SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION

BASES

SECTION

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in ~~Figure 2.1-1.~~ the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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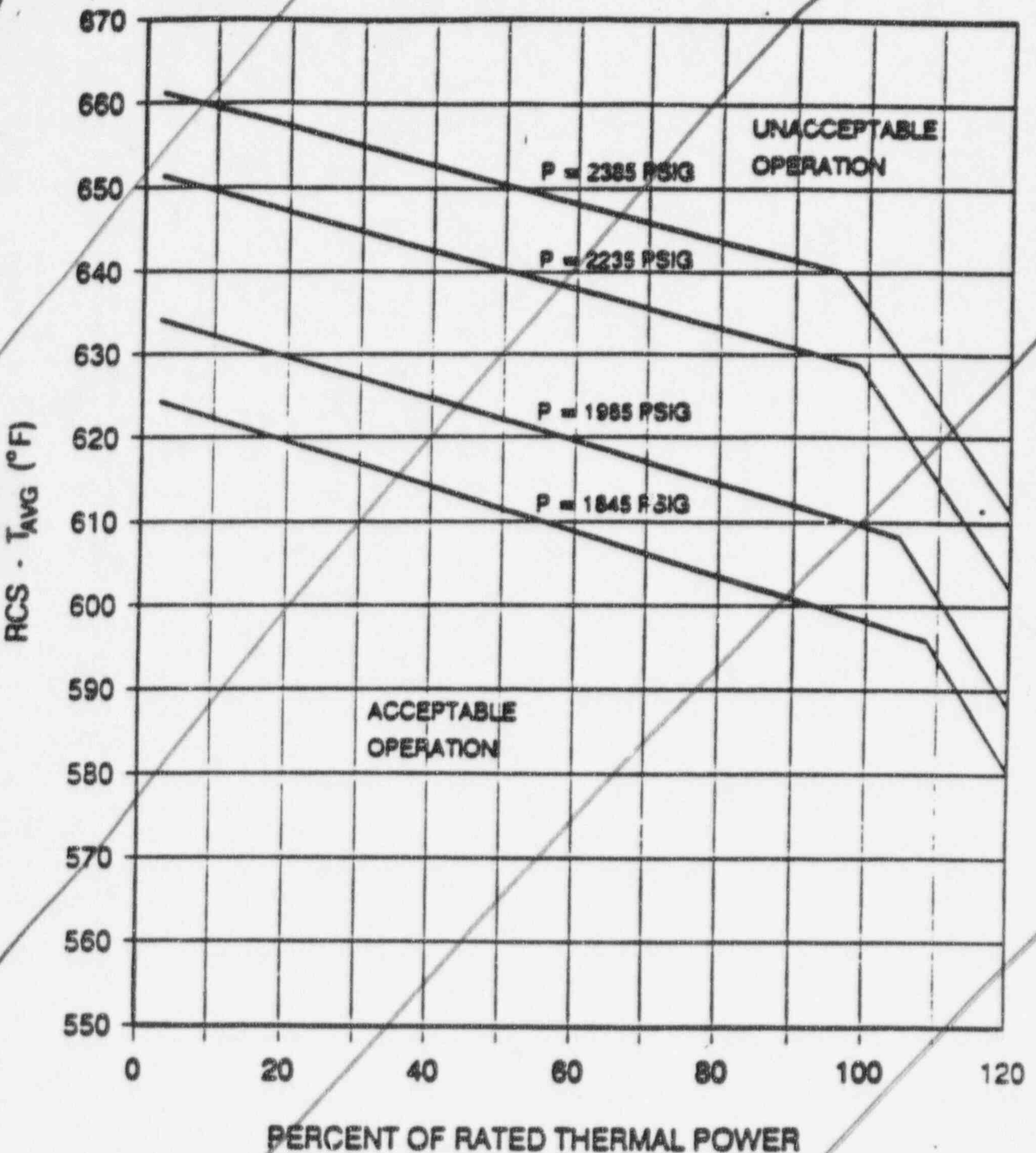

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FIGURE 2.1-1a

UNIT 1 REACTOR CORE SAFETY LIMITS

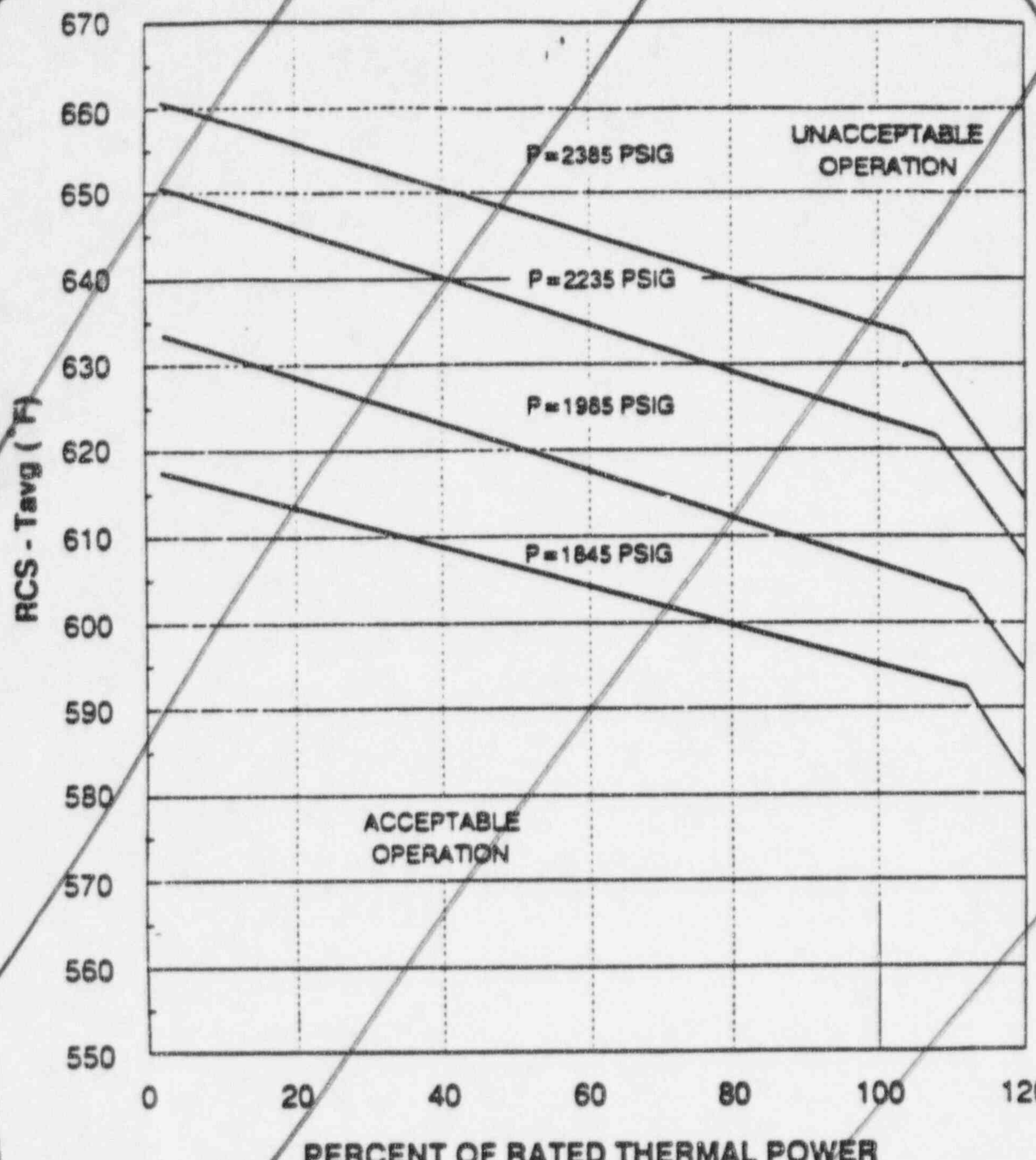

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FIGURE 2.1-1b

UNIT 2 REACTOR CORE SAFETY LIMITS

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local heat flux 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 DNB design basis is that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence level that DNB will not occur when the minimum DNBR is at the DNBR limit. In meeting this design basis, uncertainties in plant operating parameters are considered such that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves ~~of Figure 2.1-1~~ show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis limit value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

Reactor Core Safety Limits

LIMITING SAFETY SYSTEM SETTINGS

BASES

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. In addition, the Source Range Neutron Flux trip provides similar protection during shutdown operations with the reactor trip breakers closed and the rod control system capable of control rod withdrawal. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature N-16

The Overtemperature N-16 trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the N-16 detectors, and pressure is within the range between the Pressurizer High and Low Pressure trips. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the cold leg temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the ~~core Safety Limit as shown in Figure 2.1-1~~ Reactor Core Safety Limits curves. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced ~~according to the notations in Table 2.2-1~~.

Overpower N-16

The Overpower N-16 trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature trip, and provides a backup to the High Neutron Flux trip. The Overpower N-16 trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

MONTHLY OPERATING REPORTS (Continued)

shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.6a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1). Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2). Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

3). Control Rod Insertion Limits for Specification 3/4.1.3.6,

4). AXIAL FLUX DIFFERENCE Limits and target band for Specification 3/4.2.1.,

5). Heat Flux Hot Channel Factor, K(Z), W(Z), F_Q^{RTP}, and the $F_Q^C(Z)$ allowances for Specification 3/4.2.2,

6). Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

7). Reactor Core Safety Limits for Specification 2.1.1.

6.9.1.6b The following analytical methods used to determine the core operating limits are for Units 1 and 2, unless otherwise stated, and shall be those previously approved by the NRC in:

1). WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 2.1- Reactor Core Safety Limits.)

2). WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary). (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

3). T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

4). NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

CORE OPERATING LIMITS REPORT (Continued)

5). WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor (W(z) surveillance requirements for F_Q Methodology).)

6). WCAP-10079-P-A, "NOTRUMP, A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985, (W Proprietary).

7). WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE", August 1985, (W Proprietary).

8). WCAP-11145-P-A, "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL GENERIC STUDY WITH THE NOTRUMP CODE", October 1986, (W Proprietary).

9). RXE-90-006-P, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," February 1991. (Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor.)

, and 2.1 - Reactor Core Safety Limits

10). RXE-88-102-P, "TUE-1 Departure from Nucleate Boiling Correlation", January 1989.

11). RXE-88-102-P, [illegible], "TUE-1 DNB Correlation - Supplement 1", December 1990.

12). RXE-89-002, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", June 1989.

13). RXE-91-001, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", February 1991.

14). RXE-91-002, "Reactivity Anomaly Events Methodology", May 1991. (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

15). RXE-90-007, "Large Break Loss of Coolant Accident Analysis Methodology", December 1990.

16). TXX-88306, "Steam Generator Tube Rupture Analysis", March 15, 1988.

17). RXE-91-005, "Methodology for Reactor Core Response to Steamline Break Events," May, 1991.

ENCLOSURE 1 TO TXX-95213

GENERIC LETTER 88-16 - REMOVAL OF CYCLE-SPECIFIC PARAMETER

LIMITS FROM TECHNICAL SPECIFICATIONS

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

OCT 0 4 1988

TO ALL POWER REACTOR LICENSEES AND APPLICANTS

SUBJECT: REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS FROM TECHNICAL SPECIFICATIONS (GENERIC LETTER 88-16)

License amendments are generally required each fuel cycle to update the values of cycle-specific parameter limits in Technical Specifications (TS). The processing of changes to TS that are developed using an NRC-approved methodology is an unnecessary burden on licensee and NRC resources. A lead-plant proposal for an alternative that eliminates the need for a license amendment to update the cycle-specific parameter limits each fuel cycle was submitted for the Oconee plant with the endorsement of the Babcock and Wilcox Owners Group. On the basis of the NRC review and approval of that proposal, the enclosed guidance for the preparation of a license amendment request for this alternative was developed by the NRC staff.

Generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal. As a consequence, the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. These changes also allow the NRC staff to trend the values of these limits relative to past experience. This alternative allows continued trending of these limits without the necessity of prior NRC review and approval.

Licensees and applicants are encouraged to propose changes to TS that are consistent with the guidance provided in the enclosure. Conforming amendments will be expeditiously reviewed by the NRC Project Manager for the facility. Proposed amendments that deviate from this guidance will require a longer, more detailed review. Please contact the Project Manager if you have questions on this matter.

Sincerely,

8810050058

Dennis M. Crutchfield

Dennis M. Crutchfield
Acting Associate Director for Projects
Office of Nuclear Reactor Regulation

RECEIVED

OCT 21 1988

WILLIAM G. COUNSIL

Enclosure:
As stated

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GUIDANCE FOR TECHNICAL SPECIFICATION CHANGES FOR CYCLE-SPECIFIC PARAMETER LIMITS

INTRODUCTION

A number of Technical Specifications (TS) address limits associated with reactor physics parameters that generally change with each reload core, requiring the processing of changes to TS to update these limits each fuel cycle. If these limits are developed using an NRC-approved methodology, the license amendment process is an unnecessary burden on the licensee and the NRC. An alternative to including the values of these cycle-specific parameters in individual specifications is provided and is responsive to industry and NRC efforts on improvements in TS.

This enclosure provides guidance for the preparation of a license amendment request to modify TS that have cycle-specific parameter limits. An acceptable alternative to specifying the values of cycle-specific parameter limits in TS was developed on the basis of the review and approval of a lead-plant proposal for this change to the TS for the Oconee units. The implementation of this alternative will result in a resource savings for the licensees and the NRC by eliminating the majority of license amendment requests on changes in values of cycle-specific parameters in TS.

DISCUSSION

This alternative consists of three separate actions to modify the plant's TS: (1) 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, (2) the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information, and (3) the modification of individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report.

In the evaluation of this alternative, the NRC staff concluded that it is essential to safety that the plant is operated within the bounds of cycle-specific parameter limits and that a requirement to maintain the plant within the appropriate bounds must be retained in the TS. However, the specific values of these limits may be modified by licensees, without affecting nuclear safety, provided that these changes are determined using an NRC-approved methodology and consistent with all applicable limits of the plant safety analysis that are addressed in the Final Safety Analysis Report (FSAR). Additionally, it was concluded that a formal report should be submitted to NRC with the values of these limits. This will allow continued trending of this information, even though prior NRC approval of the changes to these limits would not be required.

The current method of controlling reactor physics parameters to assure conformance to 10 CFR 50.36 is to specify the specific value(s) determined to be within specified acceptance criteria (usually the limits of the safety analyses) using an approved calculation methodology. The alternative contained in this guidance controls the values of cycle-specific parameters and assures conformance to 10 CFR 50.36, which calls for specifying the lowest functional

performance levels acceptable for continued safe operation, by specifying the calculation methodology and acceptance criteria. This permits operation at any specific value determined by the licensee, using the specified methodology, to be within the acceptance criteria. The Core Operating Limits Report will document the specific values of parameter limits resulting from licensee's calculations including any mid-cycle revisions to such parameter values.

The following items show the changes to the TS for this alternative. A defined formal report, "Core Operating Limits Report" (the name used as an example for the title for this report), shall be added to the Definitions section of the TS, as follows.

> <u>[CORE] OPERATING LIMITS REPORT</u>
> 1.XX The [CORE] OPERATING LIMITS REPORT is the unit-specific document that provides [core] operating limits for the current operating reload cycle. These cycle-specific [core] operating limits shall be determined for each reload cycle in accordance with Specification 6.9.X. Plant operation within these operating limits is addressed in individual specifications.

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

> <u>[CORE] OPERATING LIMITS REPORT</u>
> [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 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 Desk with copies to the Regional Administrator and Resident Inspector.

Individual specifications shall be revised to state that the values of cycle-specific parameters shall be maintained within the limits identified in the defined formal report. Typical modifications for individual specifications are as follows.

> The regulating rods shall be positioned <u>within the acceptable operating range for regulating rod position provided in the [CORE] OPERATING LIMITS REPORT</u>. (Used where the operating limit covers a range of acceptable operation, typically defined by a curve.)

> The [cycle-specific parameter] shall be <u>within the limit provided in the [CORE] OPERATING LIMITS REPORT.</u> (Used where the operating limit has a single point value.)

SUMMARY

The alternative to including the values of cycle-specific parameter limits in individual specifications includes (1) the addition of a new defined term for the formal report that provides the cycle-specific parameter limits, (2) the addition of its associated reporting requirement to the Administrative Controls section of the TS, and (3) the modification of individual specifications to replace these limits with a reference to the defined formal report for the values of these limits. With this alternative, reload license amendments for the sole purpose of updating cycle-specific parameter limits will be unnecessary.

Enclosure

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
88-15	ELECTRIC POWER SYSTEMS - INADEQUATE CONTROL OVER DESIGN PROCESSES	09/12/88	ALL POWER REACTOR LICENSEES AND APPLICANTS
88-14	INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT	08/08/88	ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS
88-13	OPERATOR LICENSING EXAMINATIONS	08/08/88	ALL POWER REACTOR LICENSEES AND APPLICANTS FOR AN OPERATING LICENSE.
88-12	REMOVAL OF FIRE PROTECTION REQUIREMENTS FROM TECHNICAL SPECIFICATIONS	08/02/88	ALL POWER REACTOR LICENSEES AND APPLICANTS
88-11	NRC POSITION ON RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS	07/12/88	ALL LICENSEES OF OPERATING REACTORS AND HOLDERS OF CONSTRUCTION PERMITS
88-10	PURCHASE OF GSA APPROVED SECURITY CONTAINERS	07/01/88	ALL POWER REACTOR LICENSEES AND HOLDERS OF PART 95 APPROVALS
88-09	PILOT TESTING OF FUNDAMENTALS EXAMINATION	05/17/88	ALL LICENSEES OF ALL BOILING WATER REACTORS AND APPLICANTS FOR A BOILING WATER REACTOR OPERATOR'S LICENSE UNDER 10 CFR PART 55
88-08	MAIL SENT OR DELIVERED TO THE OFFICE OF NUCLEAR REACTOR REGULATION	05/03/88	ALL LICENSEES FOR POWER AND NON-POWER REACTORS AND HOLDERS OF CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS
88-07	MODIFIED ENFORCEMENT POLICY RELATING TO 10 CFR 50.49, "ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS"	04/07/88	ALL POWER REACTOR LICENSEES AND APPLICANTS