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APR 29 1982

Docket No. 50-305

Mr. Eugene R. Mathews, Vice President
Power Supply and Engineering
Wisconsin Public Service Corporation
Post Office Box 1200
Green Bay, Wisconsin 54305

Dear Mr. Mathews:

The Commission has issued the enclosed Amendment No. 41 to Facility Operating License No. DPR-43 for Kewaunee Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated August 7, 1981, November 23, 1981, December 8, 1981, and December 23, 1981.

The amendment revises the Technical Specifications in respect to Power Distribution Control, Allowable Control Rod Misalignment and Control Rod Position Indication Systems.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED

Robert B. A. Licciardo, Project Manager
Operating Reactors Branch #1
Division of Licensing

Cp

Enclosures:
1. Amendment No. 41 to DPR-43
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

Concurrence, as read, as to form of FR notice and [illegible] only

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD	
SURNAME	CParrish	RLicciardo:ds	SVarga	TNovak	[illegible]	
DATE	04/9/82	04/12/82	04/[illegible]/82	04/[illegible]/82	04/22/82	

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

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER AND LIGHT COMPANY
MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
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 August 7, 1981, November 23, 1981, December 8, 1981, and December 23, 1981, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 29, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS ii	TS ii
TS 3.10-1	TS 3.10-1
TS 3.10-2	TS 3.10-2
TS 3.10-2a	---------
TS 3.10-3	TS 3.10-3
TS 3.10-4	TS 3.10-4
---------	TS 3.10-4a
TS 3.10-5	TS 3.10-5
TS 3.10-6	TS 3.10-6
---------	TS 3.10-6a
TS 3.10-8	TS 3.10-8
TS 3.10-9	TS 3.10-9
TS 3.10-10	TS 3.10-10
TS 3.10-10a	----------
TS 3.10-11	TS 3.10-11
TS 3.10-12	TS 3.10-12
TS 3.10-13	TS 3.10-13
TS 3.10-14	TS 3.10-14
TS 3.10-15	TS 3.10-15
TS 3.10-16	TS 3.10-16
TS 3.10-17	TS 3.10-17
---------	TS 3.10-18
---------	TS 3.10-19
---------	TS 3.10-20
---------	TS 3.10-21
Figure TS 3.10-2	Figure 3.10-2
Figure TS 3.10-3	Figure 3.10-3
Figure TS 3.10-4	Figure 3.10-4
Figure TS 3.10-5	Figure 3.10-5
Figure TS 3.10-6	Figure 3.10-6
Figure TS 3.10-7	Figure 3.10-7

Amendment No. 41

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, boron, or part length rod position.

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:

(i) Westinghouse Electric Corporation Fuel

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.22/P) \times K(Z)$ for $P > .5$

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.44) \times K(Z)$ for $P \leq .5$

(ii) Exxon Nuclear Company Fuel

$F_Q^N(Z) \times 1.03 \times 1.05 \leq F_Q^T(Ej)/P \times K(Z)$ for $P \geq .5$

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.42) \times K(Z)$ 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 FQ of interest

F_Q^T (Ej) is the function given in Figure TS 3.10-6

Ej is exposure of the fuel rod for the F_Q of interest

B. $F_{\Delta H}^N$ Limits For All Fuel

$F_{\Delta H}^N \times 1.04 \leq 1.55\ (1 + 0.2(1 - P))$ For 0 to 24,000 MWD/MTU burnup fuel

$F_{\Delta H}^N \times 1.04 \leq 1.52\ (1 + 0.2(1 - P))$ For greater than 24,000 MWD/MTU burnup fuel

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 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 specification 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:

A. Westinghouse Electric Corporation Fuel

$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.22/P) \times K(Z)$

B. Exxon Nuclear Company Fuel

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

where:

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

V(Z) is defined in Figure TS 3.10.-7.

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

5. Power distribution maps using the movable detector system shall be made to confirm the relationship of specification 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 more than 10% higher than the power level at which the last power distribution measurement was performed in accordance with 3.10.b.5.A above.

   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 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 3.10.b.4 are not satisfied and the relationships of 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 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 3.10.b.4 exceed the limits specified in the right hand sides.

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 sections 3.10.b.9 through 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 3.10.b.10 through 3.10.b.12 below, the indicated axial flux difference shall be maintained within a $\pm$ 5% band about the target flux difference.

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

11. At power levels greater than 50 percent and less than or equal to 90 percent of rated power:

A. The indicated axial flux difference may deviate from its $\pm$ 5% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by

-10 percent and +10 percent 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 below 90% and above 50%. If the cumulative time exceeds one hour, then the reactor power shall be reduced immediately to less than or equal to 50% power and the high neutron flux setpoint reduced to less than or equal to 55% of rated power.

B. A power increase to a level greater than 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 greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two 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 one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90% of rated power.

13. Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.b.10 or the flux difference time requirement of 3.10.b.11A. 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 exceeds 1.02, one of the following actions shall be taken within two hours:

   A. Eliminate the tilt.

   B. Restrict maximum core power level two percent for every one percent of indicated power tilt ratio exceeding 1.0.

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

3. Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:

   A. Restrict maximum core power level by 2 percent of rated values for every one percent of indicated power tilt ratio exceeding 1.0.

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

4. If the indicated quadrant tilt exceeds 1.09 and there is no simultaneous indication of rod misalignment, the reactor shall immediately be brought to a No Load condition ($\leq$ 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 limit is shown in Figure TS 3.10-3.

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 and the part length rods fully withdrawn.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In specifications 3.10.e.1 and 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.

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

2. When reactor power is less than 85% of rating, the rod cluster control assemblies shall be maintained within $\pm$ 24 steps from their repsective banks. If a rod cluster control assembly is misaligned from its bank by more than $\pm$ 24 steps hwen reactor power is less than 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and specification 3.10.b applied.

3. And, in addition to 3.10.e.1 and 3.10.e.2 above, 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 percent and 100 percent 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) every shift, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.

   B. During operation below 50 percent 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 3.10.f.1.(A) above, then specification 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 specification 3.10.e or 3.10.h.

BASIS

SHUTDOWN REACTIVITY

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 steamline break accident which requires more shutdown reactivity at end of core life (due to a more negative moderator temperature coefficient at end-of-life boron concentrations).

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

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

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

POWER DISTRIBUTION CONTROL

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 analysis.[1, 3] Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.[2]

In addition to conditions imposed for Condition I and II events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F.

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

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local neutron flux in the core at core elevation Z divided by the core averaged neutron flux, assuming nominal fuel and 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 specification 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 Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope indicate that peak clad temperatures remain below the $2200^{\circ}F$ limit.

The $F_Q^N(Z)$ limits of specification 3.10.b.1.A include consideration of enhanced fission gas release at high burnup, off-gassing (release of absorbed gases), and other effects in fuel supplied by Exxon Nuclear Company; this results in an additional penalty in the form of the function $F_Q^T(Ej)$, as shown in Figure TS 3.10-6, which is applied to Exxon fuel. References 7 and 8 discuss these phenomena.

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

In specification 3.10.b.1 and 3.10.b.4 F_Q^N 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 integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in 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 $F^N_{\Delta H}$.

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

The use of $F^N_{\Delta H}$ in specification 3.10.b.5 is to monitor "upburn" which is defined as an increase in $F^N_{\Delta H}$ 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

The $F^N_{\Delta H}$ limits of specification 3.10.b.1 include consideration of fuel rod bow effects. Since the effects of rod bow are dependent on fuel burnup an additional penalty is incorporated in a decrease in the $F^N_{\Delta H}$ limit of 2% for 0-15000 MWD/MTU fuel burnup, 4% for 15000-24000 MWD/MTU fuel burnup, and 6% for greater than 24000 MWD/MTU fuel burnup. These penalties are counter-balanced by credits for increased Reactor Coolant flow and lower core inlet temperature. The Reactor Coolant System flow has been determined to exceed design by greater than 8%. Since the flow channel protective trips are set on a percentage of full flow, significant margin to DNB is provided. One half of the additional flow is taken as a DNB credit to offset 2% of the $F^N_{\Delta H}$ penalty. The existence of 4% additional reactor coolant flow will be verified after each refueling at power prior to exceeding 95% power. If the reactor coolant flow measured per loop averages less than 92560 gpm, the $F^N_{\Delta H}$ limit shall be reduced at the rate of 1% for every 1.8% of reactor coolant design flow (89000 gpm design flow rate) for fuel with greater than 15000 MWD/MTU burnup. Uncertainties in reactor coolant flow have already been accounted for in the flow channel protective trips for design flow. The assumed T inlet for DNB analysis was 540°F while the normal T inlet at 100% power is approximately 532°F. The reduction of maximum allowed T inlet at 100% power to 536°F as addressed in specification 3.10.k provides an additional 2% credit to offset the rod bow penalty. The combination of the penalties and offsets results in a required 2% reduction of allowed $F^N_{\Delta H}$ for high burnup fuel, 24000 MWD/MTU. The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits.

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 $\geq$ 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.
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.[9]

Conformance with specification 3.10.b.9 through 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 section 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 $\pm 5\%$ flux difference are permitted from the indicated reference value. Figure TS 3.10-5 shows a typical construction of the 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; however, to simplify the specification, a limitation of one 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 below 90% and above 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 one 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, without part length rods, 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

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 two percent 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 Technical Specifications, Section 4.1.

The two hour time interval in specification 3.10.c is considered ample to identify a dropped or misaligned rod. In the event that the tilt condition cannot be eliminated within the two 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 utilizing the movable detector system. For a tilt condition $\leq$ 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 two percent for each one percent 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. In the event a tilt condition of $\leq$ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for flux mapping and turbine synchronization.

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a low power condition for investigation by flux mapping. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2% for each 1% the tilt ratio exceeds 1.0) for the 8 hour period necessary to correct the rod misalignment.

ROD MISALIGNMENT LIMITATIONS

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 Specification 3.10.e. A relaxation in those limits for power levels below 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 non-linearity 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 pressu : 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 re-activated) 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

The rod position indicator channel is sufficiently accurate to detect a rod $\pm$ 7.5 inches away from its demand position. If the 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

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 re-analysis of all accidents sensitive to the changed initial condition.

ROD DROP TIME

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

DNB PARAMETERS

The DNB related accident analysis assumed as initial conditions that the T inlet was 4^{o}F above nominal design or T avg was 4^{o}F above nominal design. The Reactor Coolant System pressure was assumed to be 30 psi below nominal design.

REFERENCES

(1) FSAR Section 4.3

(2) FSAR Section 4.4

(3) FSAR Section 14

(4) "Rod Misalignment Analysis," July 27, 1981, submitted to NRC with proposed Technical Specification Amendment 46 by letter from E. R. Mathews (WPSC) to D. G. Eisenhut (NRC) dated August 7, 1981.

(5) Letter from E. R. Mathews, (WPSC), to D. G. Eisenhut, (NRC), dated January 8, 1980, submitting information on Clad Swelling and Fuel Blockage Models.

(6) Letter from E. R. Mathews, (WPSC), to A. Schwencer, (NRC), dated December 14, 1979, submitting the ECCS Re-analysis properly accounting for the zirconium/water reaction.

(7) George C. Cooke, Philip J. Valentine: "Exposure Sensitivity Study for ENC XN-1 Reload Fuel at Kewaunee Using the ENC-WREM-IIA PWR Evaluation Model, WN-NF-79-72," Exxon Nuclear Company, October, 1979.

(8) Letter from L. C. O'Malley, (Exxon Nuclear Company) to E. D. Novak, (WPSC), providing FQ exposure dependence as a function of rod burnup.

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

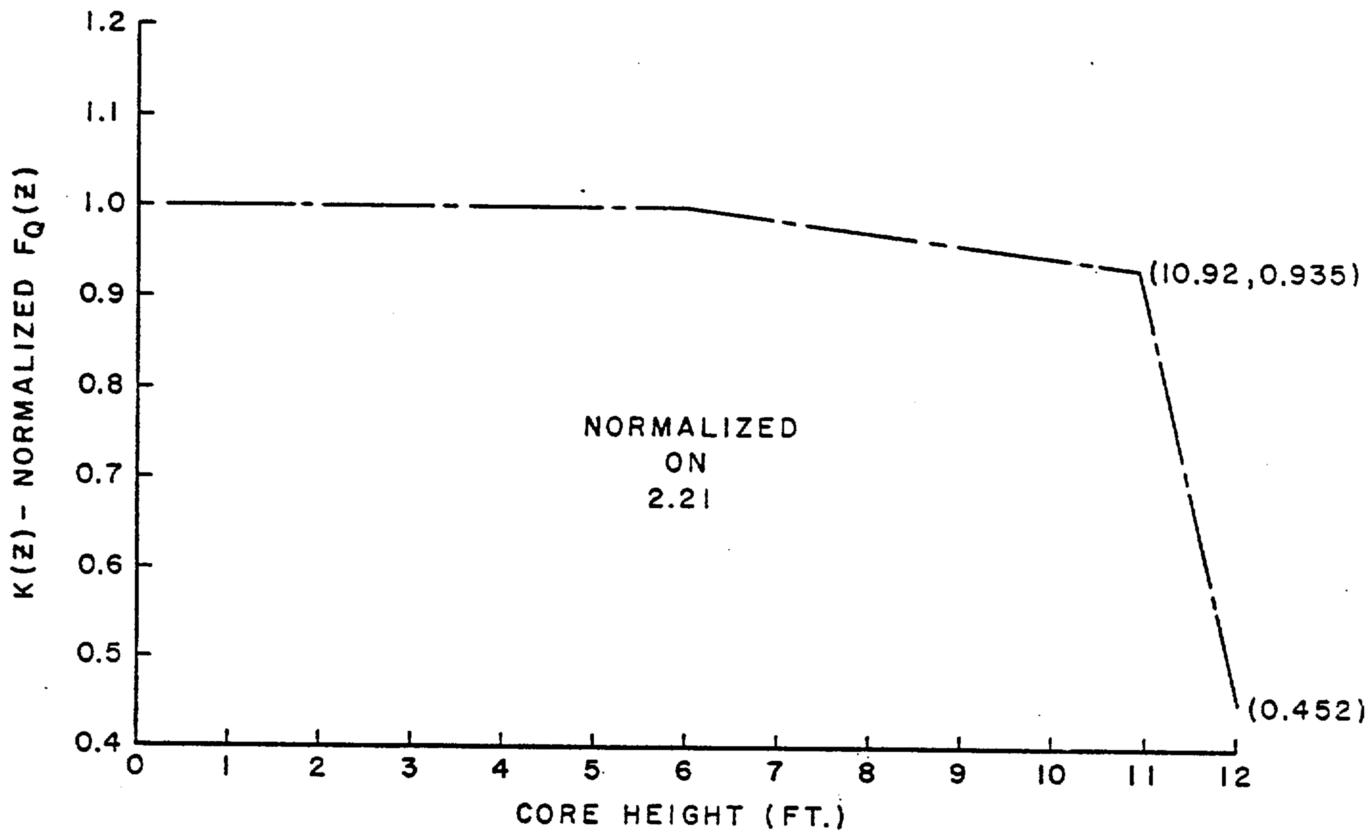
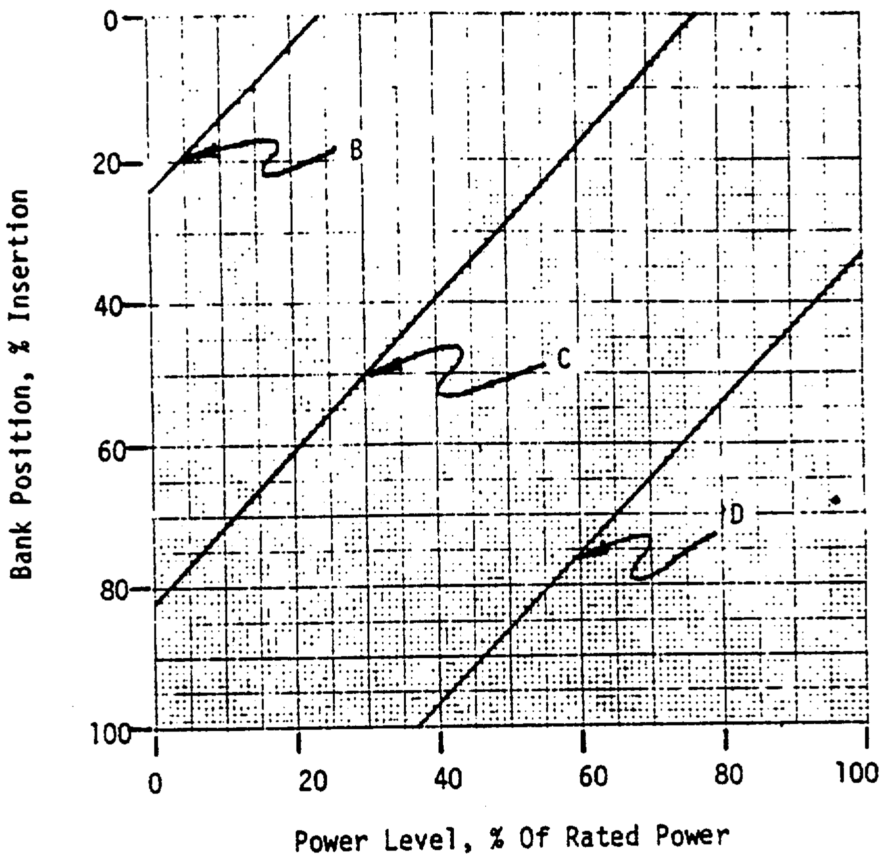
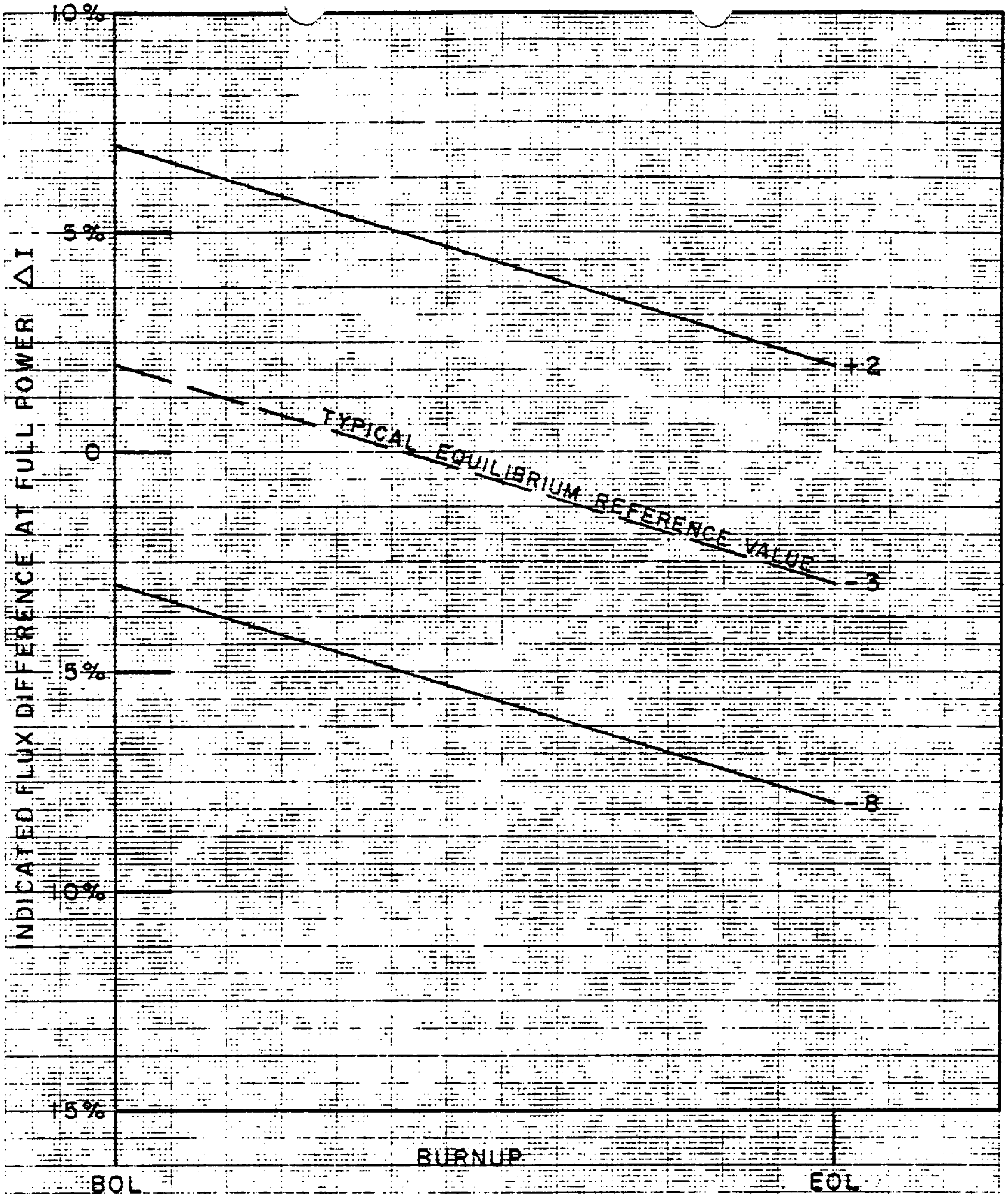


FIGURE TS 3.10-2 HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



CONTROL BANK INSERTION LIMITS

FIGURE TS3.10-3



PERMISSIBLE OPERATING BAND ON
INDICATED FLUX DIFFERENCE AS A
FUNCTION OF BURNUP (TYPICAL)

Figure TS 3.10-4

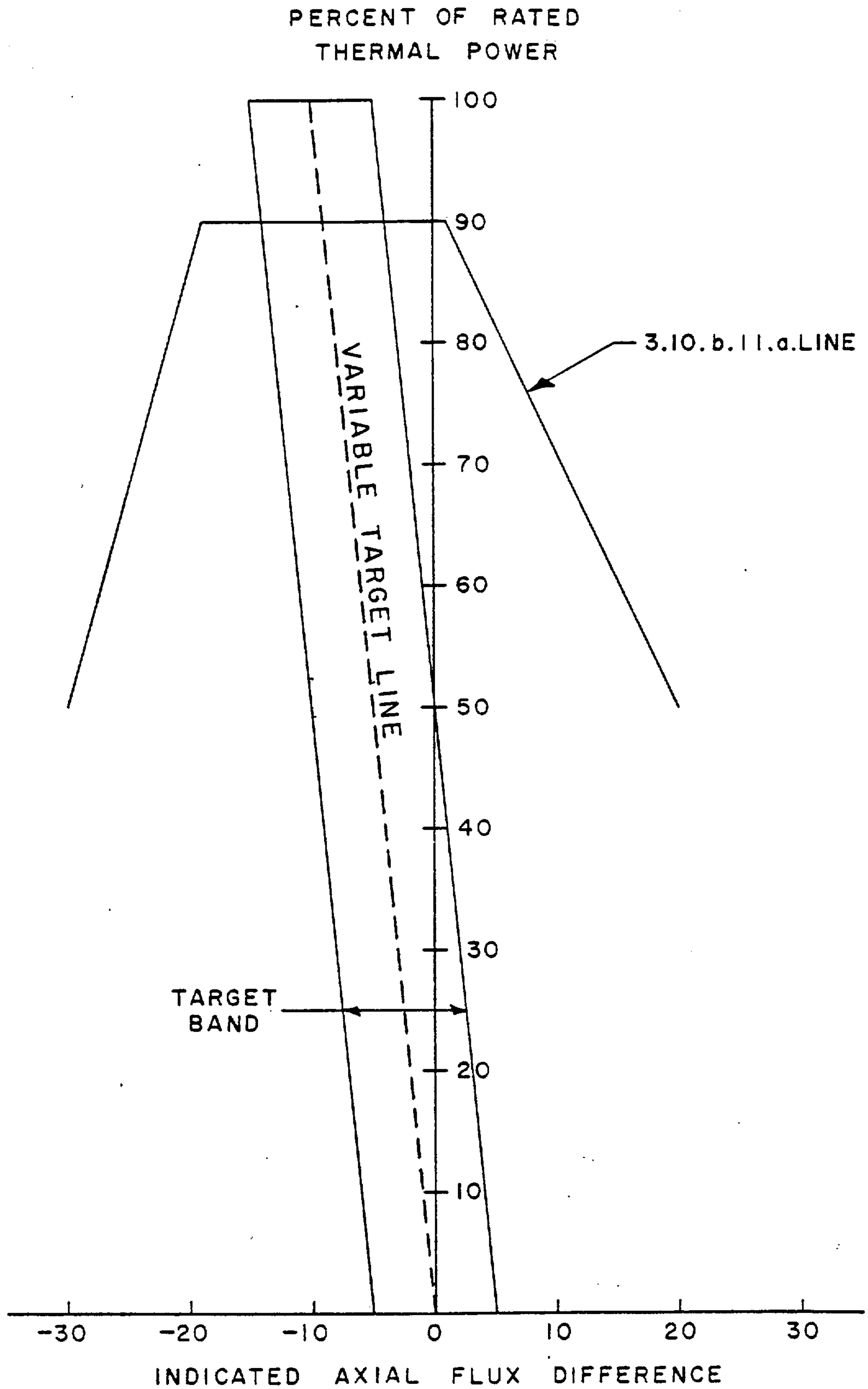
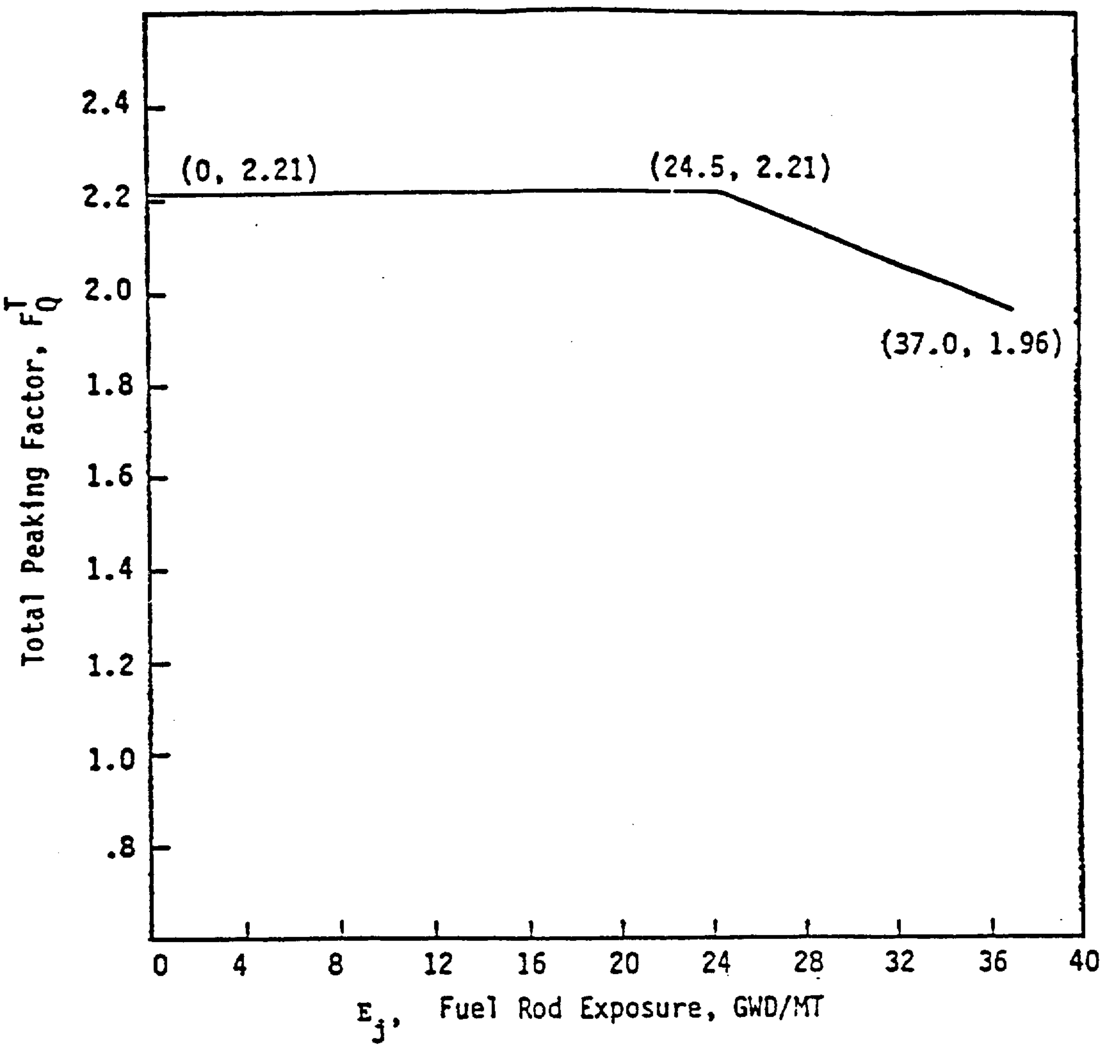


FIGURE TS 3.10-5 TARGET BAND ON INDICATED FLUX DIFFERENCE AS A FUNCTION OF OPERATING POWER LEVEL (TYPICAL)

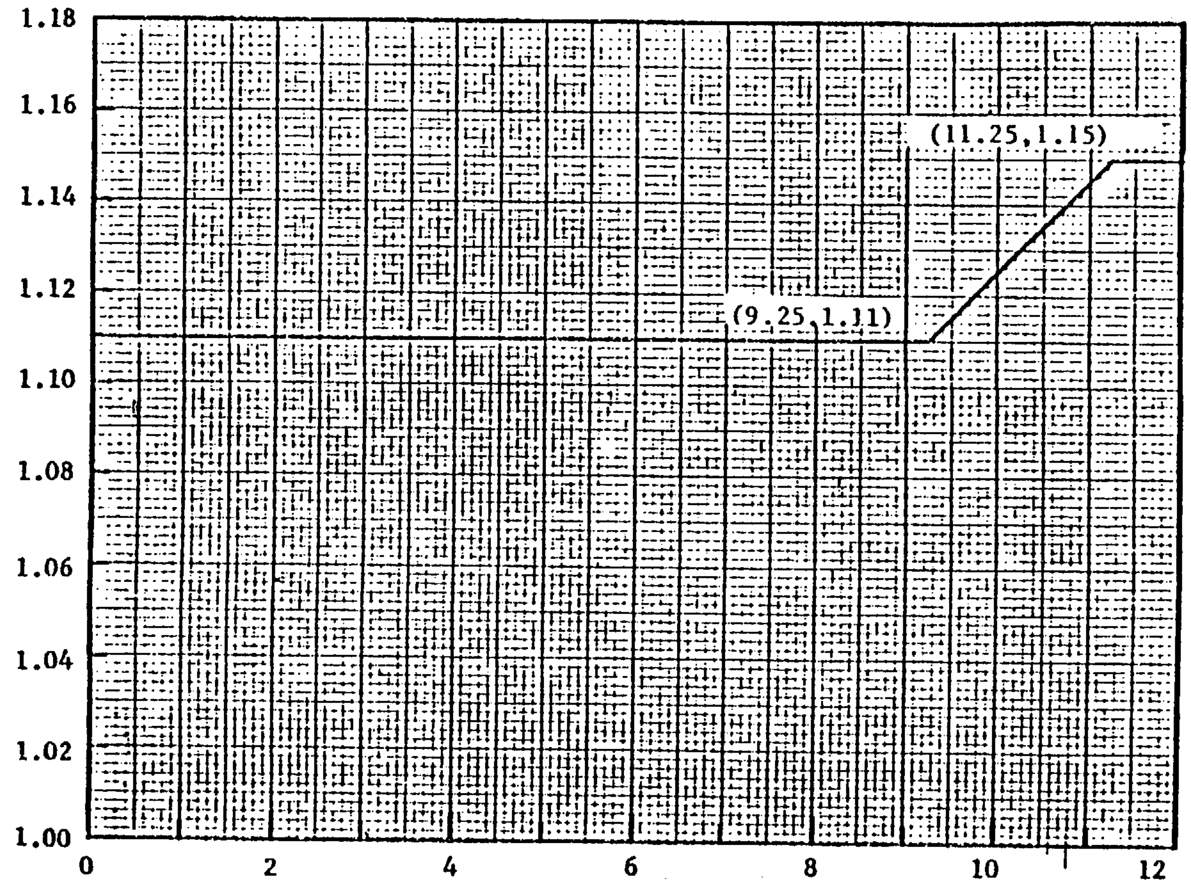
AMENDMENT No 41



F_Q^T versus Rod Exposure: $F_Q^T(E_j)$

(Reference specification 3.10.b.1.a.(ii))

Figure TS 3.10-6



V(Z) as a Function of Core Height

Figure 3.10-7
Amendment No. 41

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 41 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

I. Introduction

By letters dated November 23, December 8, and December 23, 1981, Wisconsin Public Service Corporation (the licensee) submitted their proposed Amendment No. 48 to the Technical Specifications for the Kewaunee Nuclear Power Plant. The proposal requested changes for:

a) Allowable Control Rod Misalignment

b) Power Distribution Control

c) Control Rod Position Measurement

Each of these requested changes has been evaluated to establish its particular features, and related safety and environmental impacts, and the necessary safety conclusions have been drawn.

Proposed Amendment No. 48 supersedes proposed Amendment No. 46 in respect of the item "Rod Misalignment" therein; Amendment No. 46 was submitted by the licensee by letter dated August 7, 1981.

II. Power Distribution Control and Allowable Control Rod Misalignment

A. Introduction

In letters dated November 23, December 8 and December 23, 1981, Wisconsin Public Service (WPS) has proposed (their Amendment 48) revisions to Kewaunee Nuclear Power Plant Technical Specifications. These revisions deal with control rod misalignment and power distribution control Technical Specifications. Both of these subjects have been the focus of much work by the NRC staffs since May 1981.

In 1979 the NRC staff reviewed the LER's and Technical Specification requirements related to the Control Rod Position System for Westinghouse PWRs. Westinghouse had performed safety analyses for control rod misalignment up to 15 inches or 24 steps. The actual misalignment may be 15 inches when an indicated deviation of 7.5 inches exists because the analog control rod position indication system has an uncertainty of 7.5 inches. At that time WPS (Ref. 1) was requested to review their Technical Specifications to ensure that the control rods were required to be maintained within 7.5 inches indicated and that the rod position indication system was verified to be accurate to within 7.5 inches.

WPS responded (Ref. 2) that based on their analysis of a misaligned rod, their operating history and normal mode of operation, their Technical Specifications assured that core power distribution limits would not be exceeded. Their Technical Specifications stated that a rod cluster control assembly could not be misaligned by more than 15 inches without action. They interpreted this as 15 inches indicated.

WPS continued to perform analysis and again (Ref. 3) informed NRC that they believe that the existing Technical Specifications which allowed up to 15 inches indicated misalignment were adequate. The NRC staff did not agree with this.

Since that time there have been many discussions, various interim positions, and a plant visit by NRC staff. WPS's concern in agreeing to a specification limiting them to ± 7.5 inches indicated was a result of a drift problem with the analog control rod position indicating system which made it impossible for them to maintain the ± 7.5 inches indicated for some rods.

B. Evaluation

Revised calibration procedures described in the following SER have been worked out which allow adjustment to compensate for the effects of power ascension. The Technical Specifications as stated in WPS' proposed Amendment 48 allows a ± 7.5 inch indicated misalignment. This is consistent with the Westinghouse analysis and the Standard Technical Specifications. For powers lower than 85 percent, larger misalignments - up to ± 15 inches indicated - are allowed because of the increased margin in peaking factors and greater shutdown margin obtained while operating at lower power levels. The increased flexibilitly is desired to account for the non-linearity inherent in the rod position indication system and for the effects of temperature and power on the rod position system. The staff concludes that the Technical Specifications relating to allowable control rod misalignment as proposed in their Amendment 48 are acceptable.

The Technical Specification changes dealing with the Reactor Physics Methodology are consistent with the Technical Specifications proposed by Exxon for Westinghouse designed reactors in "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors - Phase 2," XN-NF-77-57(A), May 1981. This is an approved document. The Kewanuee Technical Specifications also include a penalty factor for fuel with burnup greater than 24,000 MWD/MTU as proposed by Exxon. The staff finds these Technical Specification changes acceptable.

C. SUMMARY

The proposed changes in Kewaunee Technical Specifications on control rod misalignment are in conformance with Westinghouse Standard Technical Specifications and are, therefore, acceptable. The proposed changes to the Technical Specifications on the power distribution control strategy are similar to those found acceptable in previous applications and are, therefore, acceptable. The proposed changes will not significantly reduce the safety margin for the Kewaunee Nuclear Power Plant nor adversely affect the health and safety of the public.

References

1. Letter from A. Schwencer (NRC) to E. R. Mathews (WPSC) dated October 29, 1979 concerning control rod position indication systems at Westinghouse PWRs.

2. Letter from E. R. Mathews (WPSC) to A. Schwencer (NRC) dated December 5, 1979 on subject of Rod Misalignment Technical Specifications.

3. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated May 5, 1981 on subject of Rod Misalignment Technical Specifications.

4. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated July 8, 1981 on subject of Rod Misalignment Concerns.

5. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated July 31, 1981 on subject of Rod Misalignment Concerns.

6. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated August 3, 1981 on subject of Rod Misalignment Concerns.

7. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated August 7, 1981 on subject of Proposed Amendment 46 to Kewuanee Technical Specifications.

8. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated August 21, 1981 on subject of FQ Analysis for KNPP Cycle 7.

9. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated September 2, 1981 on subject of FQ Calculation Methdology.

10. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated September 21, 1981 on subhect of Rod Misalignment Concerns.

11. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated October 28, 1981 on subject of Control Rod Misalignment.

12. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated October 28, 1981 on subject of FQ surveillance.

13. Letter from E. R. Mathews (WPSC) to S. A. Varga (NRC) dated November 16, 1981 on subject of Preliminary Copy of Proposed Amendment 48.

14. Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (NRC) dated November 23, 1981 on subject of Proposed Amendment 48 to the KNPP Technical Specifications.

15. Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (NRC) dated December 8, 1981 on subject of Revisions to Proposed Amendment 48 to the KNPP Technical Specifications.

16. Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (NRC) dated December 23, 1981 on the subject of Revision to Proposed Amendment 48 to the KNPP Technical Specifications.

III. Control Rod Position Indication System Concerns

A. Introduction

Operating experience at Westinghouse (W) PWRs has shown that the analog control rod position instrumentation based upon the Linear Variable Transformer detection method may have inherent characteristics which make it difficult to meet the accuracy requirements assumed by the plant's safety analyses.

In the autumn of 1979, the NRC noted that if one of the detector coils should fail (a single failure) the rod position indication system would be in error by 12 steps (or 7.5 inches, since the coils are spaced 3.75 inches apart, one step being equal to 5/8 inch). In October 1979, the NRC sent generic letters to each W licensee (Reference 1) indicating that, with an uncertainty of 12 steps in the instrumentation, the accident analysis assumption of a control rod misalignment of 24 steps (i.e., 15 inches) could not be assured unless the licensee took action at the point where an indicated deviation of 12 steps occurred. These letters requested each licensee to propose revised Technical Specifications accordingly.

In the fall of 1980, operating experience at another W PWR showed that, even without a failure in the system, the inherent characteristics of the system made the goal of a ± 12 steps accuracy a difficult challenge for this generation of instrumentation.

The Technical Specifications for the Kewaunee plant required that the licensee maintain rod misalignment no greater than 15 inches (i.e., 24 steps). The licensee responded to the NRC original concern with a letter dated December 5, 1979 (Reference 2), in which the licensee stated his intent to show by core physics analysis that an indicated misalignment of 24 steps plus the uncertainty (i.e., 36 steps total) would not violate the core power distribution limits. In Reference 3, the licensee reported the completion of such analyses, with the conclusion that either power peaking factors were maintained within the specified limits or an axial offset or core flux tilt limit was reached. On this basis, the licensee stated that no changes to the Technical Specifications were being proposed.

As shown in the list of References, many formal exchanges of information have occurred between the NRC and the licensee on the matter of Control Rod Position Indication. In addition, the NRC participated in numerous telephone conferences with the licensee, reviewed many draft copies of WPS correspondence, and made a visit to the plant to gain first-hand knowledge of the instrumentation performance capabilities. Separately, the matter of the rod position instrumentation has been discussed several times with Westinghouse; reference 8 is one of the results of these discussions.

B. Background

The rod position detector is a linear variable transformer consisting of primary and secondary coils alternately stacked on a stainless steel cylindrical tube. An extension shaft from the rod drive mechanism extends up into the tube and serves as the variable "core" for the transformer. With a constant a.c. current source (200 mA) applied to the primary windings, the position of the rod drive extension shaft changes the primary to secondary coupling and produces a secondary voltage that is directly related to rod position. The secondary voltage (8.0-12.5 VAC) is sent to an electronic module which converts the a.c. signal into an appropriate d.c. voltage which is sent to the plant process computer. This module contains "Zero" and "Span" adjustments plus an "output voltage" test point (0-3.45 Vdc). A secondary amplifier on the module takes the d.c. output voltage and drives the board-mounted indicator. A built-in set of test points facilitates measurements of the primary voltage of the detector transformers. A test signal generator is provided to adjust the "rod bottom" bistables.

The characteristics of interest are of two general types. First, the channels have non-linearity in the steady-state response. Second, the channels display a time-dependent (transient) response due to thermal effects in the detector assembly.

A typical steady-state calibration curve is an arc-shaped curve, with the indicated position low at the near full-in and near full-out extremities and the indicated position high in the mid-travel region. The steady-state response also depends to some degree upon whether the last rod motion was a withdrawal or insertion. For most rods, but not all, the Zero and Span adjustments allow the steady-state calibration curve to be fitted within the $\pm$12 steps acceptance band. The Zero and Span adjustments are interdependent. Large changes in either of these adjustments can invalidate any previous output voltage-to-position calibration, necessitating a re-calibration by rod full-stroke movement. Once calibrated, however, voltage measurements can be used to determine rod position.

The transient response for the RPI's is typically of the "over shoot" type. That is, if the rod is being pulled out, the RPI indication will show a greater withdrawal and later settle (at thermal equilibrium) back to the steady-state value; if the rod is being inserted, the initial indication is greater insertion than actual. The magnitude of this thermal transient response appears empirically to be insignificant in the region of the lowest one-third of rod travel. However, near the fully withdrawn positions, this transient response at some plants can be as great as 25 steps. The time constant of the thermal recovery toward the steady-state value varies with rod location radially across the core and has values between 10 and 15 minutes. "Settling Times" of 20 to 45 minutes have been observed before steady-state is reached.

While the Rod Position Indicators (RPI's) may not be formally classified as "safety-related", these indications are important to safety. First, FSAR Chapter 15 accident analyses generally presume an instrument accuracy no worse than ±12 steps when evaluating potential rod misalignments. Secondly, the indication that the rods are at the bottom (i.e., "seated") following scram is an important function provided by these channels.

The poor performance of one RPI is a situation of limited concern. However, our on-site review of the situation confirmed that several of the indicators behave generally the same. Our concern in the more generalized case includes not only a possible non-conservative FSAR assumption, but also the potential for operator disregard or distrust of these indications because of a history of accuracy problems.

C. Evaluation

The Technical Specifications (T.S.) for this plant were written before the advent of standard technical specifications. The Kewaunee T.S. (Section 3.10.e) contain a requirement that, if a control rod becomes misaligned from its bank by more than 15 inches, remedial action would be taken. If certain actions were not completed within two hours, reactor power had to be reduced to 85% or less. This T.S. is a functional requirement in that the functional objective is stated but the specific requirements are only implied. For example, the control room indications that would define a 15-inch misalignment are not specified.

The Kewaunee T.S. requires (Section 3.10.f) that the position of a control rod be checked indirectly when an individual rod position indicating channel is out-of-service or "inoperable." However, the T.S. do not specify what constitutes an "inoperable" channel. There have been plant operating conditions where the indicated rod position deviated from the actual rod position by 12 or more steps and the channel was not declared to be inoperable.

By comparison, standard Technical Specifications for Westinghouse plants require: (1) that the rod position indicating instrument for each control rod have an inaccuracy of no greater than ± 12 steps (i.e., 7.5 inches); and (2) that, if an individual rod position indication deviates from the bank demand counter by 12 steps or more, the control rod is declared to be misaligned and action is required.

Similarly, the present Kewaunee T.S. (Section 3.10.i) require certain manual surveillance actions when the automatic Rod Deviation monitor is "inoperable." There is no T.S. requirement specifying what the setpoint for this alarm should be. The setpoint had been 20 steps. The operators would be "alerted" by this alarm but would not necessarily have taken any action until the indicated deviation reached 24 steps (i.e., 15 inches). This practice did not allow for any uncertainty or inaccuracy in the indication.

When the NRC requested the licensee to propose a T.S. limit of a 12-step indicated deviation as a definition of a misaligned rod, the licensee apparently perceived a situation that could restrict plant operations significantly without any safety improvement. The licensee is concerned with reactor safety and would take appropriate corrective action when he believed that a rod is misaligned. However, to define a misaligned rod against instrumentation of such accuracy is a different matter. During our visit to the plant, about four rod position indicators showed values that were 12 steps or more away from the demand counters. Other indications had confirmed that the rods were in fact at the positions shown by the demand counters.

The licensee had proposed to depend upon indirect measurements of rod positions. Clearly, indirect measurements from ex-core neutron detectors, thermocouples, and movable in-core detectors can provide data about core conditions from which some information regarding rod positions can be inferred. However, long-standing policy and practice of the NRC has been to require rod position information to be displayed directly. Therefore, reliance upon indirect measurements and alarms such as axial offset or flux tilt to determine misaligned rod positions is not sufficient.

After discussions with the licensee, a solution that meets the NRC requirements and avoids unduly restricting plant operations has been developed. This solution centers on several points which are discussed below.

1. A primary purpose of rod-misalignment specifications is to avoid flux peaking factors less conservative than assumed for the accident analysis. If such conditions can be recognized and corrective action initiated within a couple hours, the accident analyses are protected to an acceptable degree. That is, a potentially misaligned rod that is undetected for up to an hour is not unacceptable.

2. Previously, the NRC had considered the primary indicator of rod position to be the individual rod position channels; the demand counters had been considered to be of secondary importance. There were several reasons for this approach. One is that there is only one demand counter for each group of several rods and individual rod position is valuable. Another is that the demand counters indicate basically the input to the rod drive control system (i.e., where the rods "are told to go") and the individual position indicators independently show the output of the rod drive control system (i.e., where the rods "actually went").

   However, the operating experience with these individual rod position indicators has been plagued by less-than-desirable performance. The steady state errors and transient indications of this generation of instrumentation are significant. On the other hand, the reliability of the control rod drive system has been quite good. The demand counters almost always show the correct position of the rods and in an accurate and convenient to read manner.

Therefore, based upon this operating experience, the demand counters are now considered the immediate and primary rod position indicators at this plant. When confirmation is needed periodically, the individual rod position channels can be used. Such use must be delayed about 30 minutes to allow the transient behavior to dissipate and must be used with care as discussed further below.

3. The calibration of the individual rod position channels is performed at hot zero-power conditions, since full-stroke rod motion is involved. However, when full power is reached the conditions ambient to the channel detectors (i.e., the coil-stacks above the reactor vessel head) change. These affects vary somewhat radially across the core and at Kewaunee produce shifts in calibration of the rod position channels of three to six steps. The exact values of these shifts must be determined empirically. We are allowing the licensee to make minor corrections to accomodate these shifts in instrument calibration due to power ascension. Technical Specifications have been amended to require that, following any such adjustment, the channel be checked at an intermediate and at a low level to confirm that the overall calibration is not adversely affected by the adjustment.

4. The design of the channel output amplifier that drives the individual rod position meter, the meter itself, and the Rod Deviation alarm (which is generated by the plant process computer) are based implicitly upon the presumption of a linear relationship between the "output voltage" of the channel and actual rod position. Investigation has shown that this relationship is not linear and is actually arc-shaped (as discussed in the "Background" section of this report). The deviation of the arc from a straight line can be 12 steps or more. We are allowing the licensee to make "software corrections" to account for known non-linearity of the channel. These corrections take two forms. First, the plant process computer has been programmed to employ a curve-fitting process in converting the rod position channel output voltage signal into a rod position in steps. Second, when it becomes desirable to determine the position of a rod by manually reading the output voltage of a channel, the voltage vs steps calibration curve may be used. In these ways, the non-linearity of the channels is taken out of the position determination.

With these general improvements, the licensee has developed the following procedures to detect a misaligned control rod:

(a) The Rod Deviation alarm will have a setpoint of 12 steps when reactor power is 85% or greater. When the process computer is available, it provides continuous monitoring of the deviation between the demand counters and the individual rod position indicators. If this alarm sounds and does not clear itself within one hour, the rod will be declared to be misaligned and remedial action will be initiated.

(b) When the process computer is not available, the positions of the rods as indicated by the individual meters will be compared to demand counters by the reactor operator at least every eight-hour shift and following rod motion of greater than 6 inches. If these indications agree within the 12-step limit, no further action is necessary. If these indications deviate by 12 steps or more, the output voltage of the individual channel will be measured manually within two hours. If the rod position, as shown by the voltage measurement and rod calibration curve, deviates from the demand counter by 12 steps or more, the rod will be declared to be misaligned and remedial action will be initiated.

The licensee has now proposed changes in the Technical Specifications that change the 15-inch misalignment requirement to an indicated ± 12 steps limit to accomodate uncertainty in the rod position indicating instrumentation. The proposed Bases for the T.S. describe the rod position instrumentation and provide the accuracy limit of 7.5 inches (12 steps) for operability.

Summary

The licensee has developed a better understanding of the capabilities and limitations of the individual control rod position instrumentation. Based upon this understanding, several improvements have been made in corrections for the calibration procedures and in operating techniques. The licensee has therefore reached a position of being able to propose the Technical Specifications changes that were requested without unduly restricting plant operations. Based upon our understanding of the information provided, we conclude that the proposed changes regarding rod position instrumentation are acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 29, 1982

Principal Contributors:

Power Distribution Control and Allowable Control Rod Misalignment
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Control Rod Position Indication System Concerns
J. T. Beard

REFERENCES

1. Letter; A. Schwencer (NRC) to E.R. Matthews (Wisconsin Public Service Co. - WPS), dated October 29, 1979.

2. Letter; E.R. Matthews (WPS) to A. Schwencer (NRC), dated December 5, 1979.

3. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated May 5, 1981.

4. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated July 8, 1981.

5. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated July 31, 1981.

6. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated August 3, 1981.

7. Letter; E.R. Matthews (WPS) to D. Eisenhut (NRC), dated August 7, 1981.

8. Letter; E.P. Racke, Jr. (Westinghouse) to L.E. Phillips (NRC), dated August 12, 1981.

9. Information provided by licensee during plant visit of September 14, 15, 16, 1981.

10. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated September 21, 1981.

11. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated October 28, 1981.

12. Letter; S. Varga (NRC) to E.R. Matthews (WPS), dated November 10, 1981.

13. Letter; E.R. Matthews (WPS) to S. Varga (NRC), dated November 16, 1981.

14. Letter; E.R. Matthews (WPS) to D. Eisenhut (NRC), dated November 23, 1981.

15. Letter; E.R. Matthews (WPS) to D. Eisenhut (NRC), dated December 8, 1981.

16. Letter; E.R. Matthews (WPS) to D. Eisenhut (NRC), dated December 23, 1981.

17. Memo; L. Rubenstein (NRC) to T. Novak (NRC), dated February 4, 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-305

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 41 to Facility Operating License No. DPR-43, issued to Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees), which revised Technical Specifications for operation of the Kewaunee Nuclear Plant (the facility) located in Kewaunee Co., Wisconsin. The amendment is effective 30 days from the date of issuance.

The amendment revises the Technical Specifications in respect of Power Distribution Control, Allowable Control Rod Misalignment, and Control Rod Position Indication Systems.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since this amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated August 7, 1981, November 23, 1981, December 8, 1981, and December 23, 1981, (2) Amendment No. 41 to License No. DPR-43 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Kewaunee Public Library, 314 Milwaukee Street, Kewaunee, Wisconsin 54216. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 29th day of April, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing