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Operated by Nuclear Management Company, LLC

NRC-02-064

10CFR50.90

July 26, 2002

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
License Amendment Request 185 To The Kewaunee Nuclear Power Plant Technical Specifications, "Core Operating Limits Report Implementation."

In accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, Nuclear Management Company, LLC., hereby requests amendments to Facility Operating License DPR-43 for Kewaunee Nuclear Plant. The purpose of this license amendment request (LAR) is to implement a Core Operating Limits Report (COLR) at the Kewaunee Nuclear Plant (KNPP). NMC request approval of this LAR in February of 2003 with 60 days to implement.

NRC Generic Letter 88-16 provides guidance for licensees allowing relocation of cycle dependent variables from the Technical Specifications (TS) to a formal report, COLR. This approval requires that the values of these variables included in a COLR, are determined with NRC-approved methodologies referenced in the TS, and that changes to these variable values are reported to the NRC as they are made. The variables removed from the TS and relocated to the COLR can then be changed via the appropriate regulatory change mechanisms, principally 10 CFR 50.59. This avoids the need to frequently change TS, reducing unnecessary burden on the licensee and NRC staff.

Cycle specific parameters approved for Westinghouse units for relocation to the COLR include: (1) moderator temperature coefficient, (2) shutdown bank insertion limits, (3) control bank insertion limits, (4) axial flux difference limits, (5) nuclear heat flux hot channel factor limit (F_Q), (6) nuclear enthalpy rise hot channel factor limit ($F_{\Delta H}^N$), (7) refueling boron concentration limit, and (8) shutdown margin.

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In addition, as described in WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report," the reactor core safety limit curves and cycle specific DNB related parameter limits (RCS flow, temperature and pressure) may also be relocated. To relocate these parameters requires the addition of the Departure From Nucleate Boiling Ratio (DNBR) design limit, the fuel centerline melt temperature limit, and the retention of the minimum reactor coolant flow design limit in the TS.

Provided with this license amendment request is a description and justification of changes, safety evaluation, and no significant hazards determination. Also provided, for information only, is the COLR for the Kewaunee unit. The parameter limits contained within the COLR include limits applicable to present operation.

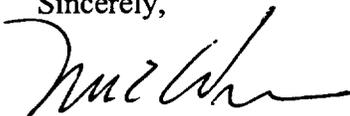
Several editorial changes are also included in this submittal. These editorial changes include capitalization of defined words, proper outline numbering, adding proper "if - then" action statements, and correcting punctuation errors.

Attachment 1 to this letter contains a description of the changes, a safety evaluation, a significant hazards determination, and environmental considerations for the proposed changes, attachment 2 contains the strike-out Technical Specification pages, attachment 3 contains the affected Technical Specification pages as revised, attachment 4 contains the bases pages, and attachment 5 contains the core operating limits report.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects, these statements are not based entirely on my personal knowledge, but on information furnished by cognizant NMC employees and consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on July 26, 2002.

Sincerely,



Mark E. Warner
Site Vice President

GOR

Attachments

cc - US NRC Region III
US NRC Senior Resident Inspector
State of Wisconsin

ATTACHMENT 1

Letter from Mark E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 185

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

Description of Proposed Changes to Technical Specification

Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," endorsed the removal of cycle-specific parameter limits from the Technical Specifications (TS). This removal is allowed provided the parameters are determined using NRC approved methodologies and changes to the parameters are reported to the NRC. This alternative consists of three separate actions to allow modifying KNPP's TS. These actions, which have been completed as shown in this submittal, are:

- the addition of the definition of a named formal report that includes the values of cycle-specific parameter limits that have been established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis,
- the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information, and
- the modification of individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report.

Cycle specific parameters previously approved for Westinghouse units for relocation to the COLR include:

- 1) moderator temperature coefficient,
- 2) shutdown bank insertion limits,
- 3) control bank insertion limits,
- 4) axial flux difference limits,
- 5) nuclear heat flux hot channel factor limit (F_Q),
- 6) nuclear enthalpy rise hot channel factor limit ($F_{\Delta H}^N$),
- 7) refueling boron concentration limit, and
- 8) shutdown margin.

In addition, as described in WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report," the reactor core safety limit curves and cycle specific departure from nucleate boiling (DNB) parameter limits (RCS flow, pressure and temperature) as well as the overtemperature ΔT and overpower ΔT trip setpoint parameter values may also be relocated. This relocation requires the addition of the departure from nucleate boiling ratio (DNBR) design limit, the fuel centerline melt temperature limit, and the retention of the minimum reactor coolant flow design limit in the TS.

Therefore, to allow for the relocation of those parameter allowed by GL 88-16 and WCAP-14483-A, the following changes are made:

- 1) An addition of a definition of the formal report that includes the values of cycle-specific parameter limits is added to the definition section of the TS as TS 1.0.q, "Core Operating Limits Report (COLR)
- 2) An addition of the requirements for the COLR is added to TS Section 6.9, "Reporting Requirements," as TS 6.9.a.4, "Core Operating Limits Report (COLR)."
- 3) Various individual TS have been modified to state that their cycle-specific parameters shall be maintained within the limits provided in the COLR.
- 4) Relocation to the COLR of the departure-from-nucleate-boiling (DNB) parameters of the reactor coolant system (RCS) average coolant temperature (T_{avg}), RCS flow rate, and pressurizer pressure, as well as the overtemperature ΔT ($OT\Delta T$) and overpower ΔT ($OP\Delta T$) trip setpoint parameter values. Also, relocate the reactor core safety limit and replace it with the fuel departure-from-nucleate-boiling ratio (DNBR) limit and the fuel centerline temperature limit.

The definition of Shutdown Margin is being moved from TS 3.10.a to the definition section. To move more towards the industry standards, KNPP TS are transitioning towards improved standard technical specification (ISTS). As a part of this transition the definition of shutdown margin is changed to that of ISTS. The ISTS definition is modified to conform with KNPP specific modes and terminology and to add clarity to the phrase "independent means" stated in ISTS. This relocation is considered administrative.

Several editorial changes are also included in this submittal. These editorial changes include capitalization of defined words, proper outline numbering, adding proper "if - then" action statements, and correcting punctuation errors.

Basis for Changes

Addition of the definition of the Core Operating Limits Report is added to the TS definition section as TS 1.0.q.

The addition of the COLR definition is in accordance with the guidance and requirements provided in GL 88-16. The definition is as provided in NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2.

This change is administrative only, as it defines the COLR and its use. The definition of the COLR does not, in itself, impose new requirements on the operation of the Kewaunee Nuclear Plant. Therefore, this change is considered administrative. It is in accordance with and required by the guidance of GL 88-16.

TS Section 6.9, "Reporting Requirements." Addition of COLR reporting requirements contained in TS 6.9.a.4.

The addition of an administrative reporting requirement which defines the approved methodologies to be used for Kewaunee is proposed. This change includes the requirements from enclosure 2 of GL 88-16. On page 2 it states:

A new administrative reporting requirement shall be added to existing reporting requirements as follows.

[CORE] OPERATING LIMITS REPORT

[6.9.x] [Core] operating limits shall be established and documented in the [CORE] OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. (If desired, the individual specifications that address [core] operating limits may be referenced.) The analytical methods used to determine the [core] operating limits shall be those previously reviewed and approved by the NRC in [identify the Topical Report(s) by number, title, and date, or identify the staff's safety evaluation report for a plant-specific methodology by NRC letter and date]. The [core] operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The [CORE] OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Deck with copies to the Regional Administrator and the Resident Inspector.

KNPP TS 6.9.a.4, was designed to meet the intent of this requirement. TS 6.9.a.4.A states the requirements for establishing and documenting the core operating limits for each reload or partial reload cycle. TS 6.9.a.4.A also lists those TS parameters that are included and relocated to the COLR. TS 6.9.a.4.B states the requirements for the methods used to determine those parameters or the methods that are used to verify the use of those parameter meets KNPP's safety analysis requirements. TS 6.9.a.4.C states that the core operating limits shall be determined such that all applicable limits of the safety analysis are met. In addition, TS 6.9.a.4.D, states that the COLR shall be provide to the NRC upon issuance.

The reporting condition proposed for addition to the TS is in accordance with the guidance in GL 88-16. The addition of this requirement ensures that the cycle-specific operating parameters are determined in accordance with NRC approved methodologies and continue to meet the acceptance criteria of the safety analyses as described in the updated Final Safety Analysis Report. Thus, reactor operation will continue to meet all regulatory and design basis requirements.

Application of each of the approved methodologies and the corresponding cycle specific parameter and are shown in Table 1. These methodologies have been approved for the use at KNPP. The addition of these methodologies to the Technical Specifications therefore, does not impose any new requirements on the operation of KNPP and is administrative.

Modify Individual TS to Specify that their limits are contained in the COLR

This change is part of the requirements for implementation of a COLR in accordance with GL 88-16. TS 6.9.a.4.A lists the TS in which the limits have been replaced by a statement directing the user to the COLR.

Relocation of the Safety Limit and DNB parameters

Relocation of the Reactor Core Safety Limit curves was approved and is subject to the conditions defined in WCAP-14483-P-A. In accordance with the conditions of approval for the WCAP, it is proposed to add the DNBR design limits and fuel centerline melt temperature limits to the Technical Specifications and relocate the reactor core safety limit curves to the COLR. Operation within the acceptable region of the reactor core safety limit curves ensures these safety limits are met. The following DNBR limits will be added to the Technical Specifications in section 2.0:

The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation

In addition, the following peak fuel centerline temperature limit will be added:

The peak fuel centerline temperature shall be maintained < 4700 °F.

In accordance WCAP-14483-A and associated conditions of approval, the cycle specific DNB parameters related to reactor coolant system temperature, pressure and flow will be relocated to the COLR. The following reactor coolant system design minimum flow limits will be retained in the Technical Specifications:

During steady-state power operation, reactor coolant flow rate shall be $\geq 93,000$ gallons per minute average per loop.

Relocation of the DNB parameter limits to the COLR is in accordance with WCAP-14483-A. The methodology accepted for reference is as described in the table previously. The flow limits are as approved with Amendment 157 on September 5, 2001.

Relocating the definition of Shutdown Margin from TS 3.10.a to definition TS 1.0.r

This change is done to place the definition in the section which the user would first look to find its meaning.

Additional changes are being made to conform to the description used in the improved standard technical specification (ISTS). This definition provides a clearer description of which rods are included in the calculation. The ISTS definition is also modified to provide additional clarity and KNPP mode terminology. This change is consistent with current industry standards (NUREG 1431, Rev 2), modified to be plant specific for the KNPP.

The clearer description comes from ISTS use of the term RCCA. Rod control cluster assemblies (RCCA) refers to all reactivity control rods in the reactor. In KNPP's current definition, the term used is control rods. This use of the term control rods could lead to some confusion as to whether the shutdown rods are included. Although the shutdown rods were included, this change provides clarity. An additional modification when using the ISTS definition is a substep which is similar to a condition in the KNPP TS but is worded different. In KNPP TS the definition refers to calculating the reactivity at the HOT SHUTDOWN condition. In ISTS, this conditions wording is changed to state the condition at which the fuel and moderator temperature conditions are at the nominal zero power design level, which is KNPP's HOT SHUTDOWN conditions or the zero power program level.

An additional reference is being added to the ISTS definition modified for the KNPP. In the definition for shutdown margin a statement is made allowing the licensee to include the reactivity of the highest worth RCCA if all rods are verified fully inserted by two independent means. In KNPP TS 3.10.e, reference is made to indication available for the operator to verify rod position relative to rod bank position. To avoid confusion of the operator, this definition is modified to reference TS 3.10.e. By referencing this TS the operators know which independent means of rod indications can be used. This change is considered administrative, as it is consistent with KNPP's current licensing basis, stated in TS 3.10.e.

Editorial Changes

Several editorial changes are also included in this submittal. These editorial changes include capitalization of defined words, proper outline numbering, adding proper "if – then" action statements, and correcting punctuation errors.

Safety Evaluation

NRC Generic Letter 88-16 provides guidance for licensees, allowing relocation of cycle dependent variables from the Technical Specifications (TS) to a Core Operating Limits Report (COLR). Relocation of these variables is allowed if they are determined with NRC-approved methodologies and the TS contain reporting requirements described in the GL. The variables removed from the TS and relocated to the COLR can then be changed via the appropriate regulatory change mechanisms, principally 10 CFR 50.59. This avoids the need to frequently change TS. Appropriate safety limits are maintained in the Specifications. Changes to the COLR parameters are reported to the NRC so that they can be monitored and trended.

Cycle specific parameters previously approved for Westinghouse units for relocation to the COLR include: (1) moderator temperature coefficient, (2) shutdown bank insertion limits, (3) control bank insertion limits, (4) axial flux difference limits, (5) nuclear heat flux hot channel factor limit (F_Q), (6) nuclear enthalpy rise hot channel factor limit ($F_{\Delta H}^N$), (7) refueling boron concentration limit, and (8) shutdown margin. In addition, as described in WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report," the reactor core safety limit curves and specific DNB related parameters may also be relocated with the addition of the departure from nucleate boiling ratio (DNBR) design limit, the fuel centerline melt temperature limit, and the retention of the minimum reactor coolant flow design limit in the TS.

This amendment request adds the definition of the COLR, and the reporting requirement as recommended by NUREG-1431, meeting the guidance in Generic Letter 88-16. The cycle specific parameters that are relocated will be determined by analyses performed in accordance with NRC approved methodologies. The methodologies are listed in the added reporting requirement section and are approved for use at KNPP and/or the KNPP Updated Safety Analysis Report (USAR).

Relocation of the specific DNB parameter limits and Reactor Core Safety Limit Curves is in accordance with approved WCAP-14483-A. Specific safety limits related to DNBR, fuel centerline melt temperature and reactor coolant system flow are added or retained in the Specifications with this change.

Therefore, the Specifications continue to meet the criteria of 10 CFR 50.36, Criterion 2, in that appropriate safety limits are maintained in the Technical Specifications. The reporting requirement ensures that the limits on the safety analyses continue to be met and that all changes to the design are performed in accordance with approved methodologies and reviewed under existing regulatory change mechanisms, principally 10 CFR 50.59. As such, these changes are essentially administrative and continue to ensure that KNPP is operated within the limits of its design and analyses.

Operation of KNPP in accordance with these proposed changes continue to ensure safe operation of the plant and will not be inimical to the health and safety of the public.

No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, Nuclear Management Company, LLC, requests amendments to Facility Operating Licenses DPR-43 for Kewaunee Nuclear Plant. The purpose of the proposed amendments is to implement a Core Operating Limits Report (COLR) at the Kewaunee Nuclear Plant (KNPP).

In accordance with the requirements of 10 CFR 50.91, NMC has evaluated operation of the Kewaunee Nuclear Plant in accordance with the proposed changes against the standards of 10 CFR 50.92. Operation of the Kewaunee Nuclear Plant in accordance with the proposed changes results in no significant hazards consideration. Our evaluation and basis for this conclusion follows.

1. Operation of the Kewaunee Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The proposed changes relocate certain cycle specific parameters from the Technical Specifications to a Core Operating Limits Report (COLR) or are administrative in nature. Appropriate design and safety limits are retained or added to the Specifications thereby meeting the requirements of 10 CFR 50.36. Specific, approved methodologies used to determine and evaluate the parameter requirements are added to the Specifications and a reporting requirement is added to ensure the NRC is apprised of all changes. Approved methodologies are required to be used to evaluate and change parameters, and appropriate safety and design limits are maintained in the Technical Specifications. Thus, operation of KNPP will continue to meet all design and safety analysis requirements. Therefore, neither the probability nor consequences of an accident previously evaluated can be increased.

2. Operation of the Kewaunee Nuclear Plant in accordance with the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.

Operation of KNPP, in accordance with the proposed changes, will continue to meet all design and safety limits. Appropriate design and safety limits continue to be controlled within the Technical Specifications. These changes will not result in a change to the design and safety limits under which KNPP operation has been determined to be acceptable. Therefore, these changes cannot result in a new or different kind of accident from any accident previously evaluated.

3. Operation of the Kewaunee Nuclear Plant in accordance with the proposed amendment does not result in a significant reduction in a margin of safety.

Appropriate safety limits continue to be controlled by the Specifications. Changes to cycle specific parameters related to these limits will be accomplished using NRC approved methodologies, thereby ensuring operation will continue within the bounds of the existing safety analyses including all applicable margins of safety. Therefore, operation in accordance with the proposed changes cannot result in a significant reduction in a margin of safety.

Operation of the Kewaunee Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated; does not create a new or different kind of accident from any accident previously evaluated; and, does not result in a significant reduction in a margin of the safety. Therefore, operation in accordance with the proposed amendments involves no significant hazards consideration

Environmental Considerations

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

TABLE 1

Relocated Parameter and Approved Methods

Parameter	NRC Approved Methodology
Reactor Core Safety Limits	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p> <p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>
Shutdown Margin	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p>
Moderator Temperature Coefficient	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p>
Shutdown Bank Insertion Limits	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p> <p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>

TABLE 1

Relocated Parameter and Approved Methods

Parameter	NRC Approved Methodology
Control Bank Insertion Limits	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p> <p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>
Height Dependent Heat Flux Hot Channel Factor (F_Q)	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p> <p>Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.</p> <p>N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.</p> <p>C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.</p>

TABLE 1

Relocated Parameter and Approved Methods

Parameter	NRC Approved Methodology
	<p>ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991</p> <p>XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.</p>
<p>Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)</p>	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, Dated August 21, 1979, Report Date September 29, 1978</p> <p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) Dated September 10 2001.</p> <p>XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.</p> <p>ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.</p> <p>EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999</p>

TABLE 1

Relocated Parameter and Approved Methods

Parameter	NRC Approved Methodology
Axial Flux Difference	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p> <p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>
Overtemperature ΔT	<p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>
Overpower ΔT	<p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>
DNB Limits	<p>KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.</p>
Refueling Boron Concentration	<p>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978</p>

ATTACHMENT 2

Letter from Mark E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 185

Strikeout TS Pages:

TS i, TS ii, TS iv, TS vi

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TS 1.0-6

TS 2.1-1

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Note:

⁽¹⁾ Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

TECHNICAL SPECIFICATIONS AND BASES

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

a. QUADRANT-TO-AVERAGE POWER TILT RATIO

The QUADRANT-TO-AVERAGE POWER TILT RATIO is defined as the ratio of maximum-to-average of the upper excore detector currents or that of the lower excore detector currents, whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

b. SAFETY LIMITS

SAFETY LIMITS are the necessary quantitative restrictions placed upon those process variables that must be controlled in order to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

c. LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are setpoints for automatic protective devices responsive to the variables on which SAFETY LIMITS have been placed. These setpoints are so chosen that automatic protective actions will correct the most severe, anticipated abnormal situation so that a SAFETY LIMIT is not exceeded.

d. LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION are those restrictions on reactor operation, resulting from equipment performance capability, that must be enforced to ensure safe operation of the facility.

e. OPERABLE-OPERABILITY

A system or component is OPERABLE or has OPERABILITY when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the LIMITING CONDITIONS FOR OPERATION defined in TS 3.0; and (2) it has been tested periodically in accordance with TS 4.0 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. OPERATING

A system or component is considered to be OPERATING when it is performing the intended function in the intended manner.

g. CONTAINMENT SYSTEM INTEGRITY

CONTAINMENT SYSTEM INTEGRITY is defined to exist when:

1. The nonautomatic Containment System isolation valves and blind flanges are closed, except as provided in TS 3.6.b.
2. The Reactor Containment Vessel and Shield Building equipment hatches are properly closed.
3. At least ONE door in both the personnel and the emergency airlocks is properly closed.
4. The required automatic Containment System isolation valves are OPERABLE, except as provided in TS 3.6.b.
5. All requirements of TS 4.4 with regard to Containment System leakage and test frequency are satisfied.
6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of TS 3.6.c.

h. PROTECTIVE INSTRUMENTATION LOGIC

1. PROTECTION SYSTEM CHANNEL

A PROTECTION SYSTEM CHANNEL is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

2. LOGIC CHANNEL

A LOGIC CHANNEL is a matrix of relay contacts which operate in response to PROTECTIVE SYSTEM CHANNEL signals to generate a protective action signal.

3. DEGREE OF REDUNDANCY

DEGREE OF REDUNDANCY is defined as the difference between the number of OPERATING channels and the minimum number of channels which, when tripped, will cause an automatic shutdown.

4. PROTECTION SYSTEM

The PROTECTION SYSTEM consists of both the Reactor PROTECTION SYSTEM and the Engineered Safety Features System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through actuated device) which are required to operate in order to produce the required protective function. Tests of PROTECTION SYSTEM will be considered acceptable when tests are run in part and it can be shown that all parts satisfy the requirements of the system.

i. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent channels measuring the same variable.

2. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output as necessary, such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel monitors. Calibration shall encompass the entire channel, including alarm and/or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

4. SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

5. FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals in Table TS 1.0-1.

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq -5\%$	≤ 140	~ 0
COLD SHUTDOWN	$\leq -1\%$	≤ 200	~ 0
INTERMEDIATE SHUTDOWN	(1)	$> 200 < 540$	~ 0
HOT SHUTDOWN	(1)	≥ 540	~ 0
HOT STANDBY	$< 0.25\%$	$\sim T_{oper}$	< 2
OPERATING	$< 0.25\%$	$\sim T_{oper}$	≥ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to <u>Figure TS 3.10-1</u> the required SHUTDOWN MARGIN as specified in the <u>Core Operating Limits Report</u> .			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. RADIOLOGICAL EFFLUENTS

1. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS 6.16.b, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by TS 6.9.b.1 and TS 6.9.b.2.

3. PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, federal and state regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

4. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

5. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (μ Ci/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0361	I-132
0.2703	I-133
0.0169	I-134
0.0838	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

- a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and
- b. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the program temperature.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES operation.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

- a. The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits ~~shown in Figure TS 2.1-1~~ specified in the COLR. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.
- b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation
- c. The peak fuel centerline temperature shall be maintained < 4700 °F

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

APPLICABILITY

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

OBJECTIVE

To prevent the principal process variables from exceeding a SAFETY LIMIT.

SPECIFICATION

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

A. Source Range (high setpoint)	within span of source range instrumentation
B. Intermediate range (high setpoint)	$\leq 40\%$ of RATED POWER
C. Power range (low setpoint)	$\leq 25\%$ of RATED POWER
D. Power range (high setpoint)	$\leq 109\%$ of RATED POWER
E. Power range fast flux rate trip (positive)	$15\% \Delta q / 5 \text{ sec}$
F. Power range fast flux rate trip (negative)	$10\% \Delta q / 5 \text{ sec}$

2. Pressurizer

A. High pressurizer pressure	$\leq 2385 \text{ psig}$
B. Low pressurizer pressure	$\geq 1875 \text{ psig}$
C. High pressurizer water level	$\leq 90\%$ of full scale

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f(\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, 0.5%°F

T = Average temperature, °F

T' = 567.3[*]°F

P = Pressurizer pressure, psig

P' = 2235[*] psig

K₁ = 4.11[*]

K₂ = 0.0090[*]

K₃ = 0.000566[*]

τ_1 = 30[*] sec.

τ_2 = 4[*] sec.

f(ΔI) = An even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

(a)1. For $q_t - q_b$ within 12[*], +9[*] %, f(ΔI) = 0.

(b)2. For each percent that the magnitude of $q_t - q_b$ exceeds +9[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.5[*] % of RATED POWER.

(c)3. For each percent that the magnitude of $q_t - q_b$ exceed -12[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of 4.5[*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 S}{\tau_3 S + 1} T - K_6 (T - T') - f(\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, $\frac{\circ}{\%}$ F

T = Average Temperature, \circ F

T' = 567.3 $\frac{\circ}{\%}$ F

K_4 \leq 1.10 $\frac{\circ}{\%}$

K_5 \geq 0.0275 $\frac{\circ}{\%}$ for increasing T; θ $\frac{\circ}{\%}$ for decreasing T

K_6 \geq 0.002 $\frac{\circ}{\%}$ for T > T'; θ $\frac{\circ}{\%}$ for T < T'

τ_3 = 40 $\frac{\circ}{\%}$ sec.

f(ΔI) = As defined above

Note: $\frac{\circ}{\%}$ As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop \geq 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint \geq 55.0 Hz

2. Low voltage setpoint \geq 75% of normal voltage

5. Steam Generators

Low-low steam generator water level \geq 5% of narrow range instrument span.

6. Reactor Trip Interlocks

Protective instrumentation settings for reactor trip interlocks shall be as follows:

- A. Above 10% of RATED POWER, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.
- B. Above 10% of RATED POWER, the single-loop loss-of-flow trip is made functional.

7. Other Trips

Undervoltage \geq 75% of normal voltage

Turbine trip

Manual trip

Safety injection trip (Refer to Table TS 3.5-1 for trip settings)

FIGURE ~~TS 2.1-1~~

**TS FIGURE 2.1-1
DELETED**

3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Reactor Coolant System (RCS).

OBJECTIVE

To specify those LIMITING CONDITIONS FOR OPERATION of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

- A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.
- B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.
- C. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures $\leq 200^{\circ}\text{F}$ unless the secondary water temperature of each steam generator is $< 100^{\circ}\text{F}$ above each of the RCS cold leg temperatures.

2. Decay Heat Removal Capability

- A. At least ~~Two~~ Two of the following ~~Four~~ Four heat sinks shall be OPERABLE whenever the average reactor coolant temperature is $\leq 350^{\circ}\text{F}$ but $> 200^{\circ}\text{F}$.
 - 1. Steam Generator 1A
 - 2. Steam Generator 1B
 - 3. Residual Heat Removal Train A
 - 4. Residual Heat Removal Train B

If less than the above number of required heat sinks are OPERABLE, then corrective action shall be taken immediately to restore the minimum number to the OPERABLE status.

B. Two residual heat removal trains shall be OPERABLE whenever the average reactor coolant temperature is $\leq 200^{\circ}\text{F}$ and irradiated fuel is in the reactor, except when in the REFUELING MODE with the minimum water level above the top of the vessel flange ≥ 23 feet, one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

a) One OPERABLE residual heat removal pump

b) One OPERABLE residual heat removal heat exchanger

c) An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shutdown situations. This flow path shall be capable of taking suction from the appropriate Reactor Coolant System hot leg and returning to the Reactor Coolant System.

2. If one residual heat removal train is inoperable, then corrective action shall be taken immediately to return it to the OPERABLE status.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be OPERABLE whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge of the charging pump are set for test pressure plus 35 psi to protect the system.

B. Both pressurizer safety valves shall be OPERABLE whenever the reactor is critical.

4. Pressure Isolation Valves

A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device during OPERATING and HOT STANDBY MODES, except as specified in 3.1.a.4.B. Valve leakage shall not exceed the amounts indicated.

B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.⁽¹⁾

C. If TS 3.1.a.4.A and TS 3.1.a.4.B cannot be met, then an orderly shutdown shall be initiated and the reactor shall be in the HOT SHUTDOWN condition within the next 4 hours, the INTERMEDIATE SHUTDOWN condition in the next 6 hours and the COLD SHUTDOWN condition within the next 24 hours.

⁽¹⁾ Manual valves shall be locked in the closed position. Motor operated valves shall be placed in the closed position with their power breakers locked out.

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

A. Two PORVs and their associated block valves shall be OPERABLE during HOT STANDBY and OPERATING modes.

1. With one or both PORVs inoperable because of excessive seat leakage, within ~~4~~one hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, action shall be initiated to:

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

2. With one PORV inoperable due to causes other than excessive seat leakage, within ~~4~~one hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; Restore the PORV to OPERABLE status within the following 72 hours or action shall be initiated to:

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

3. With both PORVs inoperable due to causes other than excessive seat leakage, within ~~4~~one hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and

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

4. With one block valve inoperable, within ~~4~~one hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the block valve to OPERABLE status within 72 hours; otherwise action shall be initiated to:

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

5. With both block valves ~~INOPERABLE~~ inoperable, within ~~4~~one hour restore the block valves to OPERABLE status or place their associated PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour; otherwise, action shall be initiated to:

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

6. Pressurizer Heaters

A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is > 350°F.

7. Reactor Coolant Vent System

- A. A reactor coolant vent path from both the reactor vessel head and pressurizer steam space shall be OPERABLE and closed prior to the average RCS temperature being heated > 200°F except as specified in TS 3.1.a.7.B and TS 3.1.a.7.C below.
- B. When the average RCS temperature is > 200°F, any one of the following conditions of inoperability may exist:
1. Both of the parallel vent valves in the reactor vessel vent path are inoperable.
 2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

If OPERABILITY is not restored within 30 days, then within one hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours
- Achieve COLD SHUTDOWN within an additional 36 hours

- C. If no Reactor Coolant System vent paths are OPERABLE, then restore at least one vent path to OPERABLE status within 72 hours. If OPERABILITY is not restored within 72 hours, then within 4-one hour action shall be initiated to:
- Achieve HOT STANDBY within 6 hours
 - Achieve HOT SHUTDOWN within the following 6 hours
 - Achieve COLD SHUTDOWN within an additional 36 hours

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33⁽¹⁾ effective full-power years.
 - A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.
 - B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

⁽¹⁾ Although the curves were developed for 33 EFY, they are limited to 28 EFY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

- C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.
2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.
 3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.
 4. The overpressure protection system for low temperature operation shall be OPERABLE whenever one or more of the RCS cold leg temperatures are \leq 200°F, and the reactor vessel head is installed. The system shall be considered OPERABLE when at least one of the following conditions is satisfied:
 - A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of \leq 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.
 1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within 5-five days or complete depressurization and venting of the RCS through a \geq 6.4 square inch vent within an additional 8-eight hours.
 2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within 8-eight hours.
 - B. A vent pathway shall be provided with an effective flow cross section \geq 6.4 square inches.
 1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, then verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

A. \leq ~~200.20~~ 0.20 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131, and

B. $\leq \frac{91}{E} \frac{\mu\text{Ci}}{\text{cc}}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium (\bar{E} is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is $> 500^\circ\text{F}$.

2. If the reactor is critical or the average temperature is $> 500^\circ\text{F}$:

A. With the specific activity of the reactor coolant $> 0.20 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding the limit shown on Figure TS 3.1-3, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of $< 500^\circ\text{F}$ within ~~6~~ six hours.

B. With the specific activity of the reactor coolant $> \frac{91}{E} \frac{\mu\text{Ci}}{\text{cc}}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature $< 500^\circ\text{F}$ within ~~6~~ six hours.

C. With the specific activity of the reactor coolant $> 0.20 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 or $> \frac{91}{E} \frac{\mu\text{Ci}}{\text{cc}}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every ~~4~~ four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4-four hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, then the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, then the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.
2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 150 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within 4-four hours or be in COLD SHUTDOWN within the next 36 hours.
3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, then operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, then the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.
4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), then the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.
5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is OPERABLE.

e. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

1. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is > 250°F.

CONTAMINANT	NORMAL STEADY-STATE OPERATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	0.10	1.00
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in TS 3.1.e.1 above are exceeded, or if it is anticipated that they may be exceeded, then corrective action shall be taken immediately.
3. If the concentrations of any of the contaminants cannot be controlled within the transient limits of TS 3.1.e.1 above or returned to the normal steady-state limit within 24 hours, then the reactor shall be brought to the COLD SHUTDOWN condition, utilizing normal operating procedures, and the cause shall be ascertained and corrected. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; Otherwise a safety review by the Plant Operations Review Committee shall be made before starting.
4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is ≤ 250°F.

CONTAMINANT	NORMAL CONCENTRATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	Saturated	Saturated
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

5. If the transient limits of TS 3.1.e.4 are exceeded or the concentrations cannot be returned to normal values within 48 hours, then the reactor shall be brought to the COLD SHUTDOWN condition and the cause shall be ascertained and corrected.
6. To meet TS 3.1.e.1 and TS 3.1.e.4 above, reactor coolant pump operation shall be permitted for short periods, provided the coolant temperature does not exceed 250°F.

f. Minimum Conditions for Criticality

1. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.
2. The reactor shall be maintained subcritical by at least 1% $\Delta k/k$ until normal water level is established in the pressurizer.
3. When the reactor is critical ~~and $\leq 60\%$ RATED POWER,~~ the moderator temperature coefficient shall be ~~≤ 5.0 pcm/ $^{\circ}$ F~~ as specified in the COLR, except during LOW POWER PHYSICS TESTING. ~~When the reactor is $> 60\%$ RATED POWER, the moderator temperature coefficient shall be zero or negative.~~
4. ~~The reactor will have a moderator temperature coefficient no less negative than -8 pcm/ $^{\circ}$ F for 95% of the cycle time at full power.~~
 4. If the limits of 3.1.f.3 cannot be met, then power operation may continue provided the following actions are taken:
 - A. Within 24 hours, develop and maintain administrative control rod withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits specified in TS 3.1.f.3. These withdrawal limits shall be in addition to the insertion limits specified in TS 3.10.d.
 - B. If the actions specified in TS 3.1.f.54.A are not satisfied, then be in HOT STANDBY within the next ~~6~~ six hours.

3.8 REFUELING OPERATIONS

APPLICABILITY

Applies to operating limitations during REFUELING OPERATIONS.

OBJECTIVE

To ensure that no incident occurs during REFUELING OPERATIONS that would affect public health and safety.

SPECIFICATION

a. During REFUELING OPERATIONS:

1. Containment Closure

- a. The equipment hatch shall be closed and at least one door in each personnel air lock shall be capable of being closed⁽¹⁾ in 30 minutes or less. In addition, at least one door in each personnel air lock shall be closed when the reactor vessel head or upper internals are lifted.
- b. Each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool shall be monitored continuously.

3. The reactor will be subcritical for 148 hours prior to movement of its irradiated fuel assemblies. Core subcritical neutron flux shall be continuously monitored by at least ~~Two~~ two neutron monitors, each with continuous visual indication in the control room and ~~One~~ one with audible indication in the containment whenever core geometry is being changed. When core geometry is not being changed at least ~~One~~ one neutron flux monitor shall be in service.

4. At least ~~One~~ one residual heat removal pump shall be OPERABLE ~~operable~~.

5. When there is fuel in the reactor, a minimum boron concentration of ~~2200 ppm and a shutdown margin of >5% Akk~~ as specified in the COLR shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor. The required boron concentration shall be verified by chemical analysis daily.

⁽¹⁾ Administrative controls ensure that:

- Appropriate personnel are aware that both personnel air lock doors are open,
- A specified individual(s) is designated and available to close the air lock following a required evacuation of containment, and
- Any obstruction(s) (e.g., cables and hoses) that could prevent closure of an open air lock can be quickly removed.

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.
7. Heavy loads, greater than the weight of a fuel assembly, will not be transported over or placed in either spent fuel pool when spent fuel is stored in that pool. Placement of additional fuel storage racks is permitted, however, these racks may not traverse directly above spent fuel stored in the pools.
8. The containment ventilation and purge system, including the capability to initiate automatic containment ventilation isolation, shall be tested and verified to be operable immediately prior to and daily during REFUELING OPERATIONS.
9. a. The spent fuel pool sweep system, including the charcoal adsorbers, shall be operating during fuel handling and when any load is carried over the pool if irradiated fuel in the pool has decayed less than 30 days. If the spent fuel pool sweep system, including the charcoal adsorber, is not operating when required, fuel movement shall not be started (any fuel assembly movement in progress may be completed).
- b. Performance Requirements
 1. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal.
 2. The results of laboratory carbon sample analysis from spent fuel pool sweep system carbon shall show $\geq 95\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C and 95% RH.
 3. Fans shall operate within $\pm 10\%$ of design flow when tested.
10. The minimum water level above the vessel flange shall be maintained at 23 feet.
11. A dead-load test shall be successfully performed on both the fuel handling and manipulator cranes before fuel movement begins. The load assumed by the cranes for this test must be equal to or greater than the maximum load to be assumed by the cranes during the REFUELING OPERATIONS. A thorough visual inspection of the cranes shall be made after the dead-load test and prior to fuel handling.
12. A licensed senior reactor operator will be on-site and designated in charge of the REFUELING OPERATIONS.
- b. If any of the specified limiting conditions for REFUELING OPERATIONS are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be performed.

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~~ margin shall be at least that ~~shown in Figure TS 3.10-1~~ as specified in the COLR. ~~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 shall be for Siemens Power Corporation Fuel as specified in the COLR.

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

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

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

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

where:

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

_____ K(Z) _____ is the function given in Figure TS 3.10.2

_____ Z _____ is the core height location for the F_q of interest

B. $F_{\Delta H}^N$ Limits shall be for Siemens Power Corporation Fuel as specified in the COLR.

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

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

where:

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

2. If, for any measured hot channel factor, the relationships of $F_{\Delta H}^N(Z)$ and $F_{\Delta H}^N$ specified in ~~the COLRTS 3.10.b.1~~ are not true, then 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, then the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.
3. Following initial loading and at regular effective full-power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied.
4. The measured $F_{\Delta H}^N(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core ~~for Siemens Power Corporation fuel: as specified in the COLR.~~

$$F_{\Delta H}^N(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.35)/P \times K(Z) \text{ ----- [Hvy]}$$

$$F_{\Delta H}^N(Z) \times 1.03 \times 1.05 \times V(Z) \leq (2.28)/P \times K(Z) \text{ ----- [Std]}$$

where:

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

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

~~$F_{\Delta H}^N(Z)$~~ is a measured $F_{\Delta H}^N$ 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~~ $F_{\Delta H}^N(Z)$ specified in the COLR 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, then either of the following actions shall be taken:
 -
 -
 -

- i. $F_o^{EQ}(Z)$ shall be increased by ~~an additional 2%~~ the penalty factor specified in the COLR for comparison to the relationship of $F_o^{EQ}(Z)$ specified in the COLR, or $F_o^{EQ}(Z)$ specified in the COLR, or
 - ii. $F_o^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every ~~7~~seven 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_o^{EQ} , the relationships of ~~TS 3.10.b.4~~ $F_o^{EQ}(Z)$ specified in the COLR are not satisfied and the relationships of ~~TS 3.10.b.4~~ $F_o^N(Z)$ and $F_{\Delta H}^N$ specified in the COLR are satisfied, then 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 of ~~specified in TS 3.10.b.4~~ $F_o^{EQ}(Z)$ specified in the COLR 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 of ~~specified in TS 3.10.b.4~~ $F_o^{EQ}(Z)$ specified in the COLR 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~~ $F_o^{EQ}(Z)$ specified in the COLR 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 the target $\pm 5\%$ band about the target flux difference as specified in the COLR.
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 \leq 90% of rated power:
 - A. The indicated axial flux difference may deviate from ~~its the $\pm 5\%$ target band, specified in the COLR,~~ for a maximum of ~~4~~one 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%~~ specified in the COLR. If the cumulative time exceeds ~~4~~one hour, then the reactor power shall be reduced to $\leq 50\%$ of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to $\leq 55\%$ of rated power.

If the indicated axial flux difference exceeds the outer envelope specified in the COLR~~defined above~~, then the reactor power shall be reduced to $\leq 50\%$ of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to $\leq 55\%$ 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 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 ~~4~~one hour cumulative maximum the flux difference may deviate from its target band at a power level $\leq 90\%$ 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, then 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~~two 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, then 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, then 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, then 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~~ to within the limits, specified in the COLR, when the reactor is critical or approaching criticality.
 2. The control banks shall be limited in physical insertion; insertion limits are ~~shown in~~ specified in the COLR ~~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 ~~4 one~~ one hour, initiate boration to restore control bank insertion to within the limits of ~~Figure TS 3.10-3~~ specified in the COLR, and
 - B. Restore control bank insertion to within the limits of ~~Figure TS 3.10-3~~ specified in the COLR within ~~2 two~~ two 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 ~~4 one~~ one 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 as~~ specified in the COLR, 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 $\geq 85\%$ 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 $\geq 85\%$, then the rod will be realigned or the core power peaking factors shall be determined within ~~4 four~~ four hours, and TS 3.10.b applied. If peaking factors are not determined within ~~4 four~~ four hours, the reactor power shall be reduced to $< 85\%$ of rating.

2. When reactor power is $< 85\%$ but $\geq 50\%$ 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 $\geq 50\%$, the rod will be realigned or the core power peaking factors shall be determined within 4-four hours, and TS 3.10.b applied. If the peaking factors are not determined within 4-four 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-eight 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-eight 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~~eight hours after a load change > 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

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

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained ~~< 568.8 °F~~within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR~~> 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 $\geq 93,000$ gallons per minute average per loop and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is ~~< 93,000 gallons per minute per loop~~not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.
2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. DNBR Parameters

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

FIGURE TS 3.10-1

**TS FIGURE 3.10-1
DELETED**

FIGURE TS 3.10-2

**TS FIGURE 3.10-2
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FIGURE TS 3.10-3

**TS FIGURE 3.10-3
DELETED**

TS FIGURE TS 3.10-4

DELETED

TS FIGURE TS 3.10-5

DELETED

TS FIGURE 3.10-6
DELETED

3. Monthly OPERATING Report

Routine reports of OPERATING statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

4. Core Operating Limits Report (COLR)

A. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

- (1) TS 2.1 Reactor Core Safety Limit
- (2) TS 2.3.a.3.A Overtemperature ΔT Setpoint
- (3) TS 2.3.a.3.B Overpower ΔT Setpoint
- (4) TS 3.1.f.3 Moderator Temperature Coefficient (MTC)
- (5) TS 3.1.f.4 Moderator Temperature Coefficient (MTC)
- (6) TS 3.8.a.5 Refueling Boron Concentration
- (7) TS 3.10.a Shutdown Margin
- (8) TS 3.10.b.1.A $F_o^N(Z)$ Limits
- (9) TS 3.10.b.1.B $F_{\Delta H}^N$ Limits
- (10) TS 3.10.b.4 $F_o^{EO}(Z)$ Limits
- (11) TS 3.10.b.5.C.1 $F_o^{EO}(Z)$ penalty
- (12) TS 3.10.b.9 Axial Flux Difference Target Band
- (13) TS 3.10.b.11.A Axial Flux Difference Envelope
- (14) TS 3.10.d.1 Shutdown Bank Insertion Limits
- (15) TS 3.10.d.2 Control Bank Insertion Limits
- (16) TS 3.10.k Core Average Temperature
- (17) TS 3.10.l Reactor Coolant System Pressure
- (18) TS 3.10.m.1 Reactor Coolant Flow

B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

- (1) SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978
- (2) KEWAUNEE NUCLEAR POWER PLANT - REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10, 2001.

- (3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.
- (4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.
- (5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.
- (6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.
- (7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.
- (8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.
- (9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

- A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

- A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.
- (1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

ATTACHMENT 3

Letter from Mark E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 185

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TS 3.1-1 through TS 3.1-9
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Note:

^[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

TECHNICAL SPECIFICATIONS AND BASES

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

a. QUADRANT-TO-AVERAGE POWER TILT RATIO

The QUADRANT-TO-AVERAGE POWER TILT RATIO is defined as the ratio of maximum-to-average of the upper excore detector currents or that of the lower excore detector currents, whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

b. SAFETY LIMITS

SAFETY LIMITS are the necessary quantitative restrictions placed upon those process variables that must be controlled in order to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

c. LIMITING SAFETY SYSTEM SETTINGS

LIMITING SAFETY SYSTEM SETTINGS are setpoints for automatic protective devices responsive to the variables on which SAFETY LIMITS have been placed. These setpoints are so chosen that automatic protective actions will correct the most severe, anticipated abnormal situation so that a SAFETY LIMIT is not exceeded.

d. LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION are those restrictions on reactor operation, resulting from equipment performance capability, that must be enforced to ensure safe operation of the facility.

e. OPERABLE-OPERABILITY

A system or component is OPERABLE or has OPERABILITY when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the LIMITING CONDITIONS FOR OPERATION defined in TS 3.0; and (2) it has been tested periodically in accordance with TS 4.0 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. OPERATING

A system or component is considered to be OPERATING when it is performing the intended function in the intended manner.

g. CONTAINMENT SYSTEM INTEGRITY

CONTAINMENT SYSTEM INTEGRITY is defined to exist when:

1. The nonautomatic Containment System isolation valves and blind flanges are closed, except as provided in TS 3.6.b.
2. The Reactor Containment Vessel and Shield Building equipment hatches are properly closed.
3. At least ONE door in both the personnel and the emergency airlocks is properly closed.
4. The required automatic Containment System isolation valves are OPERABLE, except as provided in TS 3.6.b.
5. All requirements of TS 4.4 with regard to Containment System leakage and test frequency are satisfied.
6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of TS 3.6.c.

h. PROTECTIVE INSTRUMENTATION LOGIC

1. PROTECTION SYSTEM CHANNEL

A PROTECTION SYSTEM CHANNEL is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

2. LOGIC CHANNEL

A LOGIC CHANNEL is a matrix of relay contacts which operate in response to PROTECTIVE SYSTEM CHANNEL signals to generate a protective action signal.

3. DEGREE OF REDUNDANCY

DEGREE OF REDUNDANCY is defined as the difference between the number of OPERATING channels and the minimum number of channels which, when tripped, will cause an automatic shutdown.

4. PROTECTION SYSTEM

The PROTECTION SYSTEM consists of both the Reactor PROTECTION SYSTEM and the Engineered Safety Features System. The PROTECTION SYSTEM encompasses all electric and mechanical devices and circuitry (from sensors through actuated device) which are required to operate in order to produce the required protective function. Tests of PROTECTION SYSTEM will be considered acceptable when tests are run in part and it can be shown that all parts satisfy the requirements of the system.

i. INSTRUMENTATION SURVEILLANCE

1. CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent channels measuring the same variable.

2. CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is OPERABLE, including alarm and/or trip initiating action.

3. CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output as necessary, such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel monitors. Calibration shall encompass the entire channel, including alarm and/or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

4. SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

5. FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals in Table TS 1.0-1.

j. MODES

MODE	REACTIVITY $\Delta k/k$	COOLANT TEMP T_{avg} °F	FISSION POWER %
REFUELING	$\leq -5\%$	≤ 140	~ 0
COLD SHUTDOWN	$\leq -1\%$	≤ 200	~ 0
INTERMEDIATE SHUTDOWN	(1)	$> 200 < 540$	~ 0
HOT SHUTDOWN	(1)	≥ 540	~ 0
HOT STANDBY	$< 0.25\%$	$\sim T_{oper}$	< 2
OPERATING	$< 0.25\%$	$\sim T_{oper}$	≥ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.			

k. REACTOR CRITICAL

The reactor is said to be critical when the neutron chain reaction is self-sustaining.

l. REFUELING OPERATION

REFUELING OPERATION is any operation involving movement of reactor vessel internal components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

m. RATED POWER

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. REPORTABLE EVENT

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. RADIOLOGICAL EFFLUENTS

1. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS 6.16.b, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by TS 6.9.b.1 and TS 6.9.b.2.

3. PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, federal and state regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

4. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

5. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (μ Ci/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0361	I-132
0.2703	I-133
0.0169	I-134
0.0838	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

- a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and
- b. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the program temperature.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

- a. The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.
- b. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the HTP DNB correlation
- c. The peak fuel centerline temperature shall be maintained < 4700 °F

**TS FIGURE 2.1-1
DELETED**

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

APPLICABILITY

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

OBJECTIVE

To prevent the principal process variables from exceeding a SAFETY LIMIT.

SPECIFICATION

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

A. Source Range (high setpoint)	within span of source range instrumentation
B. Intermediate range (high setpoint)	$\leq 40\%$ of RATED POWER
C. Power range (low setpoint)	$\leq 25\%$ of RATED POWER
D. Power range (high setpoint)	$\leq 109\%$ of RATED POWER
E. Power range fast flux rate trip (positive)	$15\% \Delta q / 5 \text{ sec}$
F. Power range fast flux rate trip (negative)	$10\% \Delta q / 5 \text{ sec}$

2. Pressurizer

A. High pressurizer pressure	$\leq 2385 \text{ psig}$
B. Low pressurizer pressure	$\geq 1875 \text{ psig}$
C. High pressurizer water level	$\leq 90\%$ of full scale

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f(\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average temperature, °F

T' = [*]°F

P = Pressurizer pressure, psig

P' = [*] psig

K₁ = [*]

K₂ = [*]

K₃ = [*]

τ_1 = [*] sec.

τ_2 = [*] sec.

f(ΔI) = An even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f(\Delta I) = 0$.
2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.
3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f(\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average Temperature, °F

T' = [*]°F

K₄ ≤ [*]

K₅ ≥ [*] for increasing T; [*] for decreasing T

K₆ ≥ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

f(ΔI) = As defined above

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop $\geq 90\%$ of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint ≥ 55.0 Hz

2. Low voltage setpoint $\geq 75\%$ of normal voltage

5. Steam Generators

Low-low steam generator water level $\geq 5\%$ of narrow range instrument span.

6. Reactor Trip Interlocks

Protective instrumentation settings for reactor trip interlocks shall be as follows:

- A. Above 10% of RATED POWER, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.
- B. Above 10% of RATED POWER, the single-loop loss-of-flow trip is made functional.

7. Other Trips

Undervoltage \geq 75% of normal voltage

Turbine trip

Manual trip

Safety injection trip (Refer to Table TS 3.5-1 for trip settings)

3.1 REACTOR COOLANT SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Reactor Coolant System (RCS). |

OBJECTIVE

To specify those LIMITING CONDITIONS FOR OPERATION of the Reactor Coolant System which must be met to ensure safe reactor operation.

SPECIFICATIONS

a. Operational Components

1. Reactor Coolant Pumps

- A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.
- B. When the reactor is in the OPERATING mode, except for low power tests, both reactor coolant pumps shall be in operation.
- C. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures $\leq 200^{\circ}\text{F}$ unless the secondary water temperature of each steam generator is $< 100^{\circ}\text{F}$ above each of the RCS cold leg temperatures.

2. Decay Heat Removal Capability

- A. At least two of the following four heat sinks shall be OPERABLE whenever the average reactor coolant temperature is $\leq 350^{\circ}\text{F}$ but $> 200^{\circ}\text{F}$. |
 - 1. Steam Generator 1A
 - 2. Steam Generator 1B
 - 3. Residual Heat Removal Train A
 - 4. Residual Heat Removal Train B

If less than the above number of required heat sinks are OPERABLE, then |
corrective action shall be taken immediately to restore the minimum number to the OPERABLE status.

B. Two residual heat removal trains shall be OPERABLE whenever the average reactor coolant temperature is $\leq 200^{\circ}\text{F}$ and irradiated fuel is in the reactor, except when in the REFUELING MODE with the minimum water level above the top of the vessel flange ≥ 23 feet, one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

- a) One OPERABLE residual heat removal pump
- b) One OPERABLE residual heat removal heat exchanger
- c) An OPERABLE flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shutdown situations. This flow path shall be capable of taking suction from the appropriate Reactor Coolant System hot leg and returning to the Reactor Coolant System.

2. If one residual heat removal train is inoperable, then corrective action shall be taken immediately to return it to the OPERABLE status.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be OPERABLE whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power-operated relief valves and the safety valve on the discharge of the charging pump are set for test pressure plus 35 psi to protect the system.

B. Both pressurizer safety valves shall be OPERABLE whenever the reactor is critical.

4. Pressure Isolation Valves

A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device during OPERATING and HOT STANDBY MODES, except as specified in 3.1.a.4.B. Valve leakage shall not exceed the amounts indicated.

B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.⁽¹⁾

C. If TS 3.1.a.4.A and TS 3.1.a.4.B cannot be met, then an orderly shutdown shall be initiated and the reactor shall be in the HOT SHUTDOWN condition within the next 4 hours, the INTERMEDIATE SHUTDOWN condition in the next 6 hours and the COLD SHUTDOWN condition within the next 24 hours.

⁽¹⁾ Manual valves shall be locked in the closed position. Motor operated valves shall be placed in the closed position with their power breakers locked out.

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

A. Two PORVs and their associated block valves shall be OPERABLE during HOT STANDBY and OPERATING modes.

1. With one or both PORVs inoperable because of excessive seat leakage, within one hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, action shall be initiated to:

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

2. With one PORV inoperable due to causes other than excessive seat leakage, within one hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve. Restore the PORV to OPERABLE status within the following 72 hours or action shall be initiated to:

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

3. With both PORVs inoperable due to causes other than excessive seat leakage, within one hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and

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

4. With one block valve inoperable, within one hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the block valve to OPERABLE status within 72 hours; otherwise action shall be initiated to:

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

5. With both block valves inoperable, within one hour restore the block valves to OPERABLE status or place their associated PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour; otherwise, action shall be initiated to:

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

6. Pressurizer Heaters

A. At least one group of pressurizer heaters shall have an emergency power supply available when the average RCS temperature is $> 350^{\circ}\text{F}$.

7. Reactor Coolant Vent System

- A. A reactor coolant vent path from both the reactor vessel head and pressurizer steam space shall be OPERABLE and closed prior to the average RCS temperature being heated > 200°F except as specified in TS 3.1.a.7.B and TS 3.1.a.7.C below.
- B. When the average RCS temperature is > 200°F, any one of the following conditions of inoperability may exist:
 1. Both of the parallel vent valves in the reactor vessel vent path are inoperable.
 2. Both of the parallel vent valves in the pressurizer vent path are inoperable.

If OPERABILITY is not restored within 30 days, then within one hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours
- Achieve COLD SHUTDOWN within an additional 36 hours

- C. If no Reactor Coolant System vent paths are OPERABLE, then restore at least one vent path to OPERABLE status within 72 hours. If OPERABILITY is not restored within 72 hours, then within one hour action shall be initiated to:
 - Achieve HOT STANDBY within 6 hours
 - Achieve HOT SHUTDOWN within the following 6 hours
 - Achieve COLD SHUTDOWN within an additional 36 hours

b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33⁽¹⁾ effective full-power years.
 - A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.
 - B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

⁽¹⁾ Although the curves were developed for 33 EFY, they are limited to 28 EFY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

- C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.
- 2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.
- 3. The pressurizer cooldown and heatup rates shall not exceed 200°F/hr and 100°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320°F.
- 4. The overpressure protection system for low temperature operation shall be OPERABLE whenever one or more of the RCS cold leg temperatures are \leq 200°F, and the reactor vessel head is installed. The system shall be considered OPERABLE when at least one of the following conditions is satisfied:
 - A. The overpressure relief valve on the Residual Heat Removal System (RHR 33-1) shall have a set pressure of \leq 500 psig and shall be aligned to the RCS by maintaining valves RHR 1A, 1B, 2A, and 2B open.
 - 1. With one flow path inoperable, the valves in the parallel flow path shall be verified open with the associated motor breakers for the valves locked in the off position. Restore the inoperable flow path within five days or complete depressurization and venting of the RCS through a \geq 6.4 square inch vent within an additional eight hours.
 - 2. With both flow paths or RHR 33-1 inoperable, complete depressurization and venting of the RCS through at least a 6.4 square inch vent pathway within eight hours.
 - B. A vent pathway shall be provided with an effective flow cross section \geq 6.4 square inches.
 - 1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, then verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

A. $\leq 0.20 \mu\text{Ci}/\text{gram DOSE EQUIVALENT I-131}$, and

B. $\leq \frac{91}{E} \frac{\mu\text{Ci}}{\text{cc}}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium (\bar{E} is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is $> 500^\circ\text{F}$.

2. If the reactor is critical or the average temperature is $> 500^\circ\text{F}$:

A. With the specific activity of the reactor coolant $> 0.20 \mu\text{Ci}/\text{gram DOSE EQUIVALENT I-131}$ for more than 48 hours during one continuous time interval, or exceeding the limit shown on Figure TS 3.1-3, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of $< 500^\circ\text{F}$ within six hours.

B. With the specific activity of the reactor coolant $> \frac{91}{E} \frac{\mu\text{Ci}}{\text{cc}}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature $< 500^\circ\text{F}$ within six hours.

C. With the specific activity of the reactor coolant $> 0.20 \mu\text{Ci}/\text{gram DOSE EQUIVALENT I-131}$ or $> \frac{91}{E} \frac{\mu\text{Ci}}{\text{cc}}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within four hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, then the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, then the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.
2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 150 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within four hours or be in COLD SHUTDOWN within the next 36 hours.
3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, then operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, then the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.
4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), then the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.
5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is OPERABLE.

e. Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

1. Concentrations of contaminants in the reactor coolant shall not exceed the following limits when the reactor coolant temperature is $> 250^{\circ}\text{F}$.

CONTAMINANT	NORMAL STEADY-STATE OPERATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	0.10	1.00
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in TS 3.1.e.1 above are exceeded, or if it is anticipated that they may be exceeded, then corrective action shall be taken immediately.
3. If the concentrations of any of the contaminants cannot be controlled within the transient limits of TS 3.1.e.1 above or returned to the normal steady-state limit within 24 hours, then the reactor shall be brought to the COLD SHUTDOWN condition, utilizing normal operating procedures, and the cause shall be ascertained and corrected. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise a safety review by the Plant Operations Review Committee shall be made before starting.
4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is $\leq 250^{\circ}\text{F}$.

CONTAMINANT	NORMAL CONCENTRATION (ppm)	TRANSIENT LIMITS (ppm)
A. Oxygen	Saturated	Saturated
B. Chloride	0.15	1.50
C. Fluoride	0.15	1.50

5. If the transient limits of TS 3.1.e.4 are exceeded or the concentrations cannot be returned to normal values within 48 hours, then the reactor shall be brought to the COLD SHUTDOWN condition and the cause shall be ascertained and corrected.
6. To meet TS 3.1.e.1 and TS 3.1.e.4 above, reactor coolant pump operation shall be permitted for short periods, provided the coolant temperature does not exceed 250°F .

f. Minimum Conditions for Criticality

1. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.
2. The reactor shall be maintained subcritical by at least 1% $\Delta k/k$ until normal water level is established in the pressurizer.
3. When the reactor is critical the moderator temperature coefficient shall be as specified in the COLR, except during LOW POWER PHYSICS TESTING.
4. If the limits of 3.1.f.3 cannot be met, then power operation may continue provided the following actions are taken:
 - A. Within 24 hours, develop and maintain administrative control rod withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits specified in TS 3.1.f.3. These withdrawal limits shall be in addition to the insertion limits specified in TS 3.10.d.
 - B. If the actions specified in TS 3.1.f.4.A are not satisfied, then be in HOT STANDBY within the next six hours.

3.8 REFUELING OPERATIONS

APPLICABILITY

Applies to operating limitations during REFUELING OPERATIONS.

OBJECTIVE

To ensure that no incident occurs during REFUELING OPERATIONS that would affect public health and safety.

SPECIFICATION

a. During REFUELING OPERATIONS:

1. Containment Closure

- a. The equipment hatch shall be closed and at least one door in each personnel air lock shall be capable of being closed⁽¹⁾ in 30 minutes or less. In addition, at least one door in each personnel air lock shall be closed when the reactor vessel head or upper internals are lifted.
- b. Each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool shall be monitored continuously.

3. The reactor will be subcritical for 148 hours prior to movement of its irradiated fuel assemblies. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment whenever core geometry is being changed. When core geometry is not being changed at least one neutron flux monitor shall be in service.

4. At least one residual heat removal pump shall be OPERABLE.

5. When there is fuel in the reactor, a minimum boron concentration as specified in the COLR shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor. The required boron concentration shall be verified by chemical analysis daily.

⁽¹⁾ Administrative controls ensure that:

- Appropriate personnel are aware that both personnel air lock doors are open,
- A specified individual(s) is designated and available to close the air lock following a required evacuation of containment, and
- Any obstruction(s) (e.g., cables and hoses) that could prevent closure of an open air lock can be quickly removed.

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.
7. Heavy loads, greater than the weight of a fuel assembly, will not be transported over or placed in either spent fuel pool when spent fuel is stored in that pool. Placement of additional fuel storage racks is permitted, however, these racks may not traverse directly above spent fuel stored in the pools.
8. The containment ventilation and purge system, including the capability to initiate automatic containment ventilation isolation, shall be tested and verified to be operable immediately prior to and daily during REFUELING OPERATIONS.
9. a. The spent fuel pool sweep system, including the charcoal adsorbers, shall be operating during fuel handling and when any load is carried over the pool if irradiated fuel in the pool has decayed less than 30 days. If the spent fuel pool sweep system, including the charcoal adsorber, is not operating when required, fuel movement shall not be started (any fuel assembly movement in progress may be completed).
- b. Performance Requirements
 1. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal.
 2. The results of laboratory carbon sample analysis from spent fuel pool sweep system carbon shall show $\geq 95\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C and 95% RH.
 3. Fans shall operate within $\pm 10\%$ of design flow when tested.
10. The minimum water level above the vessel flange shall be maintained at 23 feet.
11. A dead-load test shall be successfully performed on both the fuel handling and manipulator cranes before fuel movement begins. The load assumed by the cranes for this test must be equal to or greater than the maximum load to be assumed by the cranes during the REFUELING OPERATIONS. A thorough visual inspection of the cranes shall be made after the dead-load test and prior to fuel handling.
12. A licensed senior reactor operator will be on-site and designated in charge of the REFUELING OPERATIONS.
- b. If any of the specified limiting conditions for REFUELING OPERATIONS are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be performed.

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 SHUTDOWN MARGIN shall be at least that as specified in the COLR

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 shall be as specified in the COLR.
- B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If, for any measured hot channel factor, the relationships of $F_Q^N(Z)$ and $F_{\Delta H}^N$ specified in the COLR are not true, then 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, then the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

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

4. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

5. Power distribution maps using the movable detector system shall be made to confirm the relationship of $F_Q^{EQ}(Z)$ specified in the COLR 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, then either of the following actions shall be taken:
 - i. $F_Q^{EQ}(Z)$ shall be increased by the penalty factor specified in the COLR for comparison to the relationship of $F_Q^{EQ}(Z)$ specified in the COLR, or
 - ii. $F_Q^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every seven 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 $F_Q^{EQ}(Z)$ specified in the COLR are not satisfied and the relationships of $F_Q^N(Z)$ and $F_{\Delta H}^N$ specified in the COLR are satisfied, then 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 of $F_Q^{EQ}(Z)$ specified in the COLR 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 of $F_Q^{EQ}(Z)$ specified in the COLR 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 $F_Q^{EQ}(Z)$ specified in the COLR 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 the target band about the target flux difference as specified in the COLR.

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 the target band, specified in the COLR, for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed an envelope specified in the COLR. If the cumulative time exceeds one 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 specified in the COLR, 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 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 ≤ 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, then 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 two 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, then 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, then 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, then 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 withdrawn to within the limits, specified in the COLR, when the reactor is critical or approaching criticality.
2. The control banks shall be limited in physical insertion; insertion limits are specified in the COLR. If any one of the control bank insertion limits is not met:
 - A. Within one hour, initiate boration to restore control bank insertion to within the limits specified in the COLR, and
 - B. Restore control bank insertion to within the limits specified in the COLR within two 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 one 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, as specified in the COLR, 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 $\geq 85\%$ 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 $\geq 85\%$, then the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If peaking factors are not determined within four hours, the reactor power shall be reduced to $< 85\%$ of rating.
2. When reactor power is $< 85\%$ but $\geq 50\%$ 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 $\geq 50\%$, the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If the peaking factors are not determined within four 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 eight 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 eight 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 eight hours after a load change > 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

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

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant flow rate shall be $\geq 93,000$ gallons per minute average per loop and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.
2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. DNBR Parameters

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

**TS FIGURE 3.10-1
DELETED**

**TS FIGURE 3.10-2
DELETED**

**TS FIGURE 3.10-3
DELETED**

TS FIGURE TS 3.10-4

DELETED

TS FIGURE TS 3.10-5

DELETED

**TS FIGURE 3.10-6
DELETED**

3. Monthly OPERATING Report

Routine reports of OPERATING statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

4. Core Operating Limits Report (COLR)

A. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

- | | | |
|------|-----------------|---|
| (1) | TS 2.1 | Reactor Core Safety Limit |
| (2) | TS 2.3.a.3.A | Overtemperature ΔT Setpoint |
| (3) | TS 2.3.a.3.B | Overpower ΔT Setpoint |
| (4) | TS 3.1.f.3 | Moderator Temperature Coefficient (MTC) |
| (5) | TS 3.1.f.4 | Moderator Temperature Coefficient (MTC) |
| (6) | TS 3.8.a.5 | Refueling Boron Concentration |
| (7) | TS 3.10.a | Shutdown Margin |
| (8) | TS 3.10.b.1.A | $F_{\alpha}^N(Z)$ Limits |
| (9) | TS 3.10.b.1.B | $F_{\Delta H}^N$ Limits |
| (10) | TS 3.10.b.4 | $F_{\alpha}^{EQ}(Z)$ Limits |
| (11) | TS 3.10.b.5.C.i | $F_{\alpha}^{EQ}(Z)$ penalty |
| (12) | TS 3.10.b.9 | Axial Flux Difference Target Band |
| (13) | TS 3.10.b.11.A | Axial Flux Difference Envelope |
| (14) | TS 3.10.d.1 | Shutdown Bank Insertion Limits |
| (15) | TS 3.10.d.2 | Control Bank Insertion Limits |
| (16) | TS 3.10.k | Core Average Temperature |
| (17) | TS 3.10.l | Reactor Coolant System Pressure |
| (18) | TS 3.10.m.1 | Reactor Coolant Flow |

B. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

- (1) SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978
- (2) KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10, 2001.

- (3) Nissley, M.E. et. al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.
 - (4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.
 - (5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.
 - (6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.
 - (7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.
 - (8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.
 - (9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.
- C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

- A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

- A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.
- (1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

ATTACHMENT 4

Letter from Mark E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 185

Strikeout TS Bases Pages:

TS B2.1-1

TS B2.3-1 through TS B2.3-2

TS B3.1-1 through TS B3.1-13

TS B3.8-1 through TS B3.8-2

TS B3.10-1 through TS B3.10-7

BASIS - Safety Limits, Reactor Core (TS 2.1)

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

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

The curves are based on the nuclear hot channel factor limits of ~~TS 3.10.b~~ as specified in the COLR.

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

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

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

BASIS - Limiting Safety System Settings - Protective Instrumentation (TS 2.3)

Nuclear Flux

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.⁽¹⁾

The power range reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.⁽²⁾

Two sustained rate protective trip functions have been incorporated in the Reactor PROTECTION SYSTEM. The positive sustained rate trip provides protection against hypothetical rod ejection accident. The negative sustained rate trip provides protection for the core (low DNBR) in the event two or more rod control cluster assemblies (RCCAs) fall into the core. The circuits are independent and ensure immediate reactor trip independent of the initial OPERATING state of the reactor. These trip functions are the LIMITING SAFETY SYSTEM actions employed in the accident analysis.

Pressurizer

The high and low pressure trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure trip causes a reactor trip in the unlikely event of a loss-of-coolant accident.⁽³⁾ The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error⁽²⁾ and transient level overshoot before the reactor trips.

Reactor Coolant Temperature

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: 1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds), and 2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors⁽²⁾ is always below the core SAFETY LIMITS shown in ~~Figure TS-2.1-4~~ the Core Operating Limits Report. If axial peaks are greater than design, as indicated by differences between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.

⁽¹⁾ USAR Section 14.1.1

⁽²⁾ USAR Section 14.0

⁽³⁾ USAR Section 14.3.1

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.⁽²⁾

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.⁽⁴⁾

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.⁽⁵⁾

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System.⁽⁶⁾

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

⁽⁴⁾ WCAP-8092

⁽⁵⁾ USAR Section 14.1.8

BASIS - Operational Components Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 4-one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in Part 4-one will keep Departure from Nucleate Boiling Ratio (DNBR) well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.⁽¹⁾

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 200°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the Low Temperature Overpressure Protection (LTOP) System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

⁽¹⁾ USAR Section 7.2.2

The requirement for at least one train of residual heat removal when in the REFUELING MODE is to ensure sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel < 140°F. The requirement to have two trains of residual heat removal OPERABLE when there is < 23 feet of water above the reactor vessel flange ensures that a single failure will not result in complete loss-of-heat removal capabilities. With the reactor vessel head removed and at least 23 feet of water above the vessel flange, a large heat sink is available. In the event of a failure of the OPERABLE train, additional time is available to initiate alternate core cooling procedures.

Pressurizer Safety Valves (TS 3.1.a.3)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs. per hour of saturated steam at its setpoint. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, then the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against overpressurization.

Pressure Isolation Valves (TS 3.1.a.4)

The Bbasis for the Pressure Isolation Valves is discussed in the Reactor Safety Study (RSS), WASH-1400, and identifies an intersystem loss-of-coolant accident in a PWR which is a significant contributor to risk from core melt accidents (EVENT V). The design examined in the RSS contained two in-series check valves isolating the high pressure Primary Coolant System from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the EVENT V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.⁽²⁾

PORVs and PORV Block Valves (TS 3.1.a.5)

The pressurizer power-operated relief valves (PORVs) operate as part of the Pressurizer Pressure Control System. They are intended to relieve RCS pressure below the setting of the code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The pressurizer PORVs and associated block valves must be OPERABLE to provide an alternate means of mitigating a design basis steam generator tube rupture. Thus, an inoperable PORV (for reasons other than seat leakage) or block valve is not permitted in the HOT STANDBY and OPERATING MODES for periods of more than 72 hours.

⁽²⁾ Order for Modification of License dated 4/20/81

Pressurizer Heaters (TS 3.1.a.6)

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. Hot functional tests⁽³⁾ have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will ensure the ability to maintain pressurizer pressure. Surveillance tests are performed to ensure heater OPERABILITY.

Reactor Coolant Vent System (TS 3.1.a.7)

The function of the High Point Vent System is to vent noncondensable gases from the high points of the RCS to ensure that core cooling during natural circulation will not be inhibited. The OPERABILITY of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow capacity of one charging pump.⁽⁴⁾

⁽³⁾ Hot functional test (PT-RC-31)

⁽⁴⁾ Letter from E. R. Mathews to S. A. Varga dated 5/21/82

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,⁽⁵⁾ and the calculation methods of Footnote.⁽⁶⁾ The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Footnote.⁽⁷⁾

The method specifies that the allowable total stress intensity factor (K_t) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2 K_{im} + K_{it} \leq K_{IR} \quad (3.1b-1)$$

where

K_{im} is the stress intensity factor caused by membrane (pressure) stress

K_{it} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

⁽⁵⁾ Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

⁽⁶⁾ Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, astm designation E262-86.

⁽⁷⁾ WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998.

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{im} is the stress intensity factor due to membrane (pressure) stress. K_{it} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{it} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{iR} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. The heatup and cooldown limit curves have been developed by combining the most conservative pressure temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. Details of the procedure used to account for these variables are explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data for each of the limiting materials. At any given temperature, the allowable pressure is taken to be the lesser of the values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments including the pressure difference between the gage and beltline weld.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations for each of the limiting materials. Composite limit curves are then constructed for each cooldown rate of interest. Again, adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{iR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan⁽⁸⁾ and Footnote.⁽⁹⁾

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,⁽¹⁰⁾ weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,⁽¹¹⁾ WCAP 9878,⁽¹²⁾ WCAP-12020,⁽¹³⁾ WCAP-14279,⁽¹⁴⁾ and WCAP-14279, Revision 1⁽¹⁰⁾ respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 33⁽¹⁾ effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33⁽¹⁾ effective full-power years of fluence (through the end of OPERATING cycle 33⁽¹⁾). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

Note:

- ⁽¹⁾ Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

⁽⁸⁾ "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

⁽⁹⁾ 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

⁽¹⁰⁾ C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

⁽¹¹⁾ S.E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

⁽¹²⁾ S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

⁽¹³⁾ S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

⁽¹⁴⁾ E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

Pressurizer Limits (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, OPERATING limits are provided to ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection (TS 3.1.b.4)

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N-514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33^[1] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are $\leq 200^{\circ}\text{F}$ and the head is on the reactor vessel. The LTOP system is considered OPERABLE when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of $\pm 3\%$ (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

If one train of RHR suction piping to RHR 33-1 is isolated, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then the system can still be considered OPERABLE if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), then the vent path is considered secured in the open position.

Note

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

Maximum Coolant Activity (TS 3.1.c)

The maximum dose to the thyroid and whole body that an individual may receive following an accident is specified in GDC 19 and 10 CFR 100. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 100 limits.

The Reactor Coolant Specific Activity is limited to $\leq 0.2 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 to ensure the thyroid dose does not exceed the GDC-19 and 10 CFR 100 guidelines. The applicable accidents identified in the USAR⁽¹⁵⁾ are analyzed assuming an RCS activity of $0.2 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed by Figure TS 3.1-3 are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike based on Figure TS 3.1-3. The results obtained from these analyses indicate that the control room and off-site thyroid dose are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 100 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91 \mu\text{Ci}}{E \text{ cc}}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91 \mu\text{Ci}}{E \text{ cc}}$. The results obtained from these analyses indicate the control room and off-site whole body dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 100 limits.

The action of reducing average reactor coolant temperature to $< 500^\circ\text{F}$ prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

⁽¹⁵⁾ USAR Section 14.0

Leakage of Reactor Coolant (TS 3.1.d)⁽¹⁶⁾

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ $\mu\text{Ci/cc}$ (\bar{E} = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, then the yearly whole body dose resulting from this activity at the SITE BOUNDARY, using an annual average $X/Q = 2.0 \times 10^{-6} \text{ sec/m}^3$, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the SITE BOUNDARY would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve -hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

TS 3.1.d.2

Limiting the leakage through any single steam generator to 150 gpd ensures that tube integrity is maintained during a design basis main steam line break or loss-of-coolant accident. Remaining within this leakage rate provides reasonable assurance that no single tube-flaw will sufficiently enlarge to create a steam generator tube rupture as a result of stresses caused by a Loss of Coolant Accident (LOCA) or a main steam line break accident within the time allowed for detection of the accident condition and resulting commencement of plant shutdown. This operational leakage rate is less than the condition assumed in design basis safety analyses and conforms to industry standards established by the Nuclear Energy Institute through its NEI 97-06, "Generic Steam Generator Program Guidelines."

⁽¹⁶⁾ USAR Sections 6.5, 11.2.3, 14.2.4

TS 3.1.d.3

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

TS 3.1.d.4

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

TS 3.1.d.5

If leakage is to the containment, it may be identified by one or more of the following methods:

- A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.
- B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.
- C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.
- D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is ensured under all OPERATING conditions.⁽¹⁷⁾

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank.⁽¹⁸⁾ Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at $\leq 60\%$ RATED POWER. The moderator coefficient will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative.⁽¹⁹⁾

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in ~~TS 3.10~~ the COLR precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.⁽¹⁹⁾

The requirement that the pressurizer is partly voided when the reactor is $< 1\%$ subcritical ensures that the Reactor Coolant System will not be solid when criticality is achieved.

⁽¹⁷⁾ USAR Section 4.2

⁽¹⁸⁾ USAR Section 9.2

⁽¹⁹⁾ USAR Section 3.2.1

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than $> 5.0 \text{ pcm}/^\circ\text{F}$ the value as specified in the COLR has been imposed to prevent any unexpected power excursion during normal operation as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than $5.0 \text{ pcm}/^\circ\text{F}$ the value as specified in the COLR provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special OPERATING precautions will be taken. In addition, the strong negative Doppler coefficient⁽²⁰⁾ and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality less than or equal to $\leq 5.0 \text{ pcm}/^\circ\text{F}$ the value as specified in the COLR will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports OPERATING up to 60% power with a moderator temperature coefficient less than or equal to $\leq 5.0 \text{ pcm}/^\circ\text{F}$ the value as specified in the COLR. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than $-8.0 \text{ pcm}/^\circ\text{F}$ the value as specified in the COLR for at least 95% of a cycle's time at HFP to ensure the limitations associated with and anticipated transient without scram (ATWS) event are not exceeded. NRC approved methods⁽²⁰⁾⁽²¹⁾ will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than $-8.0 \text{ pcm}/^\circ\text{F}$ the value as specified in the COLR, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

⁽²⁰⁾ USAR Section 3.2.1

⁽²⁰⁾ "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

⁽²¹⁾ "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

In the event that the limits of TS 3.1.f.3 as provided in the COLR are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit in TS 3.1.f.3 as provided in the COLR. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits of TS 3.1.f.3 as provided in the COLR are not developed, then the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits of TS 3.1.f as specified in the COLR. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

BASIS – Refueling Operations (TS 3.8)

The equipment and general procedures to be utilized during REFUELING OPERATIONS are discussed in the USAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident occurs during the REFUELING OPERATIONS that would result in a hazard to public health and safety.⁽¹⁾ Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (TS 3.8.a.2) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. ~~A~~The boron concentration 2200 ppm, as required by TS 3.8.a.5, as specified in the COLR is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.⁽²⁾ With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Interlocks are utilized during REFUELING OPERATIONS to ensure safe handling. Only one assembly at a time can be handled. The fuel handling hoist is dead weight tested prior to use to assure proper crane operation. It will not be possible to lift or carry heavy objects over the spent fuel pool when fuel is stored therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred forty-eight hour decay time following plant shutdown is consistent with the spent fuel pool cooling analysis and also bounds the assumption used in the dose calculation for the fuel handling accident. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

The bypass dampers are integral to the filter housing. The test of the bypass leakage around the charcoal adsorbers will include the leakage through these dampers.

⁽¹⁾ USAR Section 9.5.2

⁽²⁾ USAR Section 14.1

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine releases to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 100 for the accidents analyzed.

The spent fuel pool sweep system will be operated for the first month after reactor is shutdown for refueling during fuel handling and crane operations with loads over the pool. The potential consequences of a postulated fuel handling accident without the system are a very small fraction of the guidelines of 10 CFR Part 100 after one month decay of the spent fuel. Heavy loads greater than one fuel assembly are not allowed over the spent fuel.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A fuel handling accident in containment does not cause containment pressurization. One containment door in each personnel air lock can be closed following containment personnel evacuation and the containment ventilation and purge system has the capability to initiate automatic containment ventilation isolation to terminate a release path to the atmosphere.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the REFUELING OPERATIONS during changes in core geometry.

Accident analysis assumes a charcoal adsorber efficiency of 90%.⁽³⁾ To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of 90%, this equates to a methyl iodide penetration of 10%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to 5%. Thus, the acceptance criteria of 95% efficient will be used for the charcoal adsorbers.

Although committing to ASTM D3803-89, it was recognized that ASTM D3803-89 Standard references Military Standards MIL-F-51068D, Filter, Particulate High Efficiency, Fire Resistant, and MIL-F-51079A, Filter, Medium Fire Resistant, High Efficiency. These specifications have been revised and the latest revisions are, MIL-F-51068F and MIL-F-51079D. These revisions have been canceled and superseded by ASME AG-1, Code on Nuclear Air and Gas Treatment. ASME AG-1 is an acceptable substitution. Consequently, other referenced standards can be substituted if the new standard or methodology is shown to provide equivalent or superior performance to those referenced in ASTM D3803-89.

⁽³⁾ USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

BASIS-Control Rod and Power Distribution Limits (TS 3.10)

Shutdown Reactivity (TS 3.10.a)

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

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

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

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

TS 1.0.r, "Shutdown Margin," states the definition of shutdown margin as used in the technical specifications. As a part of this definition is a statement which removes the assumption that the highest reactivity worth rod cluster control assembly (RCCA) is fully withdrawn. This includes the verification that all RCCA's are fully inserted by two independent means. Although not fully independent, this requirement refers to indications which are independent. These independent means include such indicators as the control board individual rod position indicators or the rod position as indicated on the power plant computer system (PPCS) or the condition monitors referenced in TS 3.10.e.

Power Distribution Control (TS 3.10.b)

Criteria

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

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

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

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

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

The $F_Q^N(Z)$ limits of ~~TS 3.10.b.1.A as specified in the COLR~~ are derived from the LOCA analyses. The LOCA analyses are performed for ~~Siemens Power Corporation~~ FRA-ANP heavy fuel and for ~~Siemens Power Corporation~~ FRA-ANP standard fuel.

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

~~In TS 3.10.b.1, $F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for Low-low Power-power Physics physics Teststests).~~

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

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

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

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

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

Rod Bow Effects

No penalty for rod bow effects needs to be included in TS 3.10.b.1 for ~~Siemens Power Corporation~~FRA-ANP fuel.⁽¹⁾

Surveillance

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is $\geq 85\%$, 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~~as specified in the COLR.
3. The control bank insertion limits as specified in the COLR 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.⁽¹⁾

Conformance with TS 3.10.b.9 through TS 3.10.b.12 ensures the F_o^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.

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

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~~ as specified in the COLR are permitted from the indicated reference value.

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 envelope as specified in the COLR. ~~in the range +10% to -10% from the target flux increasing by $\pm 1\%$ 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 $\pm 5\%$ target band as specified in the COLR for as long a period as 4 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 by using the boron system to position the full length control rods to produce the required indicated flux difference.

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

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

If a misaligned rod has caused a tilt ratio > 1.09 , then the core power shall be reduced by 2% of rated value for every 1% of indicated power tilt ratio > 1.0 . If after ~~8-eight~~ hours the rod has not been realigned, then 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, then 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, then the generator may remain connected to the grid.

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

Rod Insertion Limits (TS 3.10.d)

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

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

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

Rod Misalignment Limitations (TS 3.10.e)

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

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

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

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

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

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

Inoperable Rod Position Indicator Channels (TS 3.10.f)

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

Inoperable Rod Limitations (TS 3.10.g)

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

Rod Drop Time (TS 3.10.h)

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

Core Average Temperature (TS 3.10.k)

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

Reactor Coolant System Pressure (TS 3.10.l)

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

Reactor Coolant Flow (TS 3.10.m)

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

DNBR Parameters (TS 3.10.n)

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

ATTACHMENT 5

Letter from Mark E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 185

Affected TS Bases Pages:

TS B2.1-1

TS B2.3-1 through TS B2.3-2

TS B3.1-1 through TS B3.1-13

TS B3.8-1 through TS B3.8-2

TS B3.10-1 through TS B3.10-7

BASIS - Safety Limits-Reactor Core (TS 2.1)

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

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

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

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

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

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

BASIS - Limiting Safety System Settings - Protective Instrumentation (TS 2.3)

Nuclear Flux

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.⁽¹⁾

The power range reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.⁽²⁾

Two sustained rate protective trip functions have been incorporated in the Reactor PROTECTION SYSTEM. The positive sustained rate trip provides protection against hypothetical rod ejection accident. The negative sustained rate trip provides protection for the core (low DNBR) in the event two or more rod control cluster assemblies (RCCAs) fall into the core. The circuits are independent and ensure immediate reactor trip independent of the initial OPERATING state of the reactor. These trip functions are the LIMITING SAFETY SYSTEM actions employed in the accident analysis.

Pressurizer

The high and low pressure trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure trip causes a reactor trip in the unlikely event of a loss-of-coolant accident.⁽³⁾ The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error⁽²⁾ and transient level overshoot before the reactor trips.

Reactor Coolant Temperature

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that: 1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds), and 2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors⁽²⁾ is always below the core SAFETY LIMITS shown in the Core Operating Limits Report. If axial peaks are greater than design, as indicated by differences between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.

⁽¹⁾ USAR Section 14.1.1

⁽²⁾ USAR Section 14.0

⁽³⁾ USAR Section 14.3.1

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.⁽²⁾

The overpower and overtemperature PROTECTION SYSTEM setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.⁽⁴⁾

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.⁽⁵⁾

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting ensures a reactor trip signal will be generated before the low-flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip.

Steam Generators

The low-low steam generator water level reactor trip ensures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System.⁽⁶⁾

Reactor Trip Interlocks

Specified reactor trips are bypassed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional ensures their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal (SIS) also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

⁽⁴⁾ WCAP-8092

⁽⁵⁾ USAR Section 14.1.8

BASIS - Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in part one will keep Departure from Nucleate Boiling Ratio (DNBR) well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.⁽¹⁾

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 200°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the LTOP System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

⁽¹⁾USAR Section 7.2.2

The requirement for at least one train of residual heat removal when in the REFUELING MODE is to ensure sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel < 140°F. The requirement to have two trains of residual heat removal OPERABLE when there is < 23 feet of water above the reactor vessel flange ensures that a single failure will not result in complete loss-of-heat removal capabilities. With the reactor vessel head removed and at least 23 feet of water above the vessel flange, a large heat sink is available. In the event of a failure of the OPERABLE train, additional time is available to initiate alternate core cooling procedures.

Pressurizer Safety Valves (TS 3.1.a.3)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs. per hour of saturated steam at its setpoint. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, then the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against overpressurization.

Pressure Isolation Valves (TS 3.1.a.4)

The basis for the pressure isolation valves is discussed in the Reactor Safety Study (RSS), WASH-1400, and identifies an intersystem loss-of-coolant accident in a PWR which is a significant contributor to risk from core melt accidents (EVENT V). The design examined in the RSS contained two in-series check valves isolating the high pressure Primary Coolant System from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the EVENT V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.⁽²⁾

PORVs and PORV Block Valves (TS 3.1.a.5)

The pressurizer power-operated relief valves (PORVs) operate as part of the Pressurizer Pressure Control System. They are intended to relieve RCS pressure below the setting of the code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The pressurizer PORVs and associated block valves must be OPERABLE to provide an alternate means of mitigating a design basis steam generator tube rupture. Thus, an inoperable PORV (for reasons other than seat leakage) or block valve is not permitted in the HOT STANDBY and OPERATING MODES for periods of more than 72 hours.

⁽²⁾ Order for Modification of License dated 4/20/81

Pressurizer Heaters (TS 3.1.a.6)

Pressurizer heaters are vital elements in the operation of the pressurizer which is necessary to maintain system pressure. Loss of energy to the heaters would result in the inability to maintain system pressure via heat addition to the pressurizer. Hot functional tests⁽³⁾ have indicated that one group of heaters is required to overcome ambient heat losses. Placing heaters necessary to overcome ambient heat losses on emergency power will ensure the ability to maintain pressurizer pressure. Surveillance tests are performed to ensure heater OPERABILITY.

Reactor Coolant Vent System (TS 3.1.a.7)

The function of the High Point Vent System is to vent noncondensable gases from the high points of the RCS to ensure that core cooling during natural circulation will not be inhibited. The OPERABILITY of at least one vent path from both the reactor vessel head and pressurizer steam space ensures the capability exists to perform this function.

The vent path from the reactor vessel head and the vent path from the pressurizer each contain two independently emergency powered, energize to open, valves in parallel and connect to a common header that discharges either to the containment atmosphere or to the pressurizer relief tank. The lines to the containment atmosphere and pressurizer relief tank each contain an independently emergency powered, energize to open, isolation valve. This redundancy provides protection from the failure of a single vent path valve rendering an entire vent path inoperable.

A flow restriction orifice in each vent path limits the flow from an inadvertent actuation of the vent system to less than the flow capacity of one charging pump.⁽⁴⁾

⁽³⁾ Hot functional test (PT-RC-31)

⁽⁴⁾ Letter from E. R. Mathews to S. A. Varga dated 5/21/82

Heatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,⁽⁵⁾ and the calculation methods of Footnote.⁽⁶⁾ The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and are discussed in detail in Footnote.⁽⁷⁾

The method specifies that the allowable total stress intensity factor (K_t) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2 K_{im} + K_{it} \leq K_{IR} \quad (3.1b-1)$$

where

K_{im} is the stress intensity factor caused by membrane (pressure) stress

K_{it} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

⁽⁵⁾ Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

⁽⁶⁾ Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, astm designation E262-86.

⁽⁷⁾ WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998.

From equation (3.1b-1) the variables that affect the heatup and cooldown analysis can be readily identified. K_{im} is the stress intensity factor due to membrane (pressure) stress. K_{it} is the thermal (bending) stress intensity factor and accounts for the linearly varying stress in the vessel wall due to thermal gradients. During heatup K_{it} is negative on the inside and positive on the outer surface of the vessel wall. The signs are reversed for cooldown and, therefore, an ID or an OD one quarter thickness surface flaw is postulated in whichever location is more limiting. K_{iR} is dependent on irradiation and temperature and, therefore, the fluence profile through the reactor vessel wall and the rates of heatup and cooldown are important. The heatup and cooldown limit curves have been developed by combining the most conservative pressure temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. Details of the procedure used to account for these variables are explained in the following text.

Following the generation of pressure-temperature curves for both the steady-state (zero rate of change of temperature) and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data for each of the limiting materials. At any given temperature, the allowable pressure is taken to be the lesser of the values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments including the pressure difference between the gage and beltline weld.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location. The pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations for each of the limiting materials. Composite limit curves are then constructed for each cooldown rate of interest. Again, adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{iR} for finite cooldown rates than for steady-state under certain conditions.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above and limited application to ASME Boiler and Pressure Vessel Code Case N-588 to the circumferential beltline weld. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan⁽⁸⁾ and Footnote.⁽⁹⁾

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,⁽¹⁰⁾ weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,⁽¹¹⁾ WCAP 9878,⁽¹²⁾ WCAP-12020,⁽¹³⁾ WCAP-14279,⁽¹⁴⁾ and WCAP-14279, Revision 1⁽¹⁰⁾ respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 33⁽¹⁾ effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33⁽¹⁾ effective full-power years of fluence (through the end of OPERATING cycle 33⁽¹⁾). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

Note:

- ⁽¹⁾ Although the curves were developed for 33 EFY, they are limited to 28 EFY (corresponding to the end of cycle 28) by WPSO Letter NRC-99-017.

⁽⁸⁾ "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

⁽⁹⁾ 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

⁽¹⁰⁾ C. Kim, et al., "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," WCAP-14279, Revision 1, September 1998.

⁽¹¹⁾ S.E. Yanichko, S. L. Anderson, and K. V. Scott, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 8908, January 1977.

⁽¹²⁾ S.E. Yanichko, et al., "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP 9878, March 1981.

⁽¹³⁾ S.E. Yanichko, et al., "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-12020, November 1988.

⁽¹⁴⁾ E. Terek, et al., "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

Pressurizer Limits (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, OPERATING limits are provided to ensure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

Low Temperature Overpressure Protection (TS 3.1.b.4)

The Low Temperature Overpressure Protection System must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N-514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33^[1] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are $\leq 200^{\circ}\text{F}$ and the head is on the reactor vessel. The LTOP system is considered OPERABLE when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of $\pm 3\%$ (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

If one train of RHR suction piping to RHR 33-1 is isolated, then the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within five days. If the isolated flowpath cannot be restored within five days, then the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional eight hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, then the system can still be considered OPERABLE if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), then the vent path is considered secured in the open position.

Note

[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.

Maximum Coolant Activity (TS 3.1.c)

The maximum dose to the thyroid and whole body that an individual may receive following an accident is specified in GDC 19 and 10 CFR 100. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 100 limits.

The Reactor Coolant Specific Activity is limited to $\leq 0.2 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 to ensure the thyroid dose does not exceed the GDC-19 and 10 CFR 100 guidelines. The applicable accidents identified in the USAR⁽¹⁵⁾ are analyzed assuming an RCS activity of $0.2 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed by Figure TS 3.1-3 are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike based on Figure TS 3.1-3. The results obtained from these analyses indicate that the control room and off-site thyroid dose are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 100 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91 \mu\text{Ci}}{E \text{ cc}}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91 \mu\text{Ci}}{E \text{ cc}}$. The results obtained from these analyses indicate the control room and off-site whole body dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 100 limits.

The action of reducing average reactor coolant temperature to $< 500^\circ\text{F}$ prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

⁽¹⁵⁾ USAR Section 14.0

Leakage of Reactor Coolant (TS 3.1.d)⁽¹⁶⁾

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ $\mu\text{Ci/cc}$ (\bar{E} = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, then the yearly whole body dose resulting from this activity at the SITE BOUNDARY, using an annual average $X/Q = 2.0 \times 10^{-6} \text{ sec/m}^3$, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the SITE BOUNDARY would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

TS 3.1.d.2

Limiting the leakage through any single steam generator to 150 gpd ensures that tube integrity is maintained during a design basis main steam line break or loss-of-coolant accident. Remaining within this leakage rate provides reasonable assurance that no single tube-flaw will sufficiently enlarge to create a steam generator tube rupture as a result of stresses caused by a Loss of Coolant Accident (LOCA) or a main steam line break accident within the time allowed for detection of the accident condition and resulting commencement of plant shutdown. This operational leakage rate is less than the condition assumed in design basis safety analyses and conforms to industry standards established by the Nuclear Energy Institute through its NEI 97-06, "Generic Steam Generator Program Guidelines."

⁽¹⁶⁾ USAR Sections 6.5, 11.2.3, 14.2.4

TS 3.1.d.3

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

TS 3.1.d.4

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

TS 3.1.d.5

If leakage is to the containment, it may be identified by one or more of the following methods:

- A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.
- B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.
- C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.
- D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

If leakage is to another closed system it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is ensured under all OPERATING conditions.⁽¹⁷⁾

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank.⁽¹⁸⁾ Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at $\leq 60\%$ RATED POWER. The moderator coefficient will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative.⁽¹⁹⁾

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in the COLR precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.⁽¹⁹⁾

The requirement that the pressurizer is partly voided when the reactor is $< 1\%$ subcritical ensures that the Reactor Coolant System will not be solid when criticality is achieved.

⁽¹⁷⁾ USAR Section 4.2

⁽¹⁸⁾ USAR Section 9.2

⁽¹⁹⁾ USAR Section 3.2.1

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the value specified in the COLR has been imposed to prevent any unexpected power excursion during normal operation as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than the value specified in the COLR provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special OPERATING precautions will be taken. In addition, the strong negative Doppler coefficient⁽²⁰⁾ and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality less than or equal to the value specified in the COLR will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports OPERATING up to 60% power with a moderator temperature coefficient less than or equal to the value specified in the COLR. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than the value specified in the COLR for at least 95% of a cycle's time at HFP to ensure the limitations associated with and anticipated transient without scram (ATWS) event are not exceeded. NRC approved methods⁽²⁰⁾⁽²¹⁾ will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than the value specified in the COLR, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

⁽²⁰⁾ USAR Section 3.2.1

⁽²⁰⁾ "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

⁽²¹⁾ "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

In the event that the limits as provided in the COLR are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit as provided in the COLR. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits as provided in the COLR are not developed, then the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits as specified in the COLR. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

BASIS – Refueling Operations (TS 3.8)

The equipment and general procedures to be utilized during REFUELING OPERATIONS are discussed in the USAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident occurs during the REFUELING OPERATIONS that would result in a hazard to public health and safety.⁽¹⁾ Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (TS 3.8.a.2) and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

A minimum shutdown margin of greater than or equal to 5% $\Delta k/k$ must be maintained in the core. The boron concentration as specified in the COLR is sufficient to ensure an adequate margin of safety. The specification for REFUELING OPERATIONS shutdown margin is based on a dilution during refueling accident.⁽²⁾ With an initial shutdown margin of 5% $\Delta k/k$, under the postulated accident conditions, it will take longer than 30 minutes for the reactor to go critical. This is ample time for the operator to recognize the audible high count rate signal, and isolate the reactor makeup water system. Periodic checks of refueling water boron concentration ensure that proper shutdown margin is maintained. Specification 3.8.a.6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Interlocks are utilized during REFUELING OPERATIONS to ensure safe handling. Only one assembly at a time can be handled. The fuel handling hoist is dead weight tested prior to use to assure proper crane operation. It will not be possible to lift or carry heavy objects over the spent fuel pool when fuel is stored therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred forty-eight hour decay time following plant shutdown is consistent with the spent fuel pool cooling analysis and also bounds the assumption used in the dose calculation for the fuel handling accident. The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides added assurance that the off-site doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating, fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

The bypass dampers are integral to the filter housing. The test of the bypass leakage around the charcoal adsorbers will include the leakage through these dampers.

⁽¹⁾ USAR Section 9.5.2

⁽²⁾ USAR Section 14.1

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine releases to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 100 for the accidents analyzed.

The spent fuel pool sweep system will be operated for the first month after reactor is shutdown for refueling during fuel handling and crane operations with loads over the pool. The potential consequences of a postulated fuel handling accident without the system are a very small fraction of the guidelines of 10 CFR Part 100 after one month decay of the spent fuel. Heavy loads greater than one fuel assembly are not allowed over the spent fuel.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A fuel handling accident in containment does not cause containment pressurization. One containment door in each personnel air lock can be closed following containment personnel evacuation and the containment ventilation and purge system has the capability to initiate automatic containment ventilation isolation to terminate a release path to the atmosphere.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the REFUELING OPERATIONS during changes in core geometry.

Accident analysis assumes a charcoal adsorber efficiency of 90%.⁽³⁾ To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of 90%, this equates to a methyl iodide penetration of 10%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to 5%. Thus, the acceptance criteria of 95% efficient will be used for the charcoal adsorbers.

Although committing to ASTM D3803-89, it was recognized that ASTM D3803-89 Standard references Military Standards MIL-F-51068D, Filter, Particulate High Efficiency, Fire Resistant, and MIL-F-51079A, Filter, Medium Fire Resistant, High Efficiency. These specifications have been revised and the latest revisions are, MIL-F-51068F and MIL-F-51079D. These revisions have been canceled and superseded by ASME AG-1, Code on Nuclear Air and Gas Treatment. ASME AG-1 is an acceptable substitution. Consequently, other referenced standards can be substituted if the new standard or methodology is shown to provide equivalent or superior performance to those referenced in ASTM D3803-89.

⁽³⁾USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

BASIS-Control Rod and Power Distribution Limits (TS 3.10)

Shutdown Reactivity (TS 3.10.a)

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. To maintain the required trip reactivity, the rod insertion limits as specified in the COLR must be observed. In addition, for HOT SHUTDOWN conditions, the shutdown margin as specified in the COLR must be provided for protection against the steam line break accident.

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

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

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

TS 1.0.r, "Shutdown Margin," states the definition of shutdown margin as used in the technical specifications. As a part of this definition is a statement which removes the assumption that the highest reactivity worth rod cluster control assembly (RCCA) is fully withdrawn. This includes the verification that all RCCA's are fully inserted by two independent means. Although not fully independent, this requirement refers to indications which are independent. These independent means include such indicators as the control board individual rod position indicators or the rod position as indicated on the power plant computer system (PPCS) or the condition monitors referenced in TS 3.10.e.

Power Distribution Control (TS 3.10.b)

Criteria

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

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

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

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

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

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses. The LOCA analyses are performed for FRA-ANP heavy fuel and for FRA-ANP standard fuel.

When a $F_Q^N(Z)$ 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 3% is the appropriate allowance for manufacturing tolerance.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

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

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

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

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

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

Rod Bow Effects

No penalty for rod bow effects needs to be included in TS 3.10.b.1 for FRA-ANP fuel.⁽¹⁾

Surveillance

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is $\geq 85\%$, or an indicated 24 steps when reactor power is $< 85\%$.
2. Control rod banks are sequenced with overlapping banks as specified in the COLR.
3. The control bank insertion limits as specified in the COLR 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.⁽¹⁾

Conformance with TS 3.10.b.9 through TS 3.10.b.12 ensures the F_o^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.

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

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 as specified in the COLR are permitted from the indicated reference value.

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 envelope as specified in the COLR. 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 target band as specified in the COLR 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 by using the boron system to position the full length control rods to produce the required indicated flux difference.

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

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

If a misaligned rod has caused a tilt ratio > 1.09 , then the core power shall be reduced by 2% of rated value for every 1% of indicated power tilt ratio > 1.0 . If after eight hours the rod has not been realigned, then 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, then 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, then the generator may remain connected to the grid.

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

Rod Insertion Limits (TS 3.10.d)

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

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

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

Rod Misalignment Limitations (TS 3.10.e)

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

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

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

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

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

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

Inoperable Rod Position Indicator Channels (TS 3.10.f)

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

Inoperable Rod Limitations (TS 3.10.g)

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

Rod Drop Time (TS 3.10.h)

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

Core Average Temperature (TS 3.10.k)

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

Reactor Coolant System Pressure (TS 3.10.l)

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

Reactor Coolant Flow (TS 3.10.m)

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

DNBR Parameters (TS 3.10.n)

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

ATTACHMENT 6

Letter from Mark E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 185

Kewaunee Nuclear Power Plant

Core Operating Limits Report (COLR)

TRM 2.1

Kewaunee Nuclear Power Plant

**CORE OPERATING LIMITS REPORT
(COLR)**

CYCLE 25

REVISION 0

CORE OPERATING LIMITS REPORT CYCLE 25

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Kewaunee Nuclear Power Plant (KNPP) has been prepared in accordance with the requirements of Technical Specification (TS) 6.9.4.

A cross-reference between the COLR sections and the KNPP Technical Specifications affected by this report is given below:

COLR Section	KNPP TS	Description
2.1	2.1	Reactor Core Safety Limits
2.2	3.10.a	Shutdown Margin
2.3	3.1.f.3 3.1.f.4	Moderator Temperature Coefficient
2.4	3.10.d.1	Shutdown Bank Insertion Limit
2.5	3.10.d.2	Control Bank Insertion Limits
2.6	3.10.b.1.A 3.10.b.4 3.10.b.5.C.i	Heat Flux Hot Channel Factor ($F_Q(Z)$)
2.7	3.10.b.1.B	Nuclear Enthalpy Rise Hot Channel Factor (F^{NH})
2.8	3.10.b.9 3.10.b.11.A	Axial Flux Difference (AFD)
2.9	2.3.a.3.A	Overtemperature ΔT Setpoint
2.10	2.3.a.3.B	Overpower ΔT Setpoint
2.11	3.10.k 3.10.l 3.10.m.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
2.12	3.8.a.5	Refueling Boron Concentration
Figure 1		Safety Limits-Reactor Core
Figure 2		Required Shutdown Margin
Figure 3		K(Z) Normalized Operating Envelope
Figure 4		Control Bank Insertion Limits
Figure 5		V(Z) as a Function of Core Height

CORE OPERATING LIMITS REPORT CYCLE 25

2.0 Operating Limits

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC approved methodologies specified in Technical Specification 6.9.a.4

2.1 Reactor Core Safety Limits

The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits shown in COLR Figure 1. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line

2.2 Shutdown Margin

2.2.1 When the reactor is subcritical prior to reactor startup, the SHUTDOWN margin shall be at least that shown in COLR Figure 2.

2.3 Moderator Temperature Coefficient

2.3.1 When the reactor is critical and $\leq 60\%$ RATED POWER, the moderator temperature coefficient shall be ≤ 5.0 pcm/°F, except during LOW POWER PHYSICS TESTING. When the reactor is $> 60\%$ RATED POWER, the moderator temperature coefficient shall be zero or negative.

2.3.2 The reactor will have a moderator temperature coefficient no less negative than -8 pcm/°F for 95% of the cycle time at full power.

2.4 Shutdown Bank Insertion Limit

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

2.5 Control Bank Insertion Limit

2.5.1 The control banks shall be limited in physical insertion; insertion limits are shown in COLR Figure 4

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2.6 Nuclear Heat Flux Hot Channel Factor ($F_Q^N(Z)$)

2.6.1 $F_Q^N(Z)$ Limits for FRA-ANP Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.35)/P \times K(Z) \text{ for } P > 0.5 \quad \text{[Hvy]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.70) \times K(Z) \text{ for } P \leq 0.5 \quad \text{[Hvy]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z) \text{ for } P > 0.5 \quad \text{[Std]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z) \text{ for } P \leq 0.5 \quad \text{[Std]}$$

where:

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

K(Z) is the function given in Figure 3

Z is the core height location for the F_Q of interest

2.6.2 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 FRA-ANP fuel:

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

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

where:

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

V(Z) is defined in COLR Figure 5

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

2.6.3 The penalty factor for TS 3.10.b.5.C.i shall be 2%.

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

2.7.1 $F_{\Delta H}^N$ Limits for FRA-ANP Fuel

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

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

where:

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

2.8 Axial Flux Difference (AFD)

2.8.1 The indicated axial flux target band shall be the area maintained within $\pm 5\%$ about the target flux difference.

2.8.2 The envelope is an area whose outer limits are described 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%.

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2.9 Overtemperature ΔT Setpoint

Overtemperature ΔT setpoint parameter values:

- ΔT_0 = Indicated ΔT at RATED POWER, %
T = Average temperature, °F
T' = 567.3 °F
P = Pressurizer Pressure, psig
P' = 2235 psig
K₁ = 1.11
K₂ = 0.0090
K₃ = 0.000566
 τ_1 = 30 seconds
 τ_2 = 4 seconds
f(ΔI) = An even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that
- (a) For $q_t - q_b$ within -12, +9 %, f(ΔI) = 0
 - (b) For each percent that the magnitude of $q_t - q_b$ exceeds +9 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.5 % of RATED POWER.
 - (c) For each percent that the magnitude of $q_t - q_b$ exceed -12 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5 % of RATED POWER.

2.10 Overpower ΔT Setpoint

Overpower ΔT setpoint parameter values:

- ΔT_0 = Indicated ΔT at RATED POWER, %
T = Average temperature, °F
T' = 567.3 °F
K₄ = 1.10
K₅ = 0.0275 for increasing T; 0 for decreasing T
K₆ = 0.002 for T > T' ; 0 for T < T'
 τ_3 = 10 seconds
f(ΔI) = Same as in 2.9

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- 2.11 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
 - 2.11.1 During steady state power operation, T_{avg} shall be $< 568.8^{\circ}\text{F}$.
 - 2.11.2 During steady state power operation, Pressurizer Pressure shall be ≥ 2205 psig
 - 2.11.3 During steady state power operation, reactor coolant flow rate shall be $\geq 93,000$ gpm per loop.
- 2.12 Refueling Boron Concentration
 - 2.12.1 When there is fuel in the reactor, a minimum boron concentration of 2200 ppm shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor.

CORE OPERATING LIMITS REPORT CYCLE 25

Figure 1
Safety Limits-Reactor Core

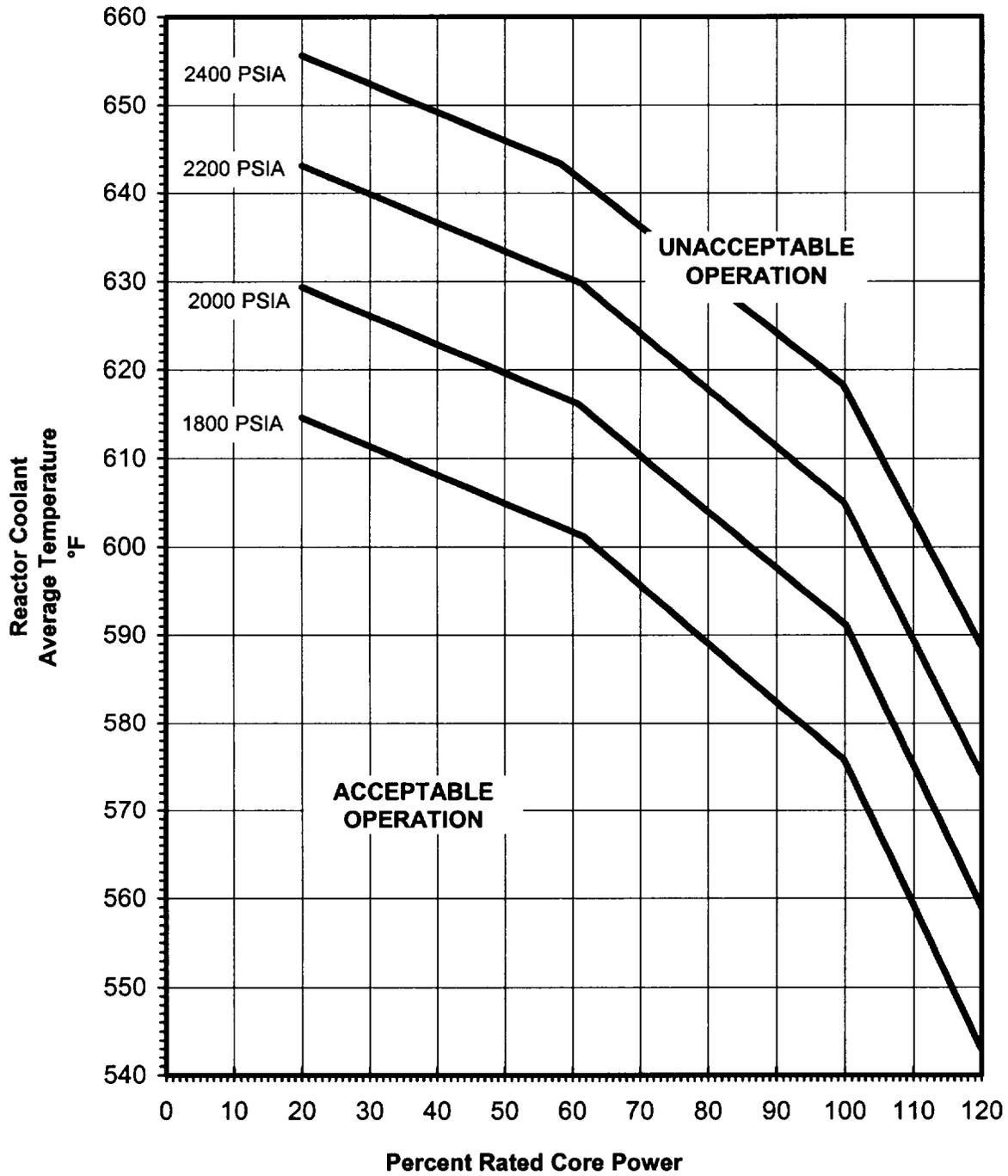
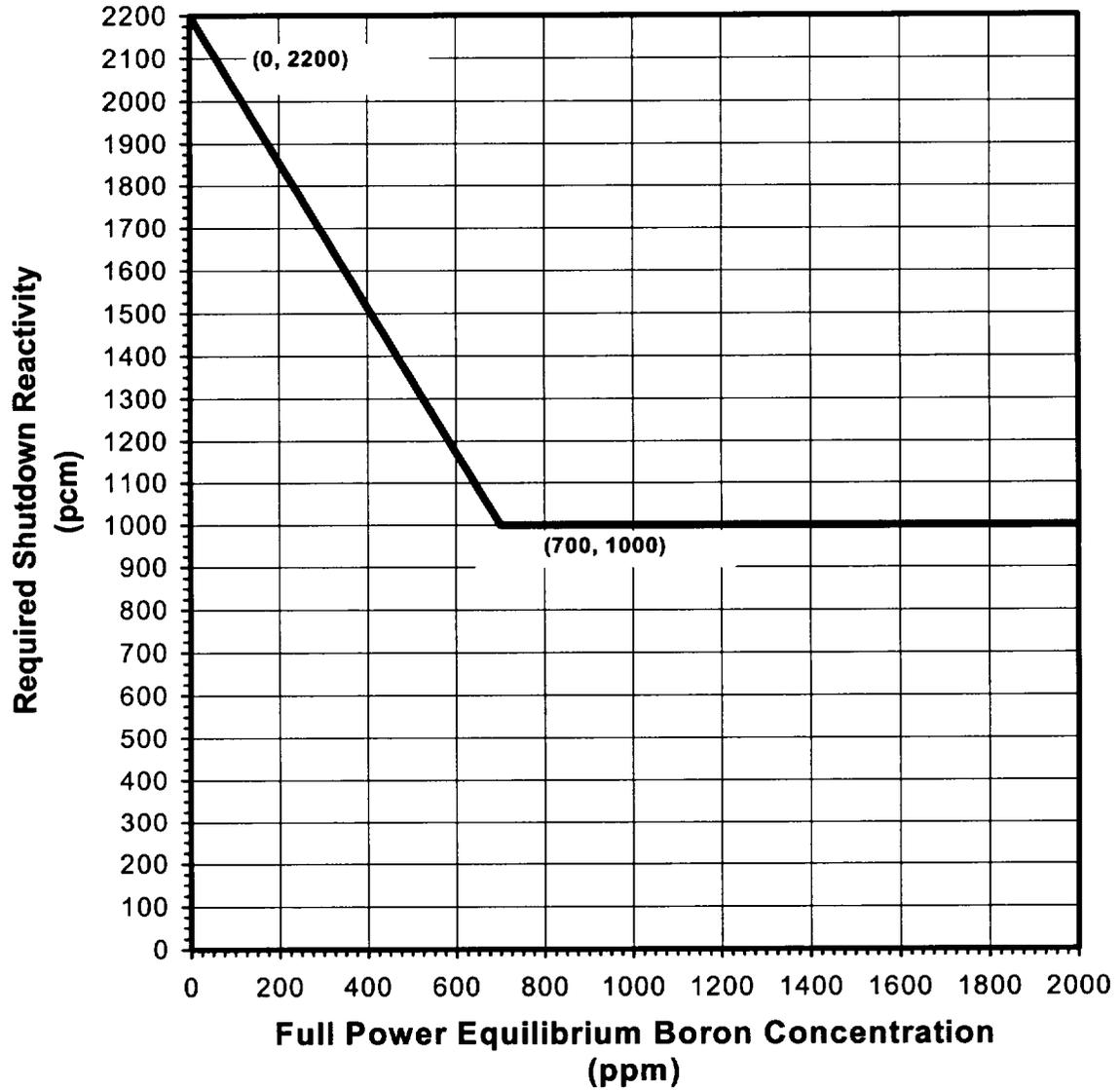
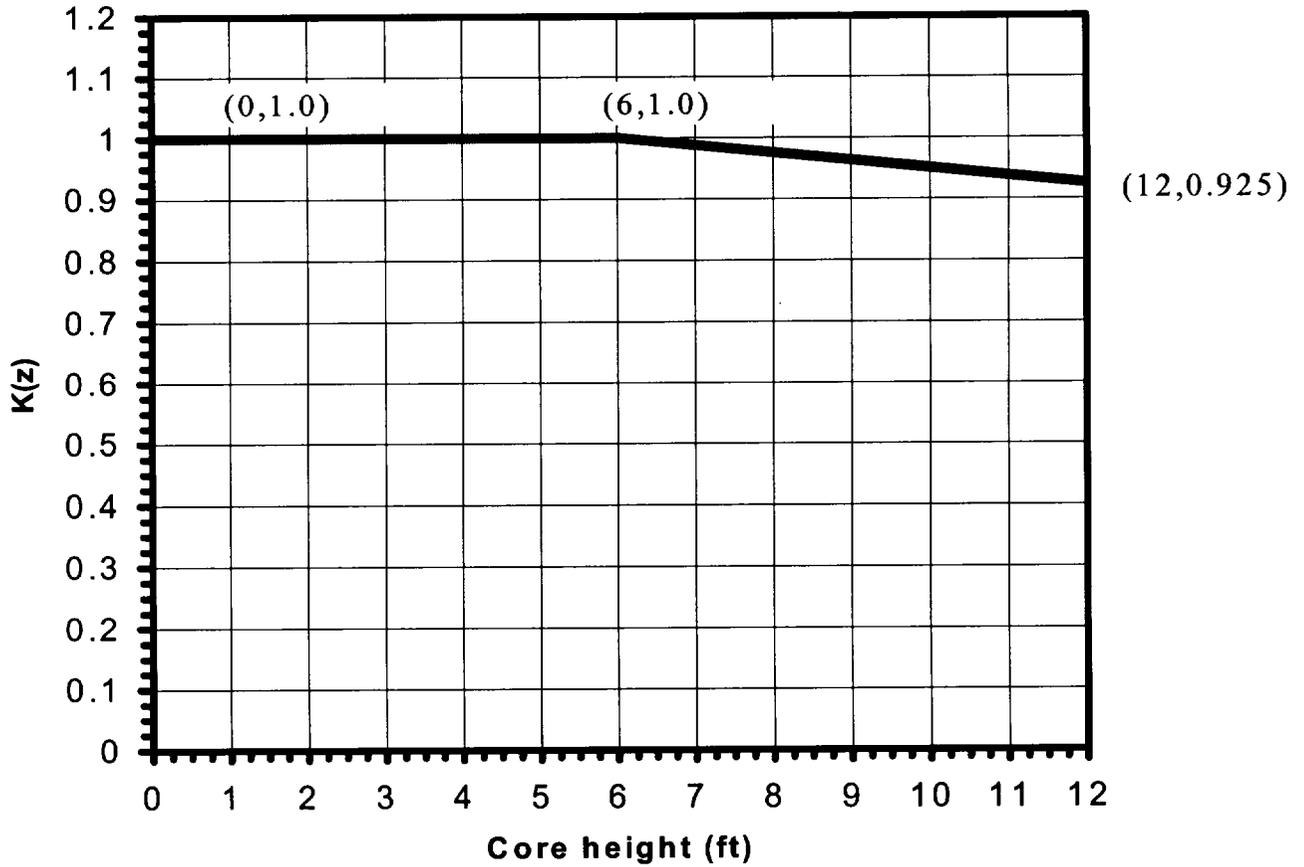


Figure 2
Required Shutdown Reactivity vs Boron Concentration



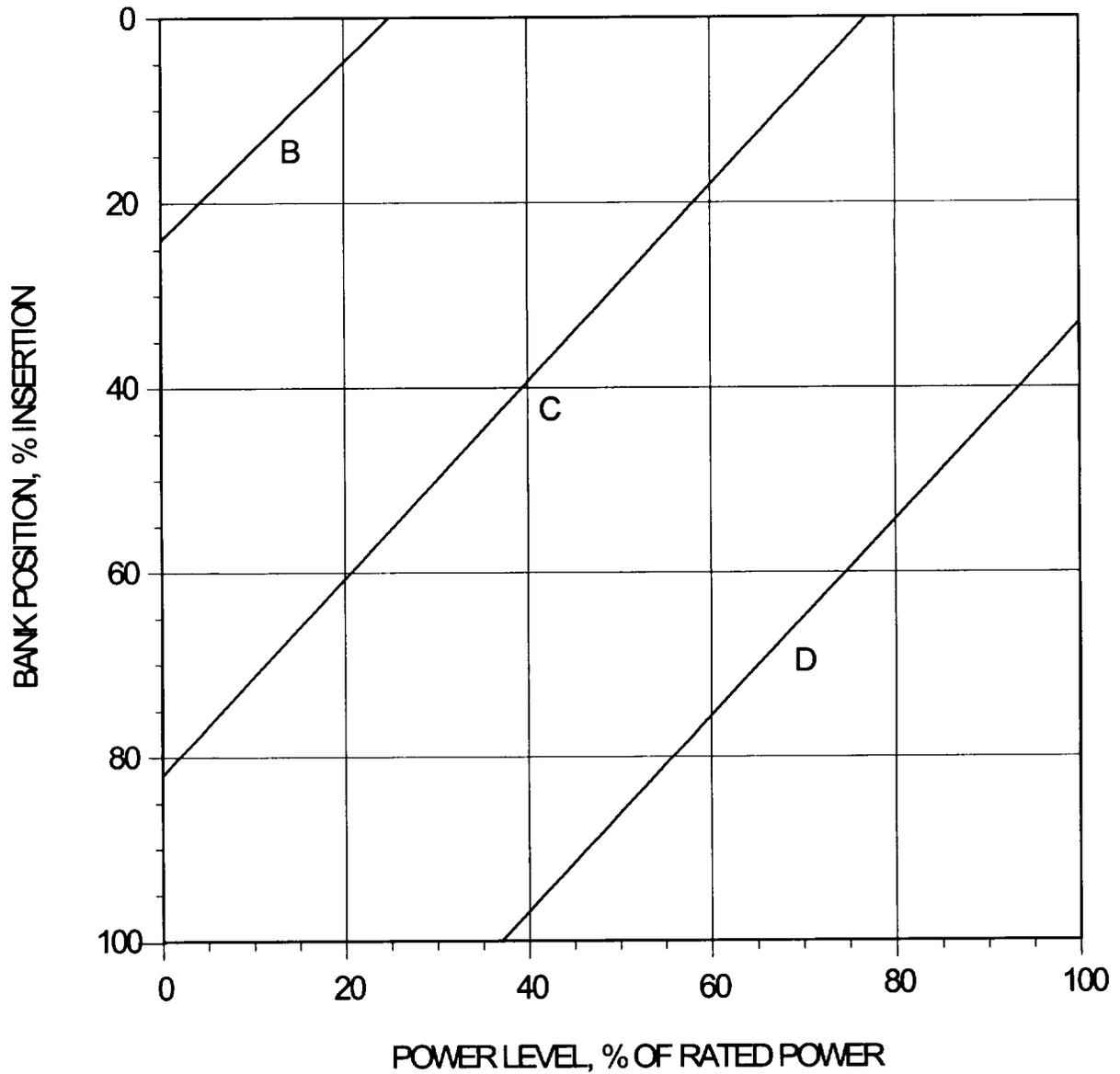
CORE OPERATING LIMITS REPORT CYCLE 25

Figure 3
Hot Channel Factor Normalized Operating Envelope



CORE OPERATING LIMITS REPORT CYCLE 25

Figure 4
Control Bank Insertion Limits



CORE OPERATING LIMITS REPORT CYCLE 25

Figure 5
V(Z) as a Function of Core Height

