December 2, 1998

Distribution w/encls:

Docket File	GHill (2)
PUBLIC	WBeckner, TSB
PD3-3 Reading	EGA1
ACRS	OGC
GGrant, RIII	RBellamy
THarris (TLH3, SE only)	

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
P. O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO.142 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1557)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 142 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated April 15, 1998, as supplemented by letters dated July 27, 1998, August 13, 1998, two different letters dated September 28, 1998, and by letter dated November 24, 1998.

The amendment revises the power distribution peaking factor limits and limits operating parameters related to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) in support of Cycle 23 fuel and reload changes. A change associated with the fuel and reload changes, is the removal, from the current licensing basis, of the fuel pool turbine missile hazards analysis.

As proposed in your letter dated November 24, 1998, the amendment conditions the license for a maximum rod average burnup of 60 GWD/MTU, for any rod, until such time as the staff has completed an environmental assessment supporting a greater limit.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original signed by:
William O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305
Enclosures: 1. Amendment No.142 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\KEWAUNEE\PA152.AMD *See previous concurrence

OFFICE	PD31:PM	E	PD32:LA	E	PERB:BC	OGC	PD31:PD	E
NAME	WLong		EBarnhill		CMiller*	MYoung*	CCarpenter	
DATE	12/ 1/98		12/1/98		11/10/98	12/1/98	12/1/98	

OFFICIAL RECORD COPY

December 2, 1998

Distribution w/encls:

Docket File	GHill (2)
PUBLIC	WBeckner, TSB
PD3-3 Reading	EGA1
ACRS	OGC
GGrant, RIII	RBellamy
THarris (TLH3, SE only)	

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
P. O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO.142 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1557)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 142 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated April 15, 1998, as supplemented by letters dated July 27, 1998, August 13, 1998, two different letters dated September 28, 1998, and by letter dated November 24, 1998.

The amendment revises the power distribution peaking factor limits and limits operating parameters related to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) in support of Cycle 23 fuel and reload changes. A change associated with the fuel and reload changes, is the removal, from the current licensing basis, of the fuel pool turbine missile hazards analysis.

As proposed in your letter dated November 24, 1998, the amendment conditions the license for a maximum rod average burnup of 60 GWD/MTU, for any rod, until such time as the staff has completed an environmental assessment supporting a greater limit.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:
William O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305
Enclosures: 1. Amendment No.142 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\KEWAUNEE\PA152.AMD *See previous concurrence

OFFICE	PD31:PM	E	PD32:LA	E	PERB:BC	OGC	PD31:PD	E
NAME	WLong		EBarnhill		CMiller*	MYoung*	CCarpenter	
DATE	12/ /98		12/ /98		11/10/98	12/1/98	12/ /98	

OFFICIAL RECORD COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 2, 1998

Mr. M. L. Marchi
Site Vice President-Kewaunee Plant
Wisconsin Public Service Corporation
P.O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1557)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 142 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated April 15, 1998, as supplemented by letters dated July 27, 1998, August 13, 1998, two different letters dated September 28, 1998, and by letter dated November 24, 1998.

The amendment revises the power distribution peaking factor limits and limits operating parameters related to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) in support of Cycle 23 fuel and reload changes. A change associated with the fuel and reload changes, is the removal, from the current licensing basis, of the fuel pool turbine missile hazards analysis.

As proposed in your letter dated November 24, 1998, the amendment conditions the license for a maximum rod average burnup of 60 GWD/MTU, for any rod, until such time as the staff has completed an environmental assessment supporting a greater limit.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

William O. Long

William O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 142 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

M. L. Marchi
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

cc:

Foley & Lardner
ATTN: Bradley D. Jackson
One South Pinckney Street
P.O. Box 1497
Madison, WI 53701-1497

Chairman
Town of Carlton
Route 1
Kewaunee, WI 54216

Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, WI 54216

Attorney General
114 East, State Capitol
Madison, WI 53702

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #1, Box 999
Kewaunee, WI 54216-9511

Regional Administrator - Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4531

James D. Loock, Chief Engineer
Public Service Commission
of Wisconsin
P. O. Box 7854
Madison, WI 53707-7854

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 142
License No. DPR-43

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

A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated April 15, 1998, as supplemented by letters dated August 13, 1998, September 28, 1998, and November 24, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, Facility Operating License No. DPR-43 is amended as follows:

   A. Changes are made to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) is hereby amended to read as follows:

      (2) <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 142, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

   B The following condition is added to 2.C:

      (5) <u>Fuel Burnup</u>

      The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

Wlilliam O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 2, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 142

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
ii	ii
B2.1-1	B2.1-1
B2.1-2	
Figure TS 2.1-1	Figure TS 2.1-1
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3
3.10-4	3.10-4
3.10-5	3.10-5
3.10-6	3.10-6
3.10-7	3.10-7
3.10-8	3.10-8
3.10-9	3.10-9
	3.10-10
B3.10-1	B3.10-1
B3.10-2	B3.10-2
B3.10-3	B3.10-3
B3.10-4	B3.10-4
B3.10-5	B3.10-5
B3.10-6	B3.10-6
B3.10-7	B3.10-7
B3.10-8	B3.10-8
B3.10-9	B3.10-9
Figure TS 3.10-1	Figure TS 3.10-1
Figure TS 3.10-2	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.

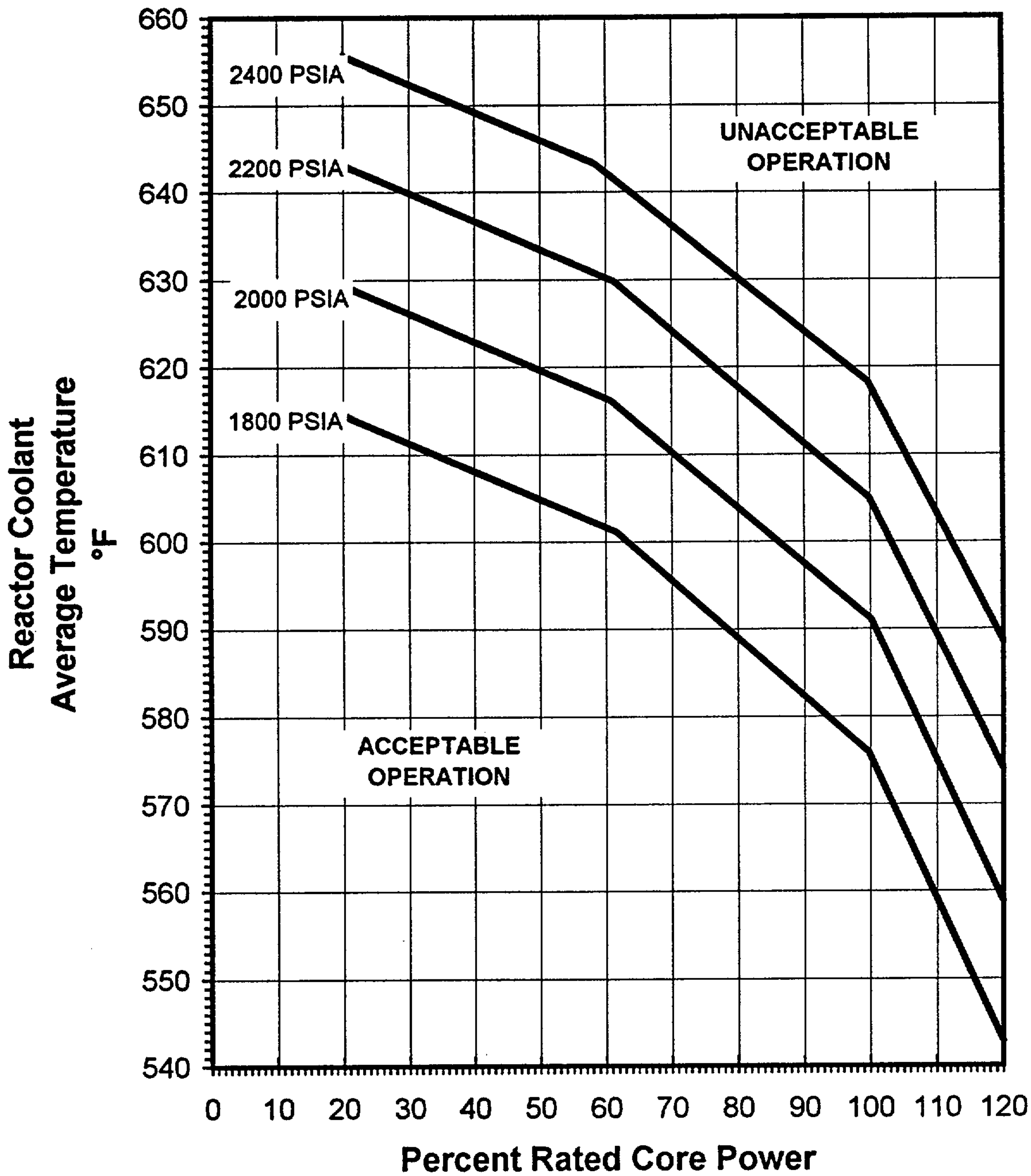
The curves are based on the nuclear hot channel factor limits of TS 3.10.b.

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. 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

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.35)/P \times K(Z)$ for $P > .5$ [Hvy]

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.70) \times K(Z)$ for $P \leq .5$ [Hvy]

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z)$ for $P > .5$ [Std]

   $F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z)$ for $P \leq .5$ [Std]

   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

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

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

   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:

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

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

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 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. 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 3.10.1, 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 less than the 2200°F limit.

The $F_Q^N(Z)$ limits of TS 3.10.b.1.A 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).

<u>$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor</u>

$F_{\Delta H}^N$, 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 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]

---

[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.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 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 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 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.

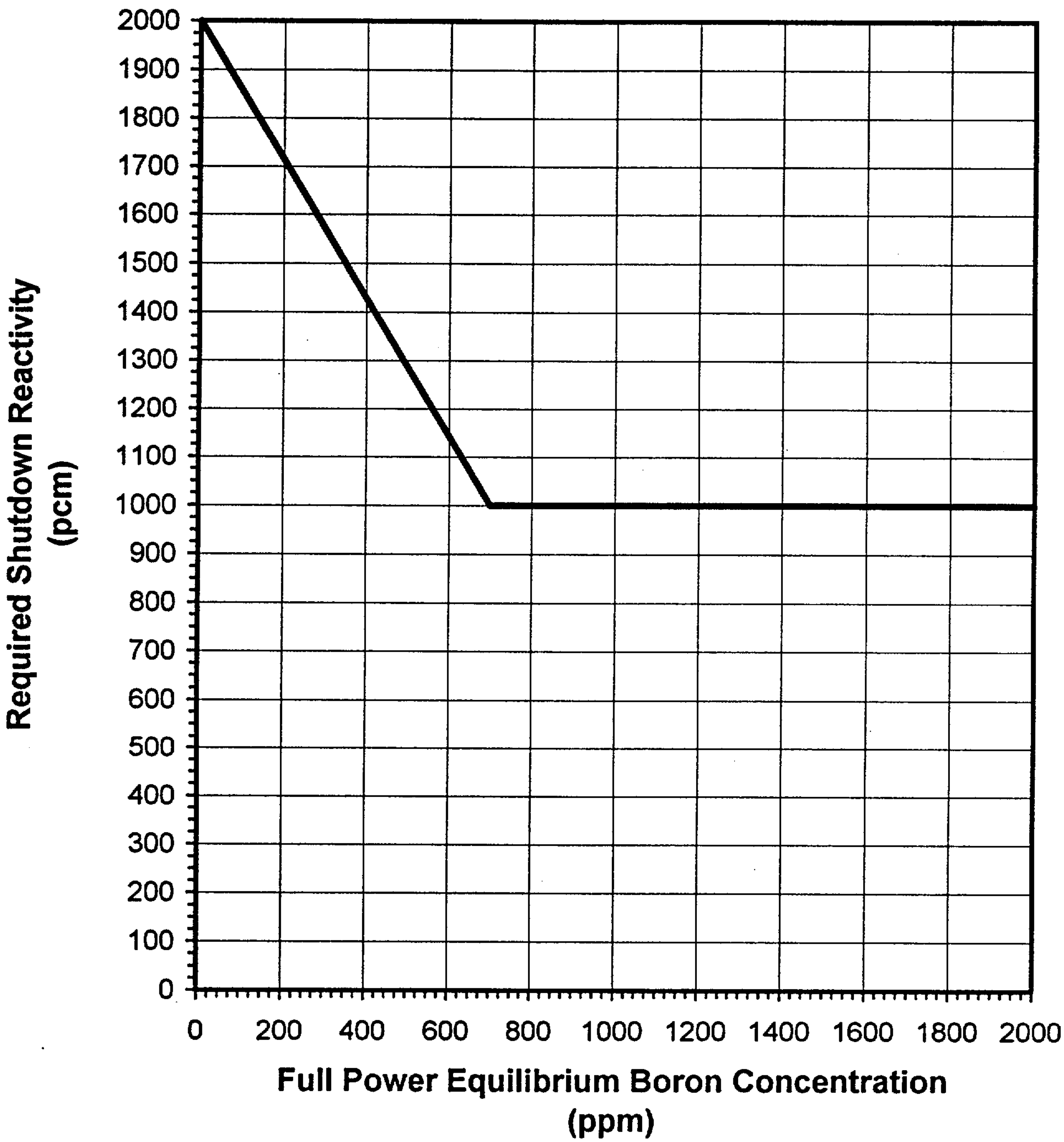
DNBR Parameters (TS 3.10.n)

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.

Two departure from nucleate boiling ratio (DNBR) correlations used in the safety analyses: the high thermal performance (HTP) DNBR correlation and the W-3 DNBR correlation. The HTP correlation applies to Siemens Power Corporation (SPC) fuel with HTP spacers. The W-3 correlation is used for the analysis of non-HTP fuel designs and for all fuel designs at low pressure and temperature conditions (e.g., the conditions analyzed during a main steam line break accident). Both DNBR correlations have been qualified and approved for application to Kewaunee. The minimum DNBR limits are 1.14 for the HTP correlation and 1.30 for the W-3 correlation.

FIGURE TS 3.10-1

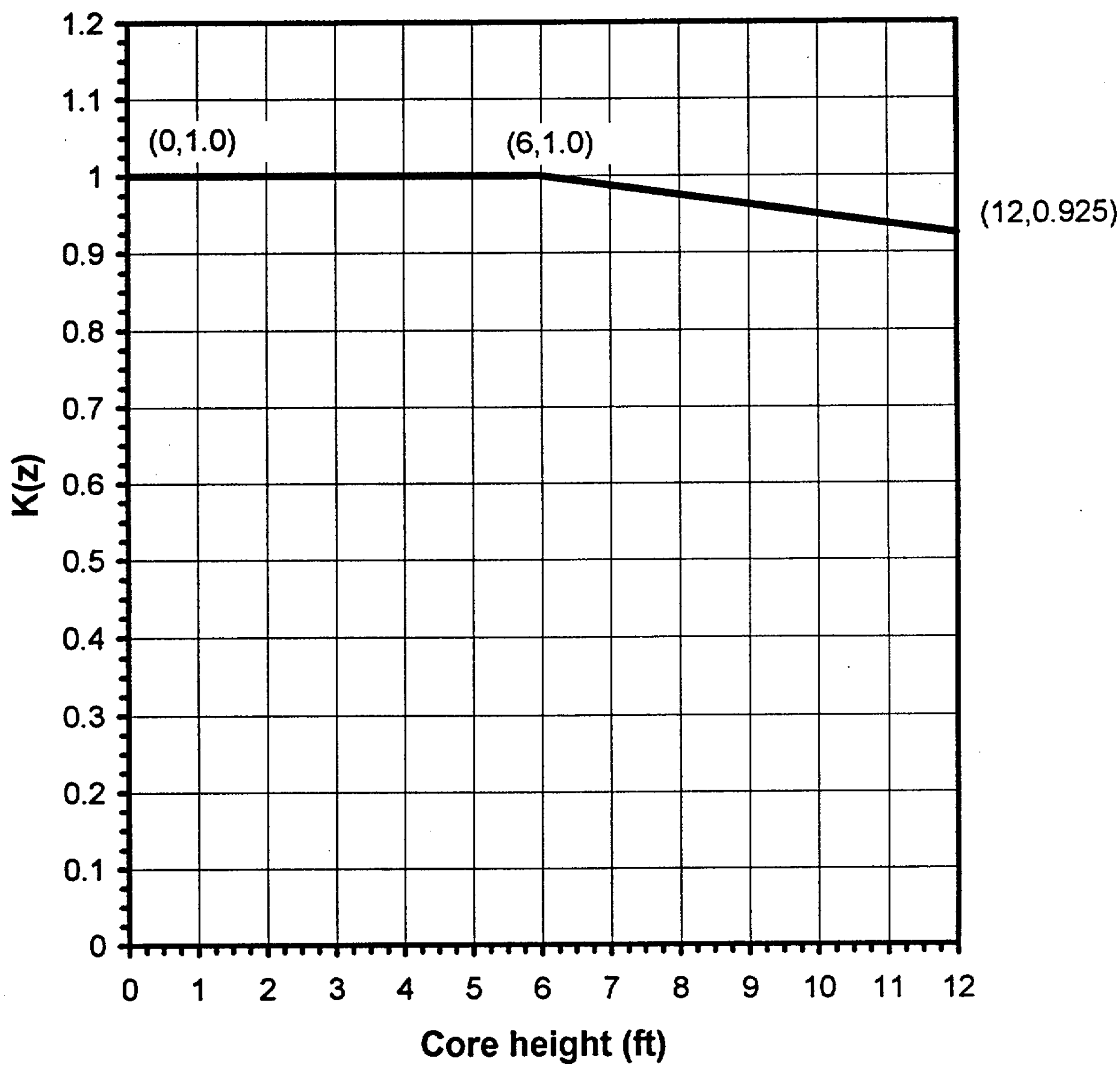
Required Shutdown Reactivity
vs.
Reactor Boron Concentration



 Amendment No. 142

FIGURE TS 3.10-2

Hot Channel Factor
Normalized Operating Envelope



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated April 15, 1998, as supplemented by letters dated July 27, 1998, August 13, 1998, two letters dated September 28, 1998, and a letter dated November 24, 1998, the Wisconsin Public Service Corporation, licensee for the Kewaunee Nuclear Power Plant (KNPP), requested an amendment to the facility Technical Specifications. The proposed amendment would revise the power distribution peaking factor limits and limits operating parameters related to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) in support of Cycle 23 fuel and reload changes. A change associated with the fuel and reload changes is the removal, from the current licensing basis, of the fuel pool turbine missile hazards analysis. The supplemental submittals provided clarifying information that did not affect the initial no significant hazards consideration.

2.0 DISCUSSIONS AND EVALUATIONS

2.1 REVISED PEAKING FACTOR LIMITS

2.1.1 USE OF NEW FUEL DESIGN

Currently, the KNPP plant is operating with Siemens (SPC) Standard fuel. Future core loadings will consist of a mixed core of SPC Standard and SPC Heavy fuel in the Cycle 23 transition core to eventually an all SPC Heavy fueled core. The proposed TSs account for the new fuel design and involve changes to the power distribution peaking factors and limits operating parameters related to Minimum Departure from Nucleate Boiling Ratio specified in TS 2.1 "Safety Limits," and TS 3.10, "Control Rod and Power Distribution Limits." In support of the reload and TS changes application, the licensee provided the results of transient and loss-of-coolant accident (LOCA) analyses (Ref. 1) for the staff to review and approve.

2.1.2 EVALUATION

The objective of the staff review was to (1) confirm that the licensee performed safety analyses with acceptable methods, (2) verify that the analytical results meet the required acceptance criteria, and (3) insure that the proposed TSs appropriately reflect the results of the acceptable safety analyses. The staff reviewed the request for the reload applications and TS changes with the supporting analyses, the licensee's responses (Refs. 4 and 5) to the staff request for additional information (RAI), and the revised TS changes (Ref. 9) resulting from the staff review questions. This evaluation encompasses the staff's review for the following areas: (1) mechanical analyses of the fuel design, (2) analytical methods, (3) the safety departure from nucleate boiling ratio (DNBR) limits, (4) transient analyses, (5) LOCA analyses, and (6) the proposed TS changes.

2.1.2.1 Mechanical Analyses of the Fuel Design

The major design features of SPC Heavy fuel relative to the current SPC Standard fuel include the following: (1) a larger pellet diameter, (2) a thinner cladding wall and a smaller diametric pellet/cladding gap, (3) a higher fill pressure in the pellet/cladding gap, and (4) High-Thermal-Performance (HTP) spacer grids. Accounting for the effects of the design features of the SPC fuel, the licensee performed mechanical analyses of the fuel design for the existing SPC Standard and new SPC Heavy fuels with fuel burnups extended to 62 GWD/MTU average in the peak rod. In response to the RAI, the licensee submitted the results of the analyses (Attachments 2 and 3 to reference 5) for the staff to review and approve. The methods used for the mechanical analyses are described in the SPC topical reports, XN-NF-82-06, Revision 1, ANF-88-133 and ANF-88-060. These methods were previously approved (References 6 and 7) by the NRC and are acceptable for use at KNPP. As a result of the staff review of the results of the mechanical analyses, the staff finds that the mechanical analyses satisfy the design criteria approved by the staff (References 6, 7 and 10) for the SPC fuel. The approved design criteria provide reasonable assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, and (3) coolability is always maintained, and meet the guidance of Section 4.2 of the SRP. Therefore, the staff concludes that the mechanical analyses of the fuel design are acceptable.

2.1.2.2 Analytical Methods

The analytical methods used for the transient and accident analyses to support the reload applications and TS changes are normally reviewed on a generic basis. The methods include the following computer codes:

- DYNODE: The DYNODE code provides a simulation of the system response and calculates system parameters such as core power, RCS flow, primary and secondary temperatures and pressures during a transient. The code had been reviewed and approved (Ref. 2) by NRC for use in the design basis transient analysis at the KNPP for licensing applications.

- VIPRE: The VIPRE code provides a simulation of the hot channel thermal-hydraulic analysis and determines the minimum DNBR using the approved correlations. The code had been reviewed and approved (Ref. 2) by NRC for use in the design basis transient analysis at the KNPP for licensing applications and is acceptable for use at KNPP.

- NOTRUMP: The NOTRUMP code consists of modeling features that meet the requirements of Appendix K to 10 CFR Part 50. As documented in WCAP-10079-A and WCAP-10054-A, NRC had previously approved the NOTRUMP code for the small break LOCA analysis.

- WCOBRA/TRAC: As documented in WCAP-10924-A, this code had been previously approved by NRC for use in the large break LOCA analysis for the Westinghouse two-loop plants such as the KNPP plants.

In response to the staff request, the licensee evaluated its compliance with the conditions specified in the safety evaluations (SEs) for methodologies referenced in the submittal (Reference 1), and determined that the SE conditions for the methodologies have been met (References 4 and 5). Accordingly, the staff concludes that the licensee adequately addresses the staff concern relating to conformance to SEs conditions.

2.1.2.3 Safety DNBR Limits

The safety DNBR limit has been imposed to assure that there is at least a 95% probability at a 95% confidence level that the hot rod in the core does not experience a departure from nucleate boiling (DNB) during transients. For SPC Heavy fuel, the licensee calculated DNBRs using the HTP correlation. For the steam line break event applying to SPC Heavy fuel that results in thermal-hydraulic conditions outside the range analyzed for the HTP correlation, and for all transient analyses applying to SPC Standard fuel, the licensee uses the W-3 correlation for DNBR calculations. The safety DNBR limit is 1.14 for the HTP correlation and 1.3 for the W-3 correlation. Since the safety limits for both HTP correlation (Reference 3) and W-3 CHF correlation (Reference 8) were previously approved by NRC for the licensing calculations, the staff concludes that the use of the approved CHF correlations with the associated safety DNBR limits to assess the fuel failure during the transients is acceptable.

2.1.2.4 Transient Analyses

The licensee presented the results of reanalyses for the transients in Attachment 5 to Reference 1. The licensee identified the limiting case for each event category discussed in Chapter 14 of the FSAR and evaluated the effects of changes in values of plant parameters (such as reduction in the RCS flow and an increase in the core peaking factors) and new fuel design features on plant transients. The licensee identified and reanalyzed the cases that would be affected by the reload fuel design and plant operating conditions, and provided the results for limiting cases for each event category for the staff to review and approve. The events analyzed are:

1. Uncontrolled CEA Withdrawal from a Subcritical Condition
2. Uncontrolled CEA Withdrawal at Power

3. Control Rod Misalignment
4. Uncontrolled Boron Dilution
5. Startup of an Inactive Loop
6. Feedwater System Malfunction
7. Excess Load Increase
8. Loss of Flow
9. Loss of Load
10. Loss of Feedwater
11. Locked Rotor
12. Main Steamline Break
13. Control Rod Ejection

The values of the plant conditions assumed in the analyses are summarized on page 5 of Attachment 5 to Reference 1. The assumed values reflect the limiting plant conditions. For example, the initial power is at 102% of the licensed power; the values are 1.70 for the nuclear enthalpy rise channel factor and 2.50 for the heat flux hot channel factor and the RCS flow is 83,500 gpm per loop. In the analyses, a full core of SPC Heavy fuel was assumed. The licensee summarized the analytical results for the most limiting cases in Tables 1 and 2 of Attachment 5 to Reference 1.

As a reactor is reloaded with SPC Heavy fuel prior to a full core of SPC Heavy fuel, there are transitional cycles when both SPC Heavy fuel and remaining fuel (SPC Standard fuel) will co-exist in the core. The differences in the adjacent fuel assemblies in the hydraulic resistance characteristics such as spacer grid designs result in local hydraulic mismatches. Such a hydraulic mismatch results in localized flow redistribution due to the open core configuration. While beneficial to SPC Heavy and Standard fuel with HTP grids due to lower grid resistance, the interbundle cross flow is detrimental to SPC Standard fuel with Bi-M grids. During the course of the review, the staff asked the licensee to calculate a mixed core DNBR penalty for transitional mixed core configurations to account for the detrimental effect of interbundle cross flow applicable to SPC Standard fuel with Bi-M grids. In response, the licensee performed thermal hydraulic analyses for using the NRC-approved VIPRE code. The results of the mixed core analyses (Ref. 4) showed that SPC Standard fuel with Bi-M grids experienced a 3.3% reduction in DNBR. The licensee applied the calculated mixed core DNBR penalty of 3.3% in the transient analyses to the approved W-3 correlation DNBR limit for SPC Standard fuel with Bi-M grids. The results (Table 2 of Ref. 4) show that the DNBR safety limit is met and the Bi-M grid standard fuel assemblies continue to have full power peaking factor limit of 1.55 F_H and 2.28 F_Q (100 % power limits) that are included in the current TSs.

Since the licensee uses the approved codes to perform transient analyses, the values used for the input parameters are conservative, and the results show that the acceptance criteria specified in Section 15 of the Standard Review Plan (SRP) for each transient category are met, the staff concludes that the transient analyses are acceptable.

In the analysis of the rod ejection accident, the licensee considered four cases including beginning-of-cycle at full-power and zero-power, and end-of-cycle at full-power and zero-power. For all cases, the calculated radial average fuel enthalpy is less than 182 calories per gram, which is less than the acceptance criterion of 280 calories per gram specified in Regulatory

Guide (RG) 1.77. In addition, the pressure surge resulting from the rod ejection accident does not exceed the reactor coolant system emergency limits (Service Level C) and thus satisfies the guidance of RG 1.77.

Recent experimental data showed failure of high burnup fuels at lower values of enthalpy than the fuel failure enthalpy limits specified in RG 1.77. However, generic analyses performed by the industry that assumed low enthalpy for fuel failures showed that the radiological consequences of rod ejection accidents meet the acceptance criteria specified in SRP Section 15.4.8 (Appendix A). The generic analyses are predicated on conservative treatment of the experimental fuel data applied to existing and planned cores operating within approved burnup limits for PWRs. In addition, there is broad agreement among the staff, the industry, and the international community that burnup degradation in the margin to low-enthalpy fuel failure is likely to be regained by application of more detailed 3-dimensional analysis methods of the fuel response to rod ejection accidents. Therefore, the staff concludes that, although the RG 1.77 fuel failure enthalpy limits may not be conservative, the generic analyses provide reasonable assurance that radiological consequences of rod ejection accidents will not violate the acceptance criterion in SRP Section 15.4.8 for the PWR cores operating within the current NRC approved burnup limits (62 GWD/MTU average in the peak rod for the SPC fuel.) The staff will not approve further extension of burnup limits until additional experimental information on fuel behavior is available to demonstrate that the fuel cladding will satisfy the regulatory acceptance criteria used in the rod ejection analyses for licensing applications.

2.1.2.5 Loss-of-Coolant-Accident (LOCA) Analysis

The large break (LB) LOCA analyses (Attachment 4 to Reference 1) were performed with a full licensed power, a total peaking factor of 2.35, a radial peaking factor of 1.70, and a RCS flow of 83,400 gpm per loop. The core bypass flow of 7% was assumed. A break spectrum sensitivity study for LBLOCAs identified that the limiting case, resulting in a highest peak cladding temperature (PCT), is the 0.6 double-ended-cold-leg-guillotine (DECLG) break for super bounded and 0.4 DECLG break for Appendix K LOCA analyses. The licensee also performed a sensitivity study to assess the mixed core effect and determined that the mixed core PCT penalty is small when the non-feed SPC Standard fuel is in the core with SPC Heavy fuel. The results of the LBLOCA analysis show the calculated PCT of 1872 °F, maximum cladding oxidation of 3.3% of the total cladding thickness and metal-water reaction of less than 0.0033% of the total amount of metal in the core.

The small break (SB) LOCA analyses were done assuming a full licensed power, a total peaking factor of 2.50, a radial peaking factor of 1.70 and a reduced RCS flow of 83,400 gpm per loop (from 85,000 gpm per loop). The licensee also assumed a tube plugging of 30% (increased from 25%) in each SG. SBLOCA analyses for various break sizes were performed. The results (Ref. 4) show that the limiting case is 3-inch diameter cold leg break. The limiting case results in a highest PCT of 1041 °F for the mixed core and 932°F for a full core with SPC Heavy fuel .

For both the SBLOCA and LBLOCA analyses, the limiting cases do not exceed the acceptance criteria of 10 CFR 50.46: PCT of 2200 °F, maximum cladding oxidation of 0.17 of the total cladding thickness and metal-water reaction of less than 0.01 of the total amount of metal in the core. Since the licensee uses the approved codes, the values used for the input parameters are conservative, and the results meet the 10 CFR 50.46 acceptance criteria, the staff concludes that the SBLOCA and LBLOCA analyses are acceptable.

2.1.2.6 Technical Specification Changes

The licensee submitted a request for Technical Specifications (TS) changes to allow operation of the KNPP Cycle 23 and future core loadings. The TS changes (Attachments 1 through 3 to Reference 1) reflect impact of the fuel design and the results of the safety analysis used to support the reload applications. As a review of the TS changes by the staff, an RAI was sent by the staff to the licensee requesting further justification on the removal of the safety DNBR value, RCS flow reduction, RCS pressure and temperature changes, and peaking factor changes. The licensee has provided responses to the staff RAI in Reference 5. In addition, the licensee resubmitted the proposed TS changes in Reference 9 that replace the original TS changes in Reference 1 for the staff to review and approve. The following evaluation is based on the staff review of the revised TS changes in Reference 9 and the related information in Reference 5.

1) Figure 2.1-1 and TS B2.1 - Core Safety Limits

The core safety limits (maximum core powers as functions of the RCS pressure and temperature) are revised to reflect the use of the HTP correlation and its associated DNBR safety limit, RCS flow, peaking factors and fuel design. Since the core safety limits are calculated based on the approved methods for licensing applications and the acceptable results of the transient and accident analyses, the staff concludes that the revised safety limits are acceptable.

2) TS.3.10.b - Core Peaking Factor Limits

The core peaking factor limits are added to include 1.70 $F_H(Z)$ and 2.35 $F_Q(Z)$ (100 % power limits) for SPC Heavy fuel. Since the proposed peaking factors were used in the acceptable analysis to support the TS changes, the staff concludes that the changes are acceptable.

3) TS 3.10.k - RCS Average Temperature

The current TS limits the reactor coolant inlet temperature to 535.5 ^{0}F. The proposed TS limits the maximum RCS average temperature to 568.8 ^{0}F. The value of the RCS average temperature with inclusion of measured uncertainties was used in the safety analyses and therefore, is acceptable.

4) TS 3.10.l - RCS Pressure

The existing TS specifies the minimum RCS pressure at 100% power steady-state operation. The proposed TS removes the "100%" power value to provide assurance that the reactor is operated within the assumptions of the safety analysis at all power levels. The change is more restrictive and is acceptable.

5) TS 3.10.m - RCS Flow Rate

The value for the minimum RCS flow per loop specified in TS 3.10.m will be changed from 89,000 gallons per minute to 85,500 gallons per minute. This TS value is greater than that assumed in the acceptable safety analyses. The TS allows the plant operation at greater values of the RCS flow rate, resulting in higher margins to the DNBR limits and PCT safety limit during transients and accidents, and, therefore, is conservative and acceptable.

6) TS 3.10.n - Minimum DNBR Safety Limits

The value of the minimum safety limit DNBR is removed. This is acceptable because TS 2.1, which includes a figure that is based on the minimum DNBR, specifies operating parameter limitations that ensure compliance with this safety limit. In the Bases section, the licensee adds values of the safety DNBR limits for the HTP correlation and the W-3 correlation with a discussion of the conditions of use of the correlations. The changes are consistent with the approach used in the Westinghouse Standard TS and are acceptable.

7) Figure 3.10-1 - Required Shutdown Reactivity Vs. Reactor Boron Concentration

This figure represents a relationship between the required shutdown reactivity reactor boron concentration. The limits for the shutdown are not changed. The required shutdown reactivity line is extended from 1300 ppm to 2000 ppm to account for the 18-month fuel cycle. The existing values are not changed in the safety analysis or in the TS. Therefore, the proposed TS is acceptable.

8) Figure TS 3.10-2 - Hot Channel Factor Normalized Operating Envelope

This figure represents the hot channel factor normalized operating envelope. It is revised to reflected the values used in the acceptable safety analyses and is, therefore, acceptable.

9) Other Changes

The Table of Contents and the Basis sections are changed to be consistent with the TS changes. The staff finds the changes in the Table of Contents and the TS Bases are editorial changes or changes for clarification. Therefore, the changes are acceptable.

2.1.2.7 <u>LICENSE CONDITION</u>

As proposed in the licensee's November 24, 1998 letter, the following licensee condition is to be included in the proposed amendment:

> The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

This condition is consistent with limiting burnup to that value previously assessed by the NRC. Therefore, this change is acceptable.

2.1.3 CONCLUSION REGARDING REVISED PEAKING FACTOR LIMITS

The staff has reviewed the licensee's reload application and the proposed TS changes with the supporting analyses to allow operations of Cycle 23 and future cycles at the KNPP plant. Based on this review, the staff concludes that the supporting safety analyses are acceptable, and the proposed TS changes adequately reflect the results of the acceptable supporting analyses and are, therefore, acceptable for reload applications.

2.2.2 RADIOLOGICAL ASSESSMENT

Operation at higher peaking factors results in an increase in the total activity and associated gap activity of the limiting fuel assembly, but no increase in the corewide inventory of noble gases and halogens. Thus, the only accidents for which the calculated radiological consequences would be affected are the fuel handling accident inside containment and the fuel handling accident outside containment. The licensee provided a reanalysis of these events in conjunction with Amendment No. 132. Amendment No. 132 was issued on May 28, 1998, with a supplement dated September 3, 1998. The Amendment No. 132 evaluation bounds the effects of the new fuel and found the radiological consequences of fuel handling accidents to be acceptable. The staff finds that any postulated release and radiological doses from a large break loss of coolant accident are not affected by the proposed changes.

2.2 TURBINE MISSILES HAZARDS ANALYSIS

2.2.1 LICENSEE'S REQUEST

General Design Criterion 4 requires that structures, systems, and components important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure. Because turbine rotors have large masses and rotate at relatively high speeds during normal reactor operation, failure of a rotor may result in the generation of high energy missiles potentially impacting and damaging safety related structures, systems and components.

Consistent with the staff's position taken on existing turbine rotor designs, the probability of turbine missile generation should be kept to no greater than 10^{-5} per reactor-year (RY) for an unfavorably oriented turbine and 10^{-4} per RY for a favorably oriented turbine.

The Kewaunee Nuclear Power Plant (KNPP) has an unfavorable turbine generator placement and orientation, and the plant is committed to keep the probability of turbine missile generation to no greater than 10^{-5} per reactor-year.

On April 15, 1998, Wisconsin Public Service Corporation (hereafter referred to as the licensee) submitted Proposed Amendment 152 to the KNPP technical specifications (TS). The purpose of the amendment was to document improvements realized by the new fuel design and reflect changes to the KNPP conditions. The amendment also proposes the elimination of high trajectory turbine missiles as a design event impacting the spent fuel. On July 27, 1998, the licensee provided additional information to justify the removal of high trajectory turbine missiles as a design event impacting the spent fuel.

The proposed amendment would change the KNPP Technical Specifications to document improvements realized by the new fuel design and reflects changes to the KNPP operating conditions. It also proposes the elimination of high trajectory turbine missiles as credible design events impacting the spent fuel.

The licensee's basis for proposing an amendment to eliminate high trajectory turbine missile as a design event impacting spent fuel is:

> 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 10^{-5}/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.

2.2.2 EVALUATION AND CONCLUSION REGARDING TURBINE MISSILES

The NRC staff has reviewed the licensee's basis for proposing to eliminate the high trajectory turbine missile as a design event impacting spent fuel and finds it acceptable. This acceptance is based on staff positions stated in Standard Review Plan (SRP) 3.5.1.3, "Turbine Missiles" and review of the turbine manufacturer's methodology for assessing the probability of turbine missile generation. Paragraph III.5 of SRP 3.5.1.3 acknowledges that the probability of a high trajectory turbine missile hitting targets is low (10^{-7} per square foot of target area) and states that risk from high trajectory turbine missiles is insignificant unless the vulnerable target area is in the order of 10^4 square feet or more. The Kewaunee spent fuel pool surface is substantially less about 10^3 square feet.

Based on its evaluation, the staff finds that amending the KNPP TS to eliminate consideration of high trajectory turbine missiles as a design event impacting spent fuel is acceptable. The staff concludes that the risk for the proposed modification of the plant TS is acceptable and meets the relevant requirements of GDC 4. This conclusion is based on the licensee having sufficiently demonstrated to the staff that the probability of turbine missile damage to structures, systems, and components important to safety is acceptably low and within the limits specified in SRP Section 3.5.1.3, "Turbine Missiles."

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on December 2, 1998 (63 FR 66589). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (63 FR 25120).

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Summer Sun, George Georgiev

Date: December 2, 1998

REFERENCES

1. Letter from C. R. Steinhardt (WPSC) to NRC, Proposed Amendment 152 to Kewaunee Nuclear Power Plant Technical Specifications, dated April 15, 1998.

2. Letter from J. G. Giitter (NRC) to D. C. Hintz (WPSC), Safety Evaluation for Reload Safety Evaluation , dated April 11, 1988.

3. Letter from R. J. Laufer (NRC) to M. L. Marchi (WPSC), Safety Evaluation for the High-Thermal-Performance Departure from Nucleate Boiling Correlation (DNB) and the Associated 1.14 Minimum DNBR Ratio Safety Limit for Kewaunee Fuel, dated December 30, 1997.

4. Letter from M. L. Marchi (WPSC) to NRC, Response to Request for Additional Information - Proposed Amendment 152 to the Kewaunee Nuclear Power Plant Technical Specifications, dated August 13, 1998.

5. Letter from M. L. Marchi (WPSC) to NRC, Proposed Amendment 152 to the Kewaunee Nuclear Power Plant Technical Specifications -Supplemental Information, dated September 28, 1998.

6. Letter from C. E. Rossi (NRC), Acceptance for Referencing of Licensing Topical Report XN-NF-82-06(P) Rev.1, "Qualification of Exxon Nuclear Fuel for Extended Burnup," dated July 18, 1986.

7. Letter from A. C. Thadani (NRC), Acceptance for Referencing of Licensing Topical Report ANF-88-133(P), "Advanced Nuclear Fuel's PWR Design Methodology for Rod Burnups of 62 GWd/MTU," dated September 9, 1991.

8. WPSRESM-NP-A, Revision 2, October 1988, Reload Safety Evaluation Methods for Application to Kewaunee.

9. Letter from M. L. Marchi (WPSC) to NRC, Proposed Amendment 152a to the Kewaunee Nuclear Power Plant Technical Specifications, dated September 28, 1998.

10. ANF-88-060(P)(A) and Supplement 1, Generic Mechanical Design Report High Thermal Performance Spacer and Immediate Flow Mixer, dated March 1991.