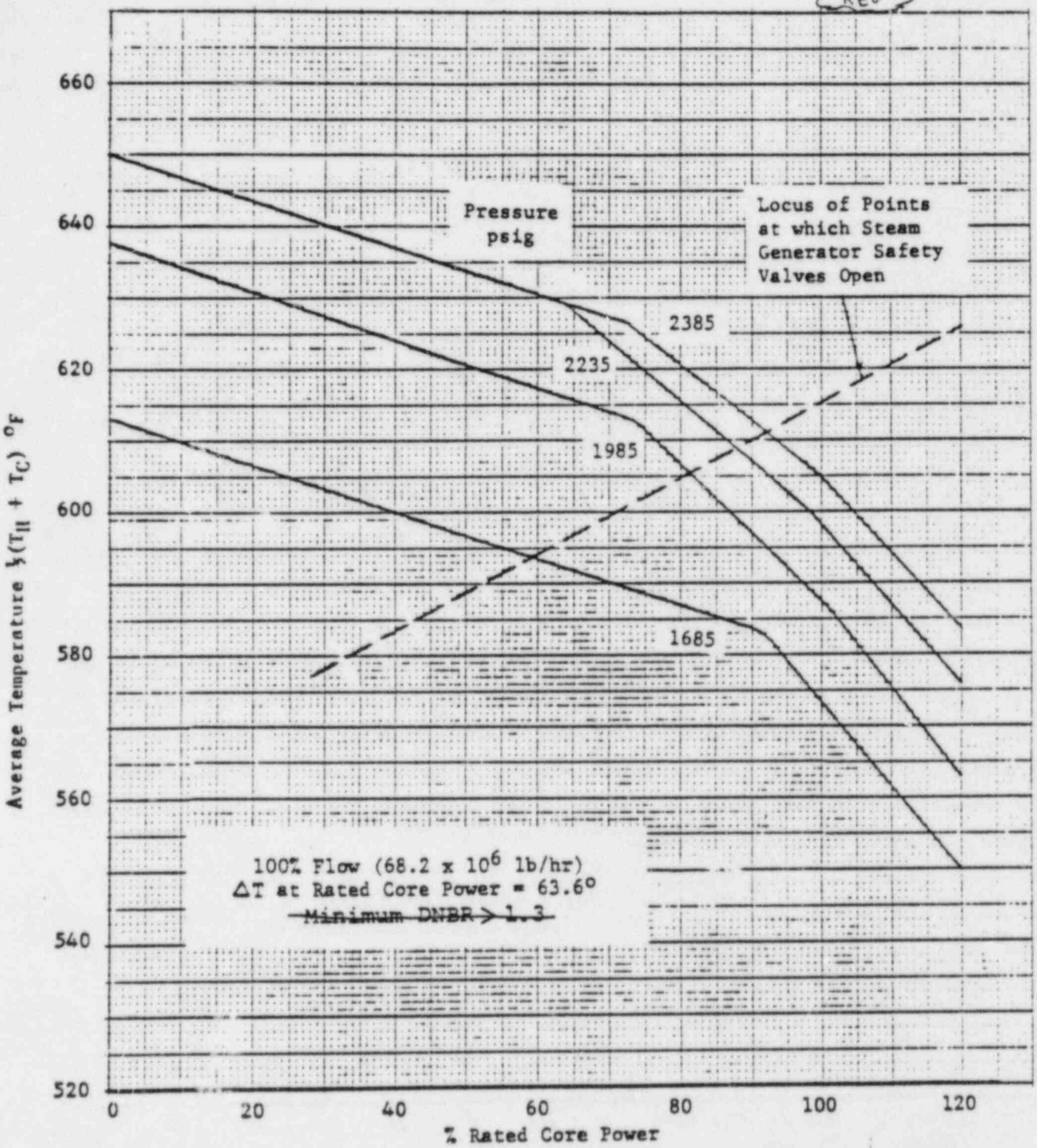
(1) FSAR 14.1.1
(2) FSAR Page 14-3
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.3
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11

for Exxon Nuclear fuel and above 1.17 for Westinghouse fuel.

for Exxon Nuclear fuel or 1.17 for Westinghouse fuel.

Figure TS.2.1-1

REV



SAFETY LIMITS, REACTOR CORE,
THERMAL AND HYDRAULIC
TWO-LOOP OPERATION

FIGURE TS.2.1-1

F. MINIMUM CONDITIONS FOR CRITICALITY

See Exhibit C page 3.1-17 for revised specification

Specification

1. ~~The reactor shall be made critical only at or above the coolant temperature at which the following reactivity coefficient is negative and remains negative for any coolant temperature increase (except during low power physics tests):~~

   ~~(a) Moderator temperature coefficient for a reactor loaded with Westinghouse fuel only.~~

   ~~(b) Isothermal temperature coefficient for a reactor either full or partially loaded with Exxon fuel.~~

2. The reactor shall not be brought to a critical condition until the reactor coolant temperature is higher than that defined by the criticality limit line shown in Figure TS.3.1-1.

3. ~~When the reactor coolant temperature is below the minimum temperature as specified in 1. above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to reactor coolant depressurization.~~

Basis

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. ~~The safety analyses conducted for Prairie Island units with Westinghouse fuel assumed a non positive moderator temperature coefficient.~~ The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during ~~startup physics testing, whereas the moderator temperature coefficient is an inferred parameter determined by subtracting the predicted fuel temperature coefficient from the experimentally determined isothermal temperature coefficient.~~

low power physics tests in order to verify analytical predictions.

Safety analyses verify the acceptability of the isothermal temperature coefficient for limits specified in 3.1.F.1.

isothermal

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive ~~moderator~~ temperature coefficient could exist at beginning of cycle (BOC). ~~For cycles with Exxon fuel, safety analyses are conducted assuming a positive moderator temperature coefficient. These analyses predict the isothermal coefficient to be negative at an all rods out, hot zero power condition.~~ Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F.1 requirements are waived during low power physics tests to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these physics tests. In addition, the strong negative Doppler coefficient [1] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical except as specified in Figure TS.3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters. The pressurizer heater and associated power cables have been sized for continuous operation at full heater power. The shutdown margin in Specification 3.10 precludes the possibility of accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[2]

References:

(1) FSAR Figure 3.2-10
(2) FSAR Table 3.2-1

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall it be heated or maintained above 200°F unless the following conditions are satisfied except as permitted in Specificiation 3.3.A.2.

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 1950 ppm.

   b. Each reactor coolant system accumulator shall be operable when reactor coolant system pressure is greater than 1000 psig.

      Operability requires:
      (1) The isolation valve is open
      (2) Volume is ~~between 1250 and 1282.9~~ 1270 ± 20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of at least 700 psig

   c. Two safety injection pumps are operable except that pump control switches in the control room shall meet the requirements of Section 3.1.G whenever the reactor coolant system temperature is less than MPT.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

   f. Automatic valves, interlocks and piping associated with the above components and required to function during accident conditions, are operable.

   g. Manual valves in the above systems that could (if one is improperly positioned) reduce injection flow below that assumed for accident analyses, shall be blocked and tagged in the proper position for injection. RHR system valves, however, may be positioned as necessary to regulate plant heatup or cooldown rates when the reactor is subcritical. All changes in valve position shall be under direct administrative control.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

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

Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

A. Shutdown Reactivity

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The 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 rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron concentration.

B. Power Distribution Limits

1. At all times, except during low power physics testing, measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, as defined below and in the bases, shall meet the following limits:

$$F_Q^N \times 1.03 \times 1.05 \leq \cancel{(2.32/P)^* \times K(Z) \times BU(E_j)} \quad (2.30/P) \times K(Z)$$

$$F_{\Delta H}^N \times 1.04 \leq \cancel{1.55 \times [1+0.2(1-P)]} \quad 1.60 \times [1+0.3(1-P)]$$

where the following definitions apply:

~~(a)~~ K(Z) is the axial dependence function shown in Figure TS.3.10-5.

~~(b)~~ Z is the core height location.

~~(c) E_j is the maximum pellet exposure in fuel rod j for which the F_Q^N is being measured.~~

~~(d) $BU(E_j)$ is the normalized exposure dependence function for Exxon Nuclear Company fuel shown in Figure TS.3.10-7. For Westinghouse fuel, $BU(E_j)$ = 1.0~~

–

~~(e)~~ P is the fraction of full power at which the core is operating. In the F_Q^N limit determination when $P \leq .50$, set P = 0.50.

~~*(2.21/P) shall be used for Westinghouse assemblies.~~

~~(f)~~ F_Q^N or $F_{\Delta H}^N$ is defined as the measured $\cancel{F_Q^N}$ or $\cancel{F_{\Delta H}^N}$, respectively, with the smallest margin or greatest excess of limit

~~(g)~~ 1.03 is the engineering hot channel factor, F_Q^E, applied to the measured F_Q^N to account for manufacturing tolerance.

~~(h)~~ 1.05 is applied to the measured F_Q^N to account for measurement uncertainty.

~~(i)~~ 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty

2. Hot channel factors, F_Q^N and $F_{\Delta H}^N$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

   (a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

   (b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux difference was last determined, by 10% or more of rated power.

   F_Q^N (equil) shall meet the following limit for ~~the middle axial 80% of the core~~:

   (2.30/P) x K(Z)

   $$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq \cancel{(2.32/P) \times K(Z) \times BU(E_j)}$$

   where V(Z) is defined Figure 3.10-~~X~~ 7 and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N or by 3.33% for each per cent that the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

   (b) If the measured F_Q^N (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

   1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

   2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N (equil) x 1.03 x 1.05 x V(Z) exceeds the ~~$(2.32/P) \times K(Z) \times BU(E_j)$~~ limit.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of rated power.

J. DNB Parameters

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg $\leq 564^{\circ}F$

b. Pressurizer Pressure ≥ 2220 psia*

c. Reactor Coolant Flow $\geq 178{,}000$ gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonomous.

Shutdown Reactivity

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core ≥ 1.30 for Exxon Nuclear fuel and ≥ 1.17 for Westinghouse fuel during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding

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

mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

During operation, the plant staff compares the measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, (described later) to the limit determined in the transient and LOCA analyses. ~~The limiting $F_Q(Z)$ includes measurement, engineering, and calculational uncertainties.~~ The terms on the right side of the equations in section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

~~$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. The maximum value of $F_Q(Z)$ is 2.32/P for the Prairie Island reactors. This value is restricted further by the K(Z) and BU(E_j) functions described below. The product of these three factors is $F_Q(Z)$.~~

The K(Z) function shown in Figure TS.3.10-5 is a normalized function that limits F_Q ~~(Z)~~ axially. ~~for three regions~~. The K(Z) specified for the lowest six (6) feet of the core is arbitrarily flat since the lower part of the core is generally not limiting. Above that region, the K(Z) value is based on large and small break LOCA analyses. ~~$F_Q(Z)$ in the uppermost region is limited to reduce the PCT expected during a small break LOCA since this region of the core is expected to uncover temporarily for some small break LOCAs.~~

~~The BU(E_j) function shown in Figure TS.3.10-7 is a normalized function that limits $F_Q(Z)$ based on exposure dependent analyses for the ENC fuel. These analyses consider pin internal pressure uncertainties, fuel swelling, rupture pressures and flow blockage.~~

F_Q^N is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux on the surface of a fuel rod ~~in the core~~ divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

V(Z) is an axially dependent function applied to the equilibrium measured F_Q^N to bound F_Q^N's that could be measured at non-equilibrium conditions. This function is based on power distribution control analyses that evaluated the effect of burnable poisons, rod position, axial effects, and xenon worth.

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

F_Q^N (equil) is the measured limiting F_Q^N obtained at equilibrium conditions during target flux determination

The 1.05 multiplier accounts for uncertainties associated with measurement of the power distribution with the moveable incore detectors and the use of those measurements to establish the assembly local power distribution.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizational (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 $F_{\Delta H}^N$.

In the specified limit of $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that:

(a) abnormal 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, while 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.

When a measurement of $F_{\Delta H}^N$ is taken, ~~experimental~~ error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector ~~flux mapping system~~ measurement.

Measurements of the hot channel factors are required as part of startup physics tests, at least once each effective 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 identify 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 15

inches from the bank demand position. An accidental misalignment limit of 13 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.

3. The control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ and F^N_Q allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In specification 3.10, F^N_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures 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. Basically control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

F_Q limit

The technical specifications on power distribution control assure that the ~~$F_Q(Z)$ upper bound envelope of 2.32/P times Figures TS.3.10-5 and TS.3.10-7~~ is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

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 noted with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). 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 was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm$5 percent ΔI are permitted from the indicated reference value. Figure TS.3.10-6 shows the allowed deviation from the target flux difference as the function of thermal power.

resulting from operation within the target band. The consequences of being outside the ±5% target band but within the Figure TS.3.10-6 limit for power levels between 50% and 90% has been evaluated and determined to result in acceptable ~~$F_Q(Z)$ values~~ peaking factors. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference In all cases the ±5 percent target band is the Limiting Condition for Operation. Only when the target band is violated do the limits under Figure TS.3.10-6 apply.

If, for any reason, the indicated axial flux difference is not controlled within the ±5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

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 for Exxon fuel and 1.17 for Westinghouse fuel by an automatic protection system. Compliance with operating procedures 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

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and core limits protected per Specification 3.10.E. A quadrant tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are set to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod.

Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During start-up and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present.

If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

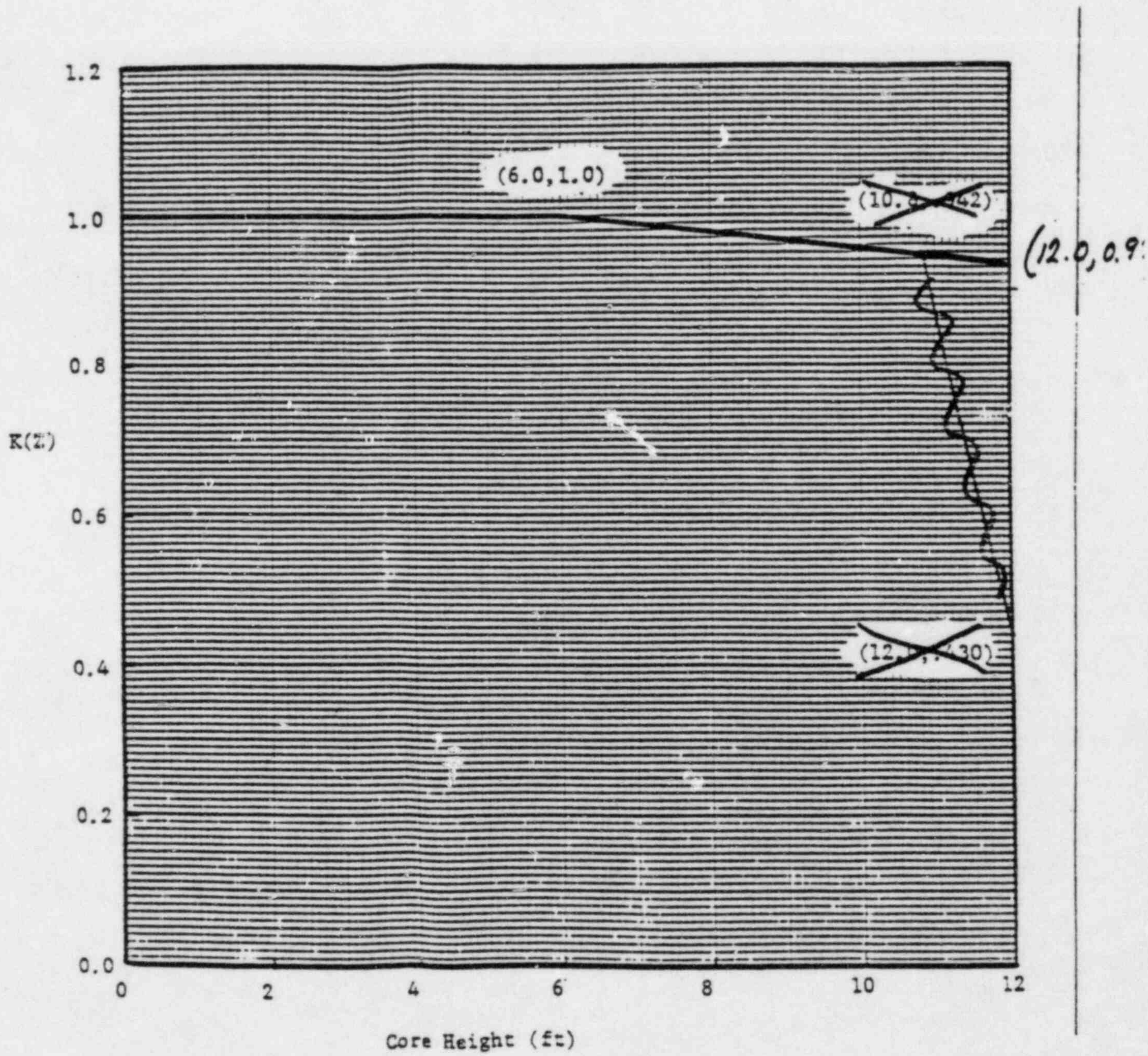
Increased surveillance is required, if the quadrant power tilt monitors are in operable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to quadrant power tilt must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

DNB Parameters

The RCS flow rate, T_{avg}, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the specification value indicated, shutdown is required to investigate adequacy of core cooling during operation.

~~For fuel regions with high burnups, the depletion of fissile nuclides and build-up of fission products greatly reduces power production capability. These combined burnup effects reduce~~ $\cancel{F^N_{\Delta H}}$ ~~sufficiently to cover residual rod bow penalties beyond a region average burnup of 40,000 MWD/MTU.~~

FIGURE TS.3.10-5
REV ~~66 10/3/83~~



HOT CHANNEL FACTOR NORMALIZED

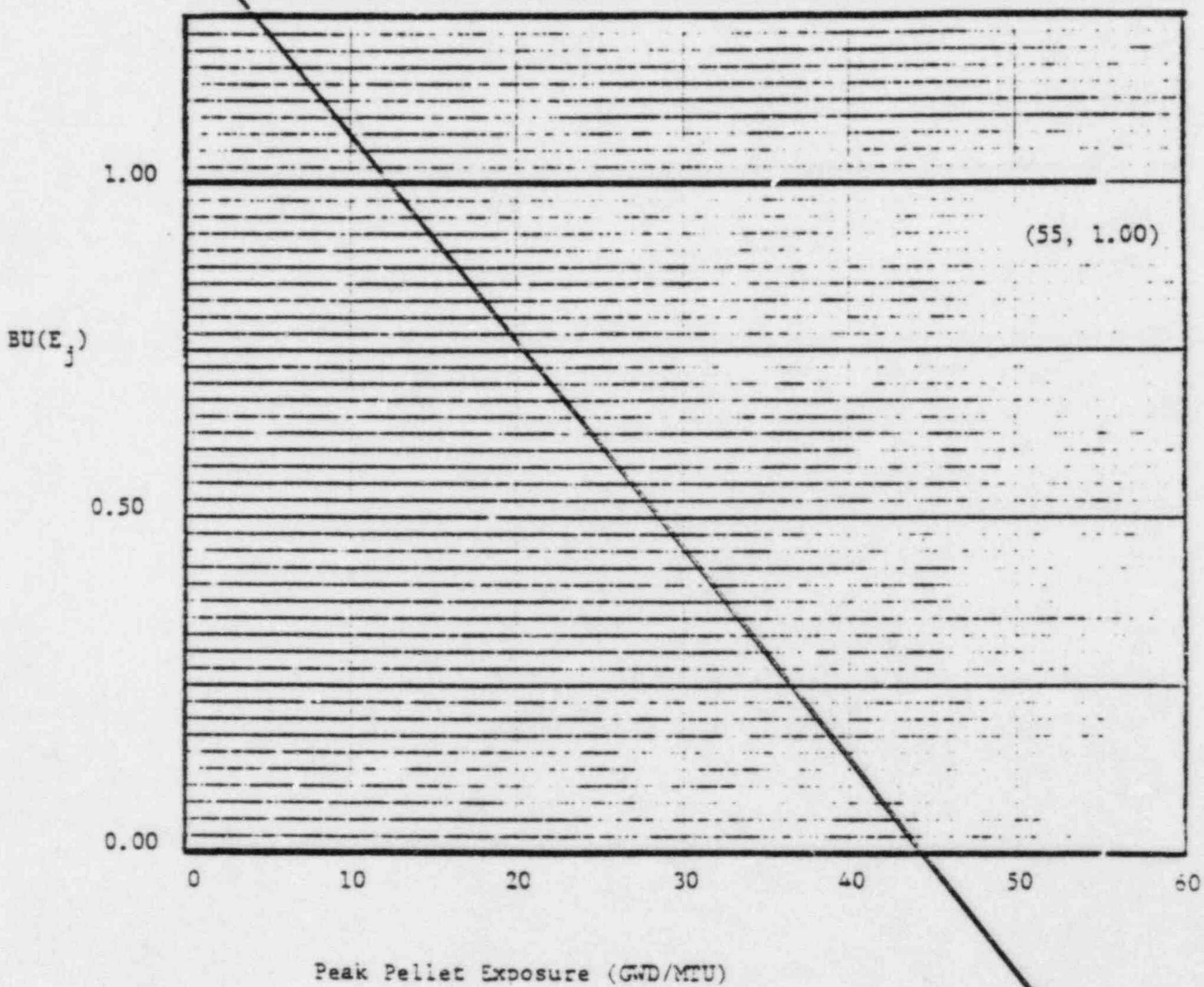
OPERATING ENVELOPE ~~FOR F_Q = 2.32~~

FIGURE TS.3.10-7
REV 67 12/28/83

Normalized Exposure Dependent

Function $BU(E_j)$ for Exxon Nuclear Company Fuel

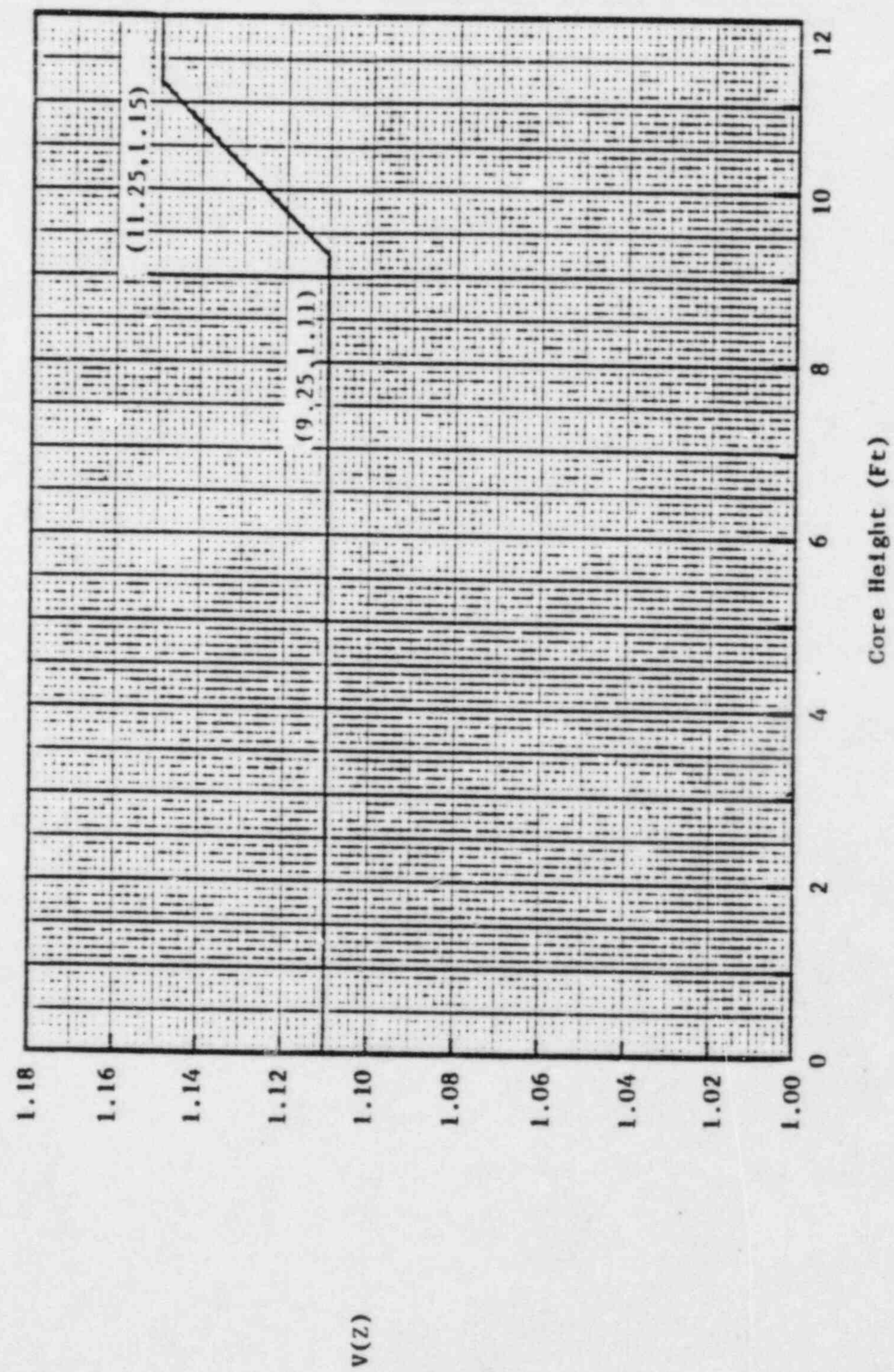
DELETE



TS. 3.10 -7

FIGURE ~~TS.3.10-8~~
REV ~~34 4/20/79~~

Redraw



V(Z) as a Function of Core Height

Exhibit C

Prairie Island Nuclear Generating Plant
License Amendment Request Dated January 13, 1986

Revised Technical Specification Pages

Exhibit C consists of the proposed Technical Specification pages with the changes shown in Exhibit B incorporated. The proposed pages are listed below:

TS-x
TS.2.1-1
TS.2.1-2 *
TS.2.3-6
Figure TS.2.1-1
TS.3.1-17
TS.3.1-18
TS.3.3-1
TS.3.10-1
TS.3.10-2
TS.3.10-8
TS.3.10-9
TS.3.10-10
TS.3.10-11
TS.3.10-13
TS.3.10-17
Figure TS.3.10-5
Figure TS.3.10-7 **

* TS.2.1-3 will be deleted with this change.

** Existing Figure TS.3.10-7 will be deleted by this change. Existing Figure TS.3.10-8 will be renumbered to TS.3.10-7

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, reactor coolant system pressure and coolant temperature during operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure TS.2.1-1. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

Basis

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; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and WRB-1 DNB correlations. The W-3 DNB correlation is used for Exxon Nuclear fuel. The WRB-1 DNB correlation is used for Westinghouse fuel. The W-3 and WRB-1 DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, 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 DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 for the Exxon Nuclear fuel and to 1.17 for the Westinghouse fuel. These limits correspond 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 solid curves of Figure TS 2.1-1 represent the loci of points of thermal power, coolant pressure, and coolant average temperature for which either the coolant enthalpy at the core exit is limiting or the DNB ratio is limiting. For the 1685 psig and 1985 psig curves, the coolant average enthalpy at the core exit is equal to saturated water enthalpy below power levels of 91% and 74% respectively. For the 2235 psig and 2385 psig curves, the coolant average temperature at the core exit is equal to 650°F below power levels of 64% and 73% respectively. For all four curves, the DNBR is limiting at higher power levels. The area of safe operation is below these curves.

The plant conditions required to violate the limits in the lower power range are precluded by the self-actuated safety valves on the steam generators. The highest nominal setting of the steam generator safety valves is 1129 psig (saturation temperature 560°F). At zero power the difference between primary coolant and secondary coolant is zero and at full power it is 50°F. The reactor conditions at which steam generator safety valves open is shown as a dashed line on Figure TS.2.1-1.

Except for special tests, power operation with only one loop or with natural circulation is not allowed. Safety limits for such special tests will be determined as a part of the test procedure.

The curves are conservative for the following nuclear hot channel factors:

$$F_{\Delta H} = 1.60\ [1 + 0.3(1-P)]\ ;\ \text{and}\ F_Q = 2.30$$

Use of these factors results in more conservative safety limits than would result from power distribution limits in Specification TS.3.10.

This combination of hot channel factors is higher than that 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 covered by Specification 3.10. Adverse power distribution factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits specified by Figure TS.3.10-1 assure that the DNB ratio is always greater at part power than at full power.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30 for Exxon Nuclear fuel and less than 1.17 for Westinghouse fuel.

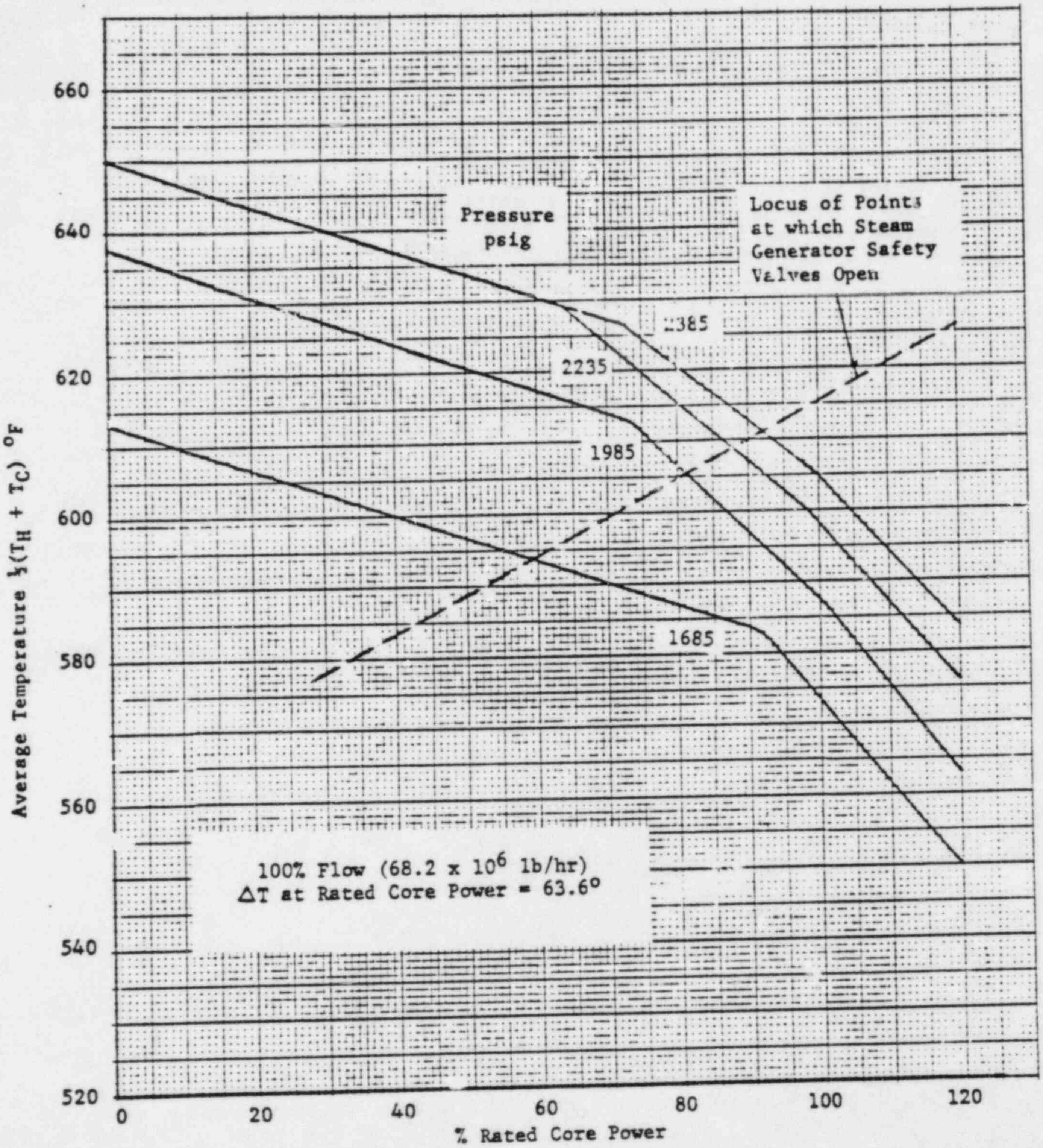
The other reactor trips specified in A.3. above provide additional protection. The trip initiated by steam/feedwater flow mismatch in coincidence with low steam generator water level is designed for protection from a sudden loss of the reactor's heat sink. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of-load transient.

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip compliments the power range nuclear flux high and low trip to assure that the criteria are met for rod ejection from partial power.

The negative power range rate trip provides protection satisfying all IEEE criteria to assure that minimum DNBR is maintained above 1.30 for Exxon Nuclear fuel and above 1.17 for Westinghouse fuel for all multiple control rod drop accidents. Analysis indicates (Section 14.1.3) that in the case of a single rod drop, a return to full power will be initiated by the automatic reactor control system in response to a continued full power turbine load demand and it will not result in a DNBR of less than 1.30 for Exxon Nuclear fuel and 1.17 for Westinghouse fuel. Thus, automatic protection for a single rod drop is not required. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

<u>References</u>:

(1) FSAR 14.1.1
(2) FSAR Page 14-3
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.3
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11



SAFETY LIMITS, REACTOR CORE,
THERMAL AND HYDRAULIC
TWO-LOOP OPERATION

FIGURE TS.2.1-1

For extended optimum fuel burnup it is necessary to either load the reactor with burnable poisons or increase the boron concentration in the reactor coolant system. If the latter approach is emphasized, it is possible that a positive isothermal temperature coeffieient could exist at beginning of cycle (BOC). Safety analyses verify the acceptability of the isothermal temperature coefficient for limits specified in 3.1.F.1. Other conditions, e.g., higher power or partial rod insertion would cause the isothermal coefficient to have a more negative value. These analyses demonstrate that applicable criteria in the NRC Standard Review Plan (NUREG 75/087) are met.

Physics measurements and analyses are conducted during the reload startup test program to (1) verify that the plant will operate within safety analyses assumptions and (2) establish operational procedures to ensure safety analyses assumptions are met. The 3.1.F.1 requirements are waived during low power physics tests to permit measurement of reactor temperature coefficient and other physics design parameters of interest. Special operating precautions will be taken during these physics tests. In addition, the strong negative Doppler coefficient[(1)] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made ciritical except as specified in Figure TS.3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by pressurizer heaters. The pressurizer heater and associated power cables have been sized for continuous operation at full heater power. The shutdown margin in Specification 3.10 precludes the possibility of accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[(2)]

References:

(1) FSAR Figure 3.2-10
(2) FSAR Table 3.2-1

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall it be heated or maintained above 200°F unless the following conditions are satisfied except as permitted in Specification 3.3.A.2.

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 1950 ppm.

   b. Each reactor coolant system accumulator shall be operable when reactor coolant system pressure is greater than 1000 psig.

      Operability requires:
      (1) The isolation valve is open
      (2) Volume is 1270 ± 20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of at least 700 psig

   c. Two safety injection pumps are operable except that pump control switches in the control room shall meet the requirements of Section 3.1.G whenever the reactor coolant system temperature is less than MPT.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

   f. Automatic valves, interlocks and piping associated with the above components and required to function during accident conditions, are operable.

   g. Manual valves in the above systems that could (if one is improperly positioned) reduce injection flow below that assumed for accident analyses, shall be blocked and tagged in the proper position for injection. RHR system valves, however, may be positioned as necessary to regulate plant heatup or cooldown rates when the reactor is subcritical. All changes in valve position shall be under direct administrative control.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

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

Objective

To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during power operation, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

A. Shutdown Reactivity

The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown in Figure TS.3.10-1 under all steady-state operating conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The 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 rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no chenges in xenon or boron concentration.

B. Power Distribution Limits

1. At all times, except during low power physics testing, measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, as defined below and in the bases, shall meet the following limits:

$$F_Q^N \times 1.03 \times 1.05 \leq (2.30/P) \times K(Z)$$

$$F_{\Delta H}^N \times 1.04 \leq 1.60 \times [1+ 0.3(1-P)]$$

where the following definitions apply:

- K(Z) is the axial dependence function shown in Figure TS.3.10-5.
- Z is the core height location.
- P is the fraction of rated power at which the core is operating. In the F_Q^N limit determination when $P \leq .50$, set $P = 0.50$.

- F_Q^N or $F_{\Delta H}^N$ is defined as the measured F_Q or $F_{\Delta H}$ respectively, with the smallest margin or greatest excess of limit.

- 1.03 is the engineering hot channel factor, F_Q^E, applied to the measured F_Q^N to account for manufacturing tolerance.

- 1.05 is applied to the measured F_Q^N to account for measurement uncertainty.

- 1.04 is applied to the measured $F_{\Delta H}^N$ to account for measurement uncertainty.

2. Hot channel factors, F_Q^N and $F_{\Delta H}^N$, shall be measured and the target flux difference determined, at equilibrium conditions according to the following conditions, whichever occurs first:

   (a) At least once per 31 effective full-power days in conjunction with the target flux difference determination, or

   (b) Upon reaching equilibrium conditions after exceeding the reactor power at which target flux differance was last determined, by 10% or more of rated power.

   F_Q^N (equil) shall meet the following limit for the middle axial 80% of the core:

   $$F_Q^N \text{ (equil)} \times V(Z) \times 1.03 \times 1.05 \leq (2.30/P) \times K(Z)$$

   where V(Z) is defined Figure 3.10-7 and other terms are defined in 3.10.B.1 above.

3. (a) If either measured hot channel factor exceeds its limit specified in 3.10.B.1, reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N or by 3.33% for each percent that the measured $F_{\Delta H}^N$ exceeds the 3.10.B.1 limit. Then follow 3.10.B.3(c).

   (b) If the measured F_Q^N (equil) exceeds the 3.10.B.2 limits but not the 3.10.B.1 limit, take one of the following actions:

      1. Within 48 hours place the reactor in an equilibrium configuration for which Specification 3.10.B.2 is satisfied, or

      2. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the measured F_Q^N (equil) x 1.03 x 1.05 x V(Z) exceeds the limit.

F. MINIMUM CONDITIONS FOR CRITICALITY

Specification

1. Isothermal Temperature Coefficient (ITC)

   a. When the reactor is critical, the isothermal temperature coefficient shall be less than 5 pcm/°F with all rods withdrawn, except during low power physics tests and as specified in 3.1.F.1.b and c.

   b. When the reactor is above 70 percent rated thermal power with all rods withdrawn, the isothermal temperature coefficient shall be negative, except as specified in 3.1.F.1.c.

   c. If the limits of 3.1.F.1.a or b cannot be met, Power Operation may continue provided the following actions are taken:

      (1) Establish and maintain control rod withdrawal limits sufficient to restore the ITC to less than the limits specified in Specification 3.1.F.1.a and b above within 24 hours or be in HOT SHUTDOWN within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Figure TS.3.10-2.

      (2) Maintain the control rods within the withdrawal limits established above until a subsequent calculation verifies that the ITC has been restored to within its limit for the all rods withdrawn condition.

      (3) Submit a special report to the Commission within 30 days, describing the value of the measured ITC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the ITC to within its limit for the all rods withdrawn condition.

2. The reactor shall not be brought to a critical condition until the reactor coolant temperature is higher than that defined by the criticality limit line shown in Figure TS.3.1-1.

Basis

At the beginning of a fuel cycle the moderator temperature coefficient has its most positive or least negative value. As the boron concentration is reduced throughout the fuel cycle, the moderator temperature coefficient becomes more negative. The isothermal temperature coefficient is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the isothermal temperature coefficient is the sum of the moderator and fuel temperature coefficients. This coefficient is measured directly during low power physics tests in order to verify analytical predictions.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of rated power.

J. DNB Parameters

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg ≤564°F

b. Pressurizer Pressure ≥2220 psia*

c. Reactor Coolant Flow ≥178,000 gpm

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

Bases

Throughout the 3.10 Technical Specifications, the terms "rod(s)" and "RCCA(s)" are synonomous.

Shutdown Reactivity

Trip shutdown reactivity is provided consistent with plant safety analyses assumptions. One percent shutdown is adequate except for the steam break analysis, which requires more shutdown reactivity due to the more negative moderator temperature coefficient at end of life (when boron concentration is low). Figure TS.3.10-1 is drawn accordingly.

Power Distribution Control

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operations) and II (Incidents of Moderate frequency) events by: (a) maintaining the minimum DNBR in the core ≥1.30 for Exxon Nuclear fuel and ≥1.17 for Westinghouse fuel during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding

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

mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

During operation, the plant staff compares the measured hot channel factors, F_Q^N and $F_{\Delta H}^N$, (described later) to the limits determined in the transient and LOCA analyses. The terms on the right side of the equations in Section 3.10.B.1 represent the analytical limits. Those terms on the left side represent the measured hot channel factors corrected for engineering, calculational, and measurement uncertainties.

F_Q^N is the measured Nuclear Hot Channel Factor, defined as the maximum local heat flux on the surface of a fuel rod divided by the average heat flux in the core. Heat fluxes are derived from measured neutron fluxes and fuel enrichment.

The K(Z) function shown in Figure TS.3.10-5 is a normalized function that limits F_Q axially. The K(Z) specified for the lowest six (6) feet of the core is abritrarily flat since the lower part of the core is generally not limiting. Above that region, the K(Z) value is based on large and small break LOCA analyses.

V(Z) is an axially dependent function applied to the equilibrium measured F_Q^N to bound F_Q^N's that could be measured at non-equilibrium conditions. This function is based on power distribution control analyses that evaluated the effect of burnable poisons, rod position, axial effects, and xenon worth.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local veriations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

The 1.05 multiplier accounts for uncertainties associated with measurement of the power distribution with the moveable incore detectors and the use of those measurements to establish the assembly local power distribution.

F_Q^N (equil) is the measured limiting F_Q^N obtained at equilibrium conditions during target flux determination.

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

When a measurement of $F^N_{\Delta H}$ is taken, measurement error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least once each effective 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 identify 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 15

   inches from the bank demand position. An accidental misalignment limit of 13 steps precludes a rod misalignment greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.

3. The control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial affect which is defined as the difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ and F^N_Q allows for radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In specification 3.10, F^N_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures 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. Basically control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the F_Q limit is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

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 noted with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). 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 was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. Figure TS.3.10-6 shows the allowed deviation from the target flux difference as the function of thermal power.

resulting from operation within the target band. The consequences of being outside the ±5% target band but within the Figure TS.3.10-6 limit for power levels between 50% and 90% has been evaluated and determined to result in acceptable peaking factors. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated axial flux difference. In all cases the ±5 percent target band is the Limiting Condition for Operation. Only when the target band is violated do the limits under Figure TS.3.10-6 apply.

If, for any reason, the indicated axial flux difference is not controlled within the ±5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

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 for Exxon fuel and 1.17 for Westinghouse fuel by an automatic protection system. Compliance with operating procedures 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

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, x-y xenon transient, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.F, and core limits protected per Specification 3.10.E. A quadrant tilt by some other means (x-y xenon transient, etc.) would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are set to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod.

Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present.

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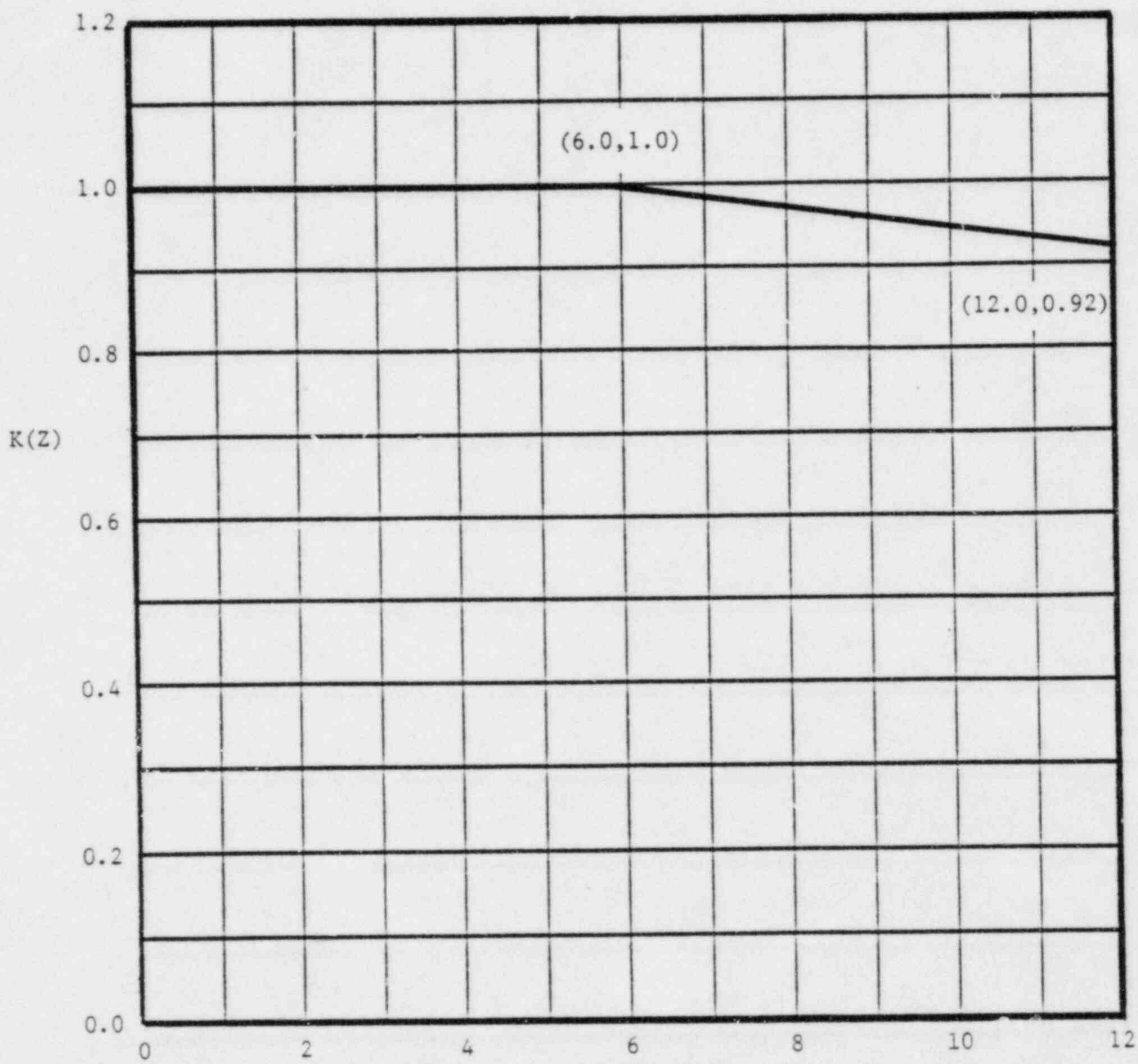
If the rod position deviation monitor and quadrant power tilt monitor(s) are inoperable, the overpower reactor trip setpoint is reduced (and also power) to ensure that adequate core protection is provided in the event that unsatisfactory conditions arise that could affect radial power distribution.

Increased surveillance is required, if the quadrant power tilt monitors are inoperable and a load change occurs, in order to confirm satisfactory power distribution behavior. The automatic alarm functions related to quadrant power tilt must be considered incapable of alerting the operator to unsatisfactory power distribution conditions.

DNB Parameters

The RCS flow rate, T_{avg}, and Pressurizer Pressure requirements are based on transient analyses assumptions. The flow rate shall be verified by calorimetric flow data and/or elbow taps. Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. If a reduction in flow rate is indicated below the specification value indicated, shutdown is required to investigate adequacy of core cooling during operation.

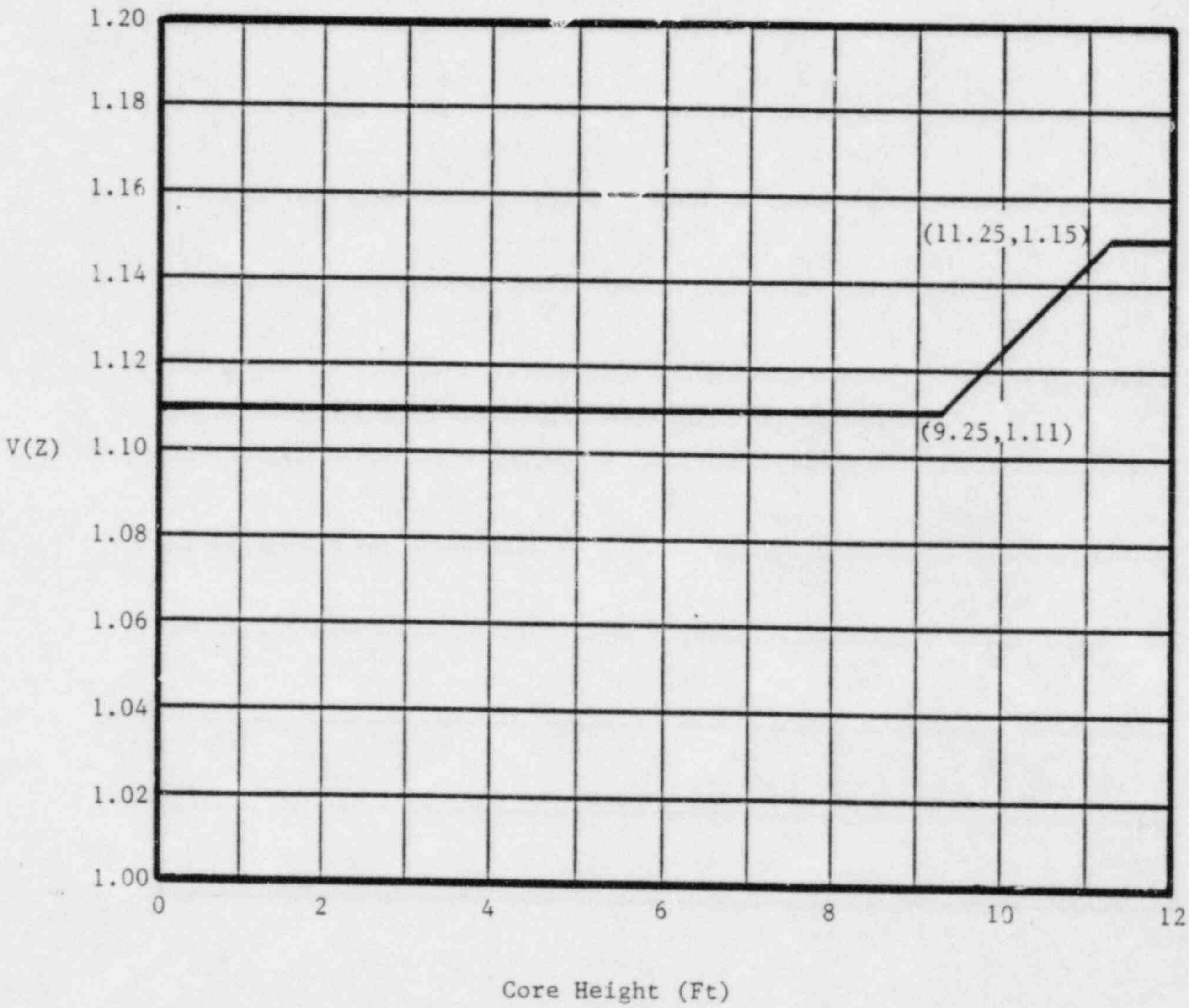
FIGURE TS.3.10-5
REV



HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

FIGURE TS.3.10-7
REV



V(Z) as a Function of Core Height

EXHIBIT D

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Dated January 13, 1986

PRAIRIE ISLAND UNIT 1 WESTINGHOUSE OFA TRANSITION RELOADS

Information addressing eleven plant specific items given on page 3 of NRC cover letter for NRC SER on WCAP-9500 (May 22, 1981)

1. For those plants using the Improved Thermal Design Procedure (ITDP), the conditions listed in the safety evaluation must be addressed and satisfied.

   Response: This item is not applicable to the Prairie Island Unit 1 since the ITDP methodology is not used.

2. A discussion in the Basis of Technical Specifications of any generic or plant-specific margins that have been used to offset the reduction in DNBR due to rod bowing.

   Response: The effects of fuel rod bowing is included explicitly in the transient analysis. A rod bow penalty is applied to the calculated MDNBR for each transient before comparison to acceptance criteria. No generic or plant specific margins have been used to offset rod bow DNBR reductions.

3. A declaration in the Technical Specifications that prohibits N-1 loop operation unless adequately justified in the plant-specific analysis.

   Response: Current Technical Specifications require an automatic reactor trip with less than 90% flow in either loop below 10% power.

4. Frequency and description of rod worth tests that would detect gross losses of reactivity worth from boron-containing control rods.

   Response: This item is not applicable to Prairie Island Unit 1 since boron-containing control rods are not used.

5. Confirmation that the predicted cladding collapse time exceeds the expected lifetime of the fuel.

   Response: The Prairie Island Unit 1 Westinghouse OFA fuel is designed utilizing the approved Westinghouse fuel performance model [(a)] and approved clad flattening model [(b)]. The OFA fuel is designed so that the calculated fuel rod clad flattening time is greater than the maximum planned fuel irradiation time in the reactor.

   (a) Miller, J V, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976.

   (b) George, R A, (et. al.), "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

6. Supplemental ECCS calculations using NRC-supplied LOCA cladding models, until a generic resolution of this issue is obtained.

   Response: This item is not applicable since a generic resolution of NRC concerns on clad ballooning and assembly flow blockage has been obtained. The Prairie Island LOCA analyses of the Westinghouse OFA fuel use approved models which incorporate the generic resolution.

7. A determination that the appropriate seismic and LOCA forces are bounded by the cases considered in WCAP-9401 or additional analyses.

   Response: The impact of LOCA and seismic forces on the integrity of OFAs has been analyzed for a homogeneous OFA core and mixed cores of OFAs and Exxon fuel assemblies. Results confirm the integrity of the OFAs since the grids do not crush and the stresses in the OFA components are within acceptable limits. The analyses used the generic methods of Reference (a) which received NRC approval via Reference (b).

   (a) Letter from E P Rahe (Westinghouse) to J R Miller (NRC) dated March 19, 1982, NS-EPR-2573, Subject: WCAP-9500 and WCAP-9401/9402 NRC SER Mixed Core Compatability Items.

   (b) Letter from C O Thomas (NRC) to E P Rahe (Westinghouse), dated November 12, 1982, Subject: Supplemental Acceptance Number 1 for Referencing of Licensing Topical Report WCAP-9500-A.

8. A description of plans for on-line fuel system monitoring:

   Response: Prairie Island Unit 1 does not have an on-line fuel system monitor other than gross fuel failure detection with the letdown line liquid process monitors. The current method of fuel monitoring by collecting periodic primary coolant samples to assess fuel integrity is sufficient.

9. A description of plans for post-irradiation poolside surveillance of fuel.

   Response: No special surveillance requirements are necessary since Prairie Island Unit 1 is not a lead plant for using the 14x14 OFAs. The Point Beach units are the lead plants utilizing standard 14x14 OFAs. Therefore, only the normal visual surveillance of representative sample of irradiated OFAs is planned during refueling shutdowns of Prairie Island Unit 1.

10. For transients analyzed to determine fuel failure, DNBR as function of time (NUREG-1.70 requirement).

    Response: The DNBR as a function of time is calculated and the results are published in the Final Reload Design Report (Reload Safety Evaluation) for each cycle. The results of this evaluation are contained in Exhibit H.

11. Initial fuel conditions (i.e., stored energy or centerline temperature) utilized in the transient and accident analyses (as per NUREG-1.70 requirements).

    Response: The initial fuel temperatures used in the NSP transient analyses were supplied by the fuel vendor (Westinghouse) using the PAD Code (WCAP 8720). The initial fuel temperatures used in LOCA analyses performed by Westinghouse used the PAD Code (WCAP 8720 Addn 2).