ust 3, 1994

Docket No. 50-305

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PA 02G5
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DHagan TWFN

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Schrock:

SUBJECT: AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. M88374)

The Commission has issued the enclosed Amendment No.110 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications (TS) in response to your application dated December 1, 1993.

The amendment incorporates technical and administrative changes to TS 3.10, "Control Rod and Power Distribution Limits." Specifically, this amendment eliminates specifications for fuel designs no longer used at Kewaunee, specifies required actions to be taken upon exceeding control bank insertion limits, and revises the limits for Departure from Nucleate Boiling (DNB) related parameters to assure operation within the assumptions of the Updated Safety Analysis Report (USAR) analyses.

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 Richard J. Laufer

Richard J. Laufer, Acting Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation



Enclosures:
1. Amendment No.110 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page



| LA:PDIII | (A)PM:PDIII-3 | (A)BC:SRXB | PD:PDIII-3 | OGC |
|---|---|---|---|---|
| MRushbrook | RLaufer:lm | TCollins | JHannon | RBachmann |
| 7/8/94 | 7/8/94 | 7/8/94 | 8/1/94 | 7/12/94 |
| YES/NO | YES/NO | YES/NO | YES/NO | YES/NO |

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

Docket No. 50-305

August 3, 1994

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service
Corporation
Post Office Box 19002
Green Bay, Wisconsin 54037-9002

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Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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1. Amendment No. 110to
   License No. DPR-43
2. Safety Evaluation

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See next page

cc:

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Chairman
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Mr. Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

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Public Service Commission of
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Hill Farms State Office Building
Madison, Wisconsin 53702

Attorney General
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U. S. Nuclear Regulatory Commission
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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. 110
License No. DPR-43

1. The 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 December 1, 1993, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 110, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: August 3, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 110

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 marginal lines indicating the area of change.

| REMOVE | INSERT |
| --- | --- |
| TS ii | TS ii |
| TS 3.10-1 through<br>TS 3.10-9 | TS 3.10-1 through<br>TS 3.10-9 |
| TS B3.10-2 through<br>TS B3.10-9 | TS B3.10-2 through<br>TS B3.10-9 |
| Figure TS 3.10-2 | Figure TS 3.10-2 |

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

a. Shutdown Reactivity

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

b. Power Distribution Limits

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

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

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

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

where:

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

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

Z is the core height location for the $F_Q$ of interest

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

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

where:

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

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

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

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

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

where:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

c. Quadrant Power Tilt Limits

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

   A. Eliminate the tilt.

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

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

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

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

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

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

d. Rod Insertion Limits

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

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

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

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

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

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

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

e. Rod Misalignment Limitations

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

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

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

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

f. Inoperable Rod Position Indicator Channels

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

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

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

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

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

g. Inoperable Rod Limitations

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

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

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

h. Rod Drop Time

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

i. Rod Position Deviation Monitor

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

j. Quadrant Power Tilt Monitor

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

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

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

m. Reactor Coolant Flow

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

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

n. If, during power operation any of the conditions of TS 3.10.k, TS 3.10.1, 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 level analyzed to maintain a minimum DNBR of 1.30.

Power Distribution Control (TS 3.10.b)

Criteria

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

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

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

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

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

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

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

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[1] USAR Section 4.3

[2] USAR Section 14

[3] USAR Section 4.4

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

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

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

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

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

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

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

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

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

Rod Bow Effects

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

Surveillance

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

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

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

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

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

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

The specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.[6]

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

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

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

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

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

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

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

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

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

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

Quadrant Power Tilt Limits (TS 3.10.c)

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

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

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

If a misaligned rod has caused a tilt ratio > 1.09, the core power shall be reduced by 2% of rated value for every 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.

<u>Rod Insertion Limits (TS 3.10.d)</u>

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

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

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

Rod Misalignment Limitations (TS 3.10.e)

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

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

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

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

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

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

Inoperable Rod Position Indicator Channels (TS 3.10.f)

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

Inoperable Rod Limitations (TS 3.10.g)

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

Rod Drop Time (TS 3.10.h)

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

Core Inlet Temperature (TS 3.10.k)

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

Reactor Coolant System Pressure (TS 3.10.l)

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

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant flow is consistent with the safety analysis.

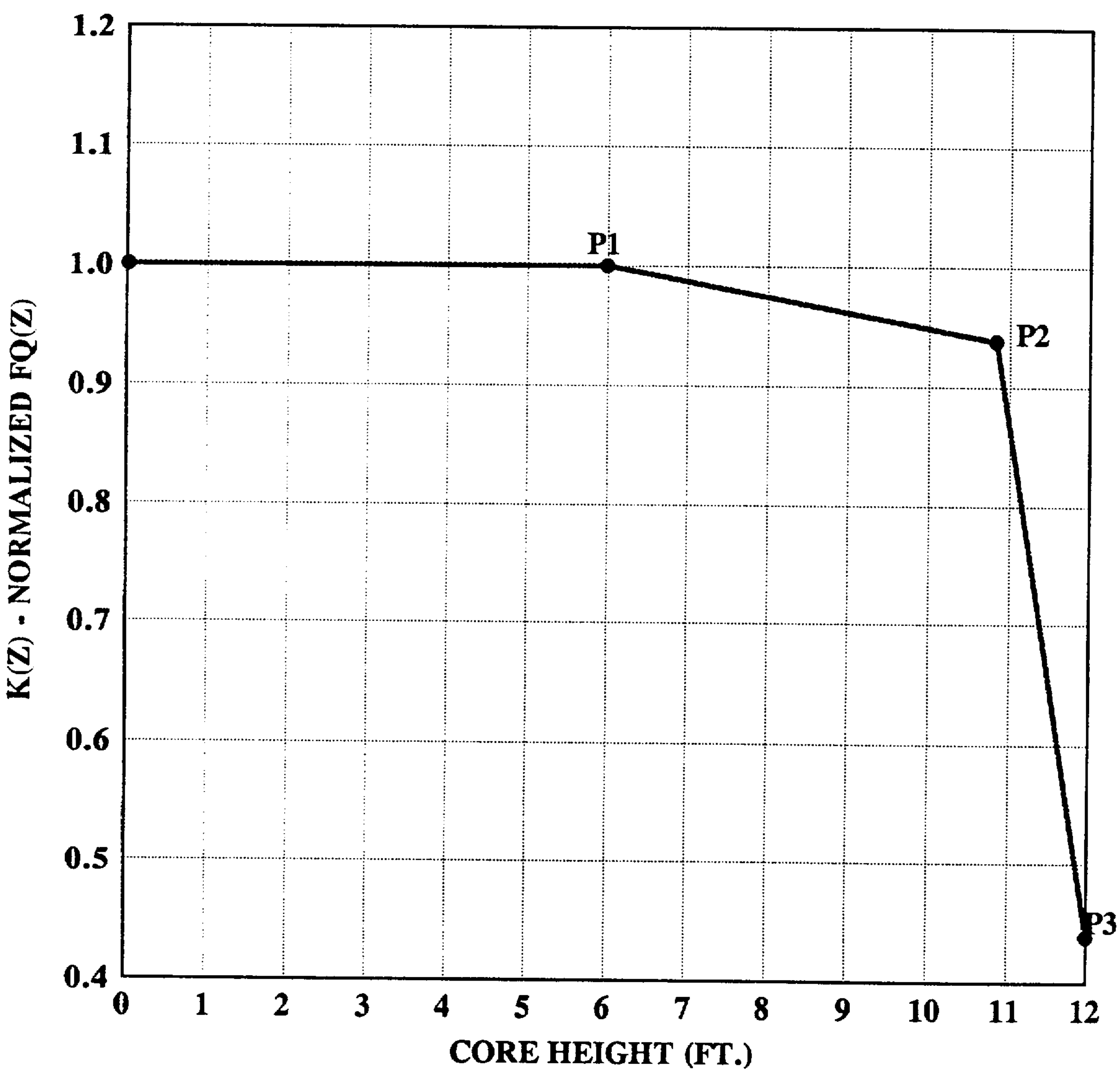
DNB Parameters (TS 3.10.n)

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

## FIGURE TS 3.10-2

## HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

| Siemens Power Corporation Fuel<br>K(Z) Coordinates |
|---|
| P1 (6, 1.0) |
| P2 (10.84, 0.938) |
| P3 (12, 0.438) |
| Normalized to 2.28 |



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO.110 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 letter dated December 1, 1993, the Wisconsin Public Service Corporation (WPSC), the licensee, submitted a request for revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would incorporate technical and administrative changes to TS 3.10, "Control Rod and Power Distribution Limits." Specifically, this proposed amendment would eliminate specifications for fuel designs no longer used at Kewaunee, specify required actions to be taken upon exceeding control bank insertion limits, and revise the limits for Departure from Nucleate Boiling (DNB) related parameters to assure operation within the assumptions of the Updated Safety Analysis Report (USAR) analyses.

## 2.0 EVALUATION

TS 3.10.b.1, 3.10.b.4, and Table TS 3.10-2

TS 3.10.b.1, 3.10.b.4, and Table TS 3.10-2 currently specify limits for the heat flux hot channel factor and the nuclear enthalpy rise hot channel factor; the heat flux hot channel factor under equilibrium conditions; and the hot channel factor normalized operating envelope, respectively. The current TS provide these hot channel factor limits for both Westinghouse Electric Corporation Fuel and Siemens Power Corporation Fuel.

The licensee's proposal eliminates the hot channel factors for Westinghouse Fuel from these TS as this type of fuel is no longer used at KNPP. The limits for Siemens fuel, the current fuel vendor, are retained and are not affected by the proposed change.

Since the proposed change is administrative in nature, and does not alter the intent or interpretation of the specifications, the staff finds it acceptable.

TS 3.10.d

TS 3.10.d currently specifies the limits for control rod insertion, but does not specify any required actions to be taken if the limits are exceeded. The licensee's proposed change adds required actions to be taken if control rod insertion limits are exceeded. The proposed TS change requires operators to initiate boration to restore shutdown margin within one hour of exceeding the control bank insertion limits, and to restore the control banks to within the required limits within two hours. If either of these requirements cannot be achieved, then within one hour the operators must initiate actions to achieve Hot Standby within six hours and Hot Shutdown within an additional six hours.

Adding the required actions to take when control rod insertion limits are exceeded is an enhancement to the TS and the added requirements are consistent with those of the Westinghouse Standard Technical Specifications (STS). The staff, therefore, finds this proposed change acceptable.

TS 3.10.k

TS 3.10.k currently states: "During steady-state 100% power operation, $T_{inlet}$ shall be maintained < 536.5 °F."

The licensee's proposal would change TS 3.10.k to state: "During steady-state 100% power operation, $T_{inlet}$ shall be maintained < 535.5 °F, except as provided by TS 3.10.n."

The reduction in reactor coolant system (RCS) inlet temperature from 536.5 °F to 535.5 °F is being proposed to make the TS limit consistent with the assumptions used in the safety analyses. The current safety analyses in the Updated Safety Analysis Report (USAR) assume an RCS inlet temperature of 539.5 °F. A four degree assumed instrument error reduces the maximum allowable RCS inlet temperature to 535.5 °F. Since this proposed change conservatively increases the margin of safety to DNB related accidents and makes the TS and the USAR consistent, the staff finds it acceptable.

The provisions of TS 3.10.n referenced in TS 3.10.k are evaluated below.

TS 3.10.l

TS 3.10.l currently states: "During steady-state 100% power operation, Reactor Coolant System pressure shall be maintained > 2200 psig."

The licensee's proposal would change TS 3.10.l to state: "During steady-state 100% power operation, Reactor Coolant System pressure shall be maintained > 2205 psig, except as provided by TS 3.10.n."

The increase in minimum reactor coolant system (RCS) pressure from 2200 psig to 2205 psig is being proposed to make the TS limit consistent with the assumptions used in the safety analyses. The current safety analyses in the Updated Safety Analysis Report (USAR) assume an initial RCS pressure of

30 psi below nominal design pressure. The nominal design pressure for KNPP is 2235 psig, therefore, to accurately reflect the accident assumption, the specified limit should be 2205 psig. Since this proposed change is conservative and makes the TS and the USAR consistent, the staff finds it acceptable.

The provisions of TS 3.10.n referenced in TS 3.10.l are evaluated below.

TS 3.10.m

TS 3.10.m currently states that:

> During steady-state power operation, reactor coolant flow rate shall be greater than or equal to 92,560 gallons per minute average per loop; or the $F_{delta\ H}^{N}$ hot channel factor limit for fuel of > 15,000 MWD/MTU shall be reduced 1% for every 1.8% of reactor coolant loop design flow below 92,560 gallons per minute. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow after each REFUELING.

The reactor coolant flow rate of TS 3.10.m was established as a partial offset for the hot channel factor for fuel rod bow effects of Westinghouse Standard Fuel. To address the concerns for reduction in DNBR due to fuel rod bowing, WPSC elected to partially offset fuel rod bowing penalties by taking credit for the actual reactor coolant flow rate exceeding the design flow rate. The design flow rate assumed in the USAR analysis is 89,000 gallons per minute average per loop.

Since the licensee no longer uses the standard fuel design manufactured by Westinghouse, and the current fuel vendor (Siemens Power Corporation) does not require assessment of a penalty for rod bow effects, the existing specification crediting excess reactor coolant flow is no longer required to compensate for reductions in DNBR due to rod bow effects.

The licensee's proposal splits the existing TS 3.10.m into two specifications; TS 3.10.m.1 describes the limits for steady-state reactor coolant flow rate and actions to be taken, if the limits are not met; and TS 3.10.m.2 describes the conditions under which reactor coolant flow is verified.

The proposed TS 3.10.m reads as follows:

m. Reactor Coolant Flow

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

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation

following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

The proposed reduction in the reactor coolant flow limit is consistent with the design flow rate and the assumptions of the safety analyses. Operation with reactor coolant flows greater than or equal to the proposed reactor coolant flow limit assures a minimum DNBR of 1.30 is maintained throughout each analyzed transient. The staff, therefore, finds this change acceptable.

Proposed TS 3.10.m.2 provides clarification, within the specification, of the conditions under which the minimum flow verification is performed. The intent of the proposed change is to clarify the power range during initial power escalation that allows for accurate verification of the RCS flow rate. Since this is an enhancement over the current TS and provides further clarification, the staff finds this change acceptable.

The provisions of TS 3.10.n referenced in TS 3.10.m are evaluated below.

TS 3.10.n

The intent of this new specification is to outline the actions required when the limits of TS 3.10.k (RCS temperature), TS 3.10.l (RCS pressure) and TS 3.10.m.1 (RCS flow) are not met. Collectively, these three specifications place limit on DNB-related parameters to assure each is maintained within the normal steady state envelope assumed in the USAR safety analysis.

The new specification will allow 2 hours to restore the parameter(s) to within limits or power shall be reduced to less than 5% of the thermal-rated power within the next 6 hours. Following analysis, thermal power may be raised not to exceed a level analyzed to maintain a minimum DNBR of 1.30. A completion time of two hours provides sufficient time to determine the cause of exceeding the limit and to correct the condition.

The addition of TS 3.10.n is an enhancement to the TS to provide guidance to the operator on actions required when a DNB-related parameter is outside the limits assumed in the safety analysis. Based on the above discussion and since the proposed specification is consistent with requirements in the STS, the staff finds this change acceptable.

TS 3.10 Basis Changes

To reflect the proposed TS changes discussed above, the licensee has also proposed changes to the Basis section of TS 3.10. The staff has reviewed the proposed changes to the Basis section and determined that they are consistent with the technical changes proposed in the TS amendment request. The staff, therefore, finds the proposed changes acceptable.

Administrative changes to TS 3.10

In addition to the change described above, the licensee is proposing an administrative change to the name of the current fuel vendor. The name "Siemens Power Corporation" has evolved from and is replacing "Exxon Nuclear" and "Siemens Nuclear Power Corporation" in TS 3.10. The changes in company name do not affect the power distribution limits or the overall quality of the fuel provided for use at KNPP.

Since this change is administrative in nature to maintain the accuracy and consistency of the TS, the staff finds it acceptable.

## 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

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. 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 (59 FR 4949). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

## 5.0 CONCLUSION

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.

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Date: August 3, 1994