SUMMARY OF CHANGES
APPLICATION OF SELECTION CRITERIA TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

Change Description	Affected Pages
No changes are required for this Volume.	N/A

VOLUME 1

APPLICATION OF SELECTION CRITERIA TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

Revision 1

APPLICATION OF SELECTION CRITERIA TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

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APPENDIX

APPLICATION OF SELECTION CRITERIA TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The purpose of this document is to confirm the results of the Westinghouse Owners Group application of the Technical Specification selection criteria on a plant specific basis for the Donald C. Cook Nuclear Plant (CNP) Units 1 and 2. Indiana Michigan Power Company (I&M) has reviewed the application and confirmed the applicability of the selection criteria to each of the Technical Specifications utilized in report WCAP-11618, "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program - Phase II Task 5, Criteria Application" (Reference 1) including Addendum 1, NRC Staff Review of NSSS Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988 and as revised in NUREG-1431, Revision 2 "Standard Technical Specifications, Westinghouse Plants" (Reference 2) and applied the criteria to each of the current CNP Units 1 and 2 Technical Specifications. Additionally, in accordance with the NRC Final Policy Statement (Reference 3), this confirmation of the application of selection criteria includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to the CNP Units 1 and 2.

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2. SELECTION CRITERIA

I&M has utilized the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 (Reference 3) to develop the results contained in the attached matrix. PRA insights as used in the Westinghouse Owners Group submittal were utilized, confirmed by I&M, and are discussed in the next section of this report. The selection criteria and discussion provided in Reference 3 are as follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing design basis accident and transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the Final Safety Analysis Report (FSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the design basis accident or transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room.

These could also include other features or characteristics that are specifically assumed in Design Basis Accident and Transient analyses even if they cannot be directly observed in the control room (e.g, moderator temperature coefficient and hot channel factors).

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2. SELECTION CRITERIA (continued)

The purpose of this criterion is to capture those process variables that have initial values assumed in the design basis accident and transient analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated design basis accident or transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the design basis accident or transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's design basis accident and transient analyses, as presented in Chapters 6 and 15 of the plant's Final Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown).

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety:

Discussion of Criterion 4: It is the Commission policy that licensees retain in their Technical Specifications LCOs, action statements and Surveillance Requirements for the following systems (as applicable), which operating experience and PSA have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

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2. SELECTION CRITERIA (continued)

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant and design-specific PSA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PSA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant specific PSA or risk survey and any available literature on risk insights and PSAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

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APPLICATION OF SELECTION CRITERIA
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3. PRA INSIGHTS

Introduction and Objectives

Reference 3 includes a statement that NRC expects licensees to utilize any plant specific PSA or risk survey and any available literature on risk insights and PSAs to strengthen the technical bases for these requirements that remain in Technical Specifications and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed as being relocated to other plant controlled documents will be maintained under programs subject to the 10 CFR 50.59 review process. These Relocated Specifications have been compared to a variety of PRA material with two purposes: 1) to identify if a Specification component or topic is addressed by PRA; and 2) if addressed, to judge if the Relocated Specification component or topic is risk-important. The intent of the PRA review was to provide an additional screen to the deterministic criteria. This review was accomplished in the generic Westinghouse Owners Group submittal WCAP-11618 and Addendum 1 to WCAP-11618 (Reference 1). The results of this generic review have been confirmed by I&M for the applicable CNP Units 1 and 2 Specifications to be relocated. Where Reference 1 did not review a CNP Units 1 and 2 Technical Specification against the criteria of Reference 3, I&M performed a review similar (but not identical) to that described below for Reference 1. The results of these reviews are presented in Appendix B.

Assumptions and Approach

The WCAP-11618 evaluation of the risk impact of the Technical Specifications that are relocation candidates was based on the following:

a. It was assumed that any of the Technical Specifications that were to be relocated would be transferred to other documents subject to control by the utility under the 10 CFR 50.59 process.

b. The risk criteria used in determining the disposition of a Technical Specification were the following:

   1. If the Technical Specification contained constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk, it should be retained;

   2. If the Technical Specification included items involved in one of these dominant sequences but had an insignificant impact on the probability or severity of that sequence, it was proposed to be relocated to another controlled document; and

   3. If the Technical Specification was not involved in risk dominant sequences, it was proposed to be relocated to another controlled document.

c. The measures related to risk used in this evaluation were core melt frequency and off-site health effects. These measures were consistent with the Final Policy Statement on Technical Specifications and the Safety Goal and Severe Accident Policy Statements.

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3. PRA INSIGHTS (continued)

d. The criteria used to determine if a sequence was risk dominant was the following: For core melt, any sequence whose frequency was commonly found to be greater than 1 X 10^{-6} per reactor year was maintained as a possible dominant sequence as a conservative first cut. This was roughly 2% of the total core melt frequency of 5 X 10^{-5} for typical PRAs. Each specific sequence identified in the screening of the Technical Specifications was evaluated against the above conservative criterion to determine if it was risk dominant.

For off-site health effects, any sequence whose frequency of serious radioactive release was commonly found to be greater than 1 x 10^{-7} per reactor year was considered to be a dominant risk sequence for the purposes of WCAP-11618. This criterion was in agreement with the NRC position in the Safety Goal Policy for a goal of 1 X 10^{-6} for a total frequency of severe off-site release, and no greater than 1 X 10^{-7} for an individual sequence.

e. Included in Section 4.0 of WCAP-11618, were two tables (Tables 3 and 4) which contained representative sequences for all identified types of initiating events considered in formal risk assessments for two types of reference plants. Table 3 was representative of a plant with a large dry containment and Table 4 contained the dominant accident sequences for a plant with a subatmospheric containment. These lists were based on industry PRAs and were reviewed for consistency with NRC sponsored PRA programs. The results were found to be consistent.

Systems identified in Tables 3 and 4 of Section 4.0 of WCAP-11618 that contributed significantly to risk as defined in Paragraph d above were listed in Tables 3A, 3B, 4A and 4B of Section 4.0. These identified systems as well as sequences and the risk dominant initiating events from Tables 3 and 4 which were involved in typical dominant core melt and serious release sequences from formal risk assessments were used to screen the requirements of the Technical Specifications reviewed. Those Technical Specifications whose requirements were relevant to these systems, sequences, and initiating events were further evaluated for risk dominance. The remaining Technical Specifications were evaluated on the basis of risk insights from references listed in Section 4.0, Appendix B of WCAP-11618. If the requirements of a Technical Specification were not found to be modeled in any reference and no significant issues were identified from a review of the risk insights, the conclusion was that it did not contain constraints of <u>prime importance</u> to limiting the likelihood or severity of sequences that are <u>commonly found</u> to <u>dominate</u> risk.

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4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the CNP Units 1 and 2 Technical Specifications. The following Summary Disposition Matrix is a summary of that application indicating which Specifications are being retained or relocated, the criteria for inclusion, if applicable, the NRC results of the criteria application as expressed in the NRC Staff Review of NSSS Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988, and any necessary explanatory notes. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A, except as noted in the Summary Disposition Matrix. In addition, Appendix B includes a summary of the evaluations performed for those CNP Units 1 and 2 specific Technical Specifications not evaluated in WCAP-11618.

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5. REFERENCES

1. WCAP-11618 (and Addendum 1), "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program — Phase II Task 5, Criteria Application," November 1987.

2. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2, April 2001.

3. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX
FOR
CNP UNITS 1 AND 2

SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
1.0	**DEFINITIONS**	1.1	YES	This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification selection criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the selection criteria, will remain as definitions in this section of Technical Specifications.
2.0	**SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS**	2.0		
2.1	Safety Limits	2.1		
2.1.1	Reactor Core	2.1.1	YES	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.
2.1.2	Reactor Coolant System Pressure	2.1.2	YES	Same as above.
2.2	Limiting Safety System Settings	3.3.1		
2.2.1	Reactor Trip System Instrumentation Setpoints	3.3.1	YES-3	The RTS LSSS have been included as part of the RTS instrumentation Specification, which has been retained since the Functions either actuate to mitigate consequences of design basis accidents and transients or are retained as directed by the NRC as the Functions are part of the RTS.
3/4.0	**LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS - APPLICABILITY**	3.0		
3.0.1	Operational Modes	LCO 3.0.1	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 2.
3.0.2	Noncompliance	LCO 3.0.2	YES	Same as above.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3.0.3	Generic Actions	LCO 3.0.3	YES	Same as above.
3.0.4	Entry into Operational Modes	LCO 3.0.4	YES	Same as above.
3.0.5	Operability Exception	3.8.1	YES	The application of Technical Specification selection criteria is not appropriate. However, this exception to the definition of OPERABILITY has been included as part of the Required Actions in new LCO 3.8.1.
3.0.6	Actions Exceptions	LCO 3.0.5	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 2.
4.0.1	Operational Modes	SR 3.0.1	YES	Same as above.
4.0.2	Time of Performance	SR 3.0.2	YES	Same as above.
4.0.3	Noncompliance	SR 3.0.3	YES	Same as above.
4.0.4	Entry into Operational Modes	SR 3.0.4	YES	Same as above.
4.0.5	ASME Code Class 1, 2, and 3 Components	5.5.6	YES	This Specification is actually a Surveillance Requirement which has been retained in the Administrative Controls programs for Inservice Testing.
4.0.6	Deleted by Amendments 243 (Unit 1) and 224 (Unit 2)	NA	NA	
4.0.7	Deleted by Amendments 243 (Unit 1) and 224 (Unit 2)	NA	NA	
4.0.8 (Unit 2 only)	Deleted by Amendment 224	NA	NA	
4.0.9 (Unit 2 only)	Deleted by Amendment 224	NA	NA	
3/4.1	REACTIVITY CONTROL SYSTEMS	3.1		
3/4.1.1	Boration Control			
3/4.1.1.1	Shutdown Margin - Tavg >200 F	3.1.1	YES-2	
3/4.1.1.2	Shutdown Margin - Tavg ≤ 200 F	3.1.1	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3/4.1.1.3	Boron Dilution	Deleted	NO	Deleted, see Boron Dilution technical change discussion in the Discussion of Changes for CTS 3/4.1.1.3.
3/4.1.1.4	Moderator Temperature Coefficient	3.1.3	YES-2	
3/4.1.1.5	Minimum Temperature for Criticality	3.4.2	YES-2	
3/4.1.2	Boration Systems			
3/4.1.2.1	Flow Paths - Shutdown	Relocated	NO	See Appendix A, Page 1.
3/4.1.2.2	Flow Paths - Operating	Relocated	NO	See Appendix A, Page 3.
3/4.1.2.3	Charging Pump - Shutdown	Relocated	NO	See Appendix A, Page 5. The LCO 3.1.2.3.b requirements have been deleted. See Charging Pump - Shutdown technical change discussion in the Discussion of Changes for CTS 3/4.1.2.3. Asterisk requirement in LCO 3.1.2.3.b maintained in ITS 3.4.12.
3/4.1.2.4	Charging Pumps - Operating	Relocated	NO	See Appendix A, Page 6.
3/4.1.2.5	Boric Acid Transfer Pumps - Shutdown	Relocated	NO	See Appendix A, Page 1.
3/4.1.2.6	Boric Acid Transfer Pumps - Operating	Relocated	NO	See Appendix A, Page 3.
3/4.1.2.7	Borated Water Sources - Shutdown	Relocated	NO	See Appendix A, Page 7.
3/4.1.2.8	Borated Water Sources - Operations (Unit 1); Borated Water Sources- Operating (Unit 2)	Relocated	NO	See Appendix A, Page 9.
3/4.1.3	Movable Control Assemblies			
3/4.1.3.1	Group Height	3.1.4	YES-2	
3/4.1.3.2	Position Indicator Channels (Unit 1); Position Indicator Channels - Operating (Unit 2)	3.1.7	YES-2	
3/4.1.3.3 (Unit 2 only)	Deleted by Amendment 194	NA	NA	
3/4.1.3.3 (Unit 1); 3/4.1.3.4 (Unit 2)	Rod Drop Time	3.1.4	YES-2	This Specification has been incorporated as a Surveillance Requirement (SR 3.1.4.3) in ITS 3.1.4.
3/4.1.3.4 (Unit 1); 3/4.1.3.5 (Unit 2)	Shutdown Rod Insertion Limit	3.1.5	YES-2	
3/4.1.3.5 (Unit 1); 3/4.1.3.6 (Unit 2)	Control Rod Insertion Limits	3.1.6	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.2	**POWER DISTRIBUTION LIMITS**	3.2		
3/4.2.1	Axial Flux Difference	3.2.3	YES-2	
3/4.2.2	Heat Flux Hot Channel Factor - FQ(Z)	3.2.1	YES-2	
3/4.2.3	Nuclear Enthalpy Hot Channel Factor - $F_{\Delta H}^{(N)}$	3.2.2	YES-2	
3/4.2.4	Quadrant Power Tilt Ratio	3.2.4	YES-2	
3/4.2.5	DNB Parameters (Unit 1); DNB and Tavg Operating Parameters (Unit 2)	3.4.1	YES-2	
3/4.2.6	Allowable Power Level - APL	3.2.1	YES-2	
3/4.3	**INSTRUMENTATION**	3.3		
3/4.3.1	Reactor Trip System Instrumentation	3.3.1 3.3.8	YES-3	
3/4.3.2	Engineered Safety Feature Actuation System Instrumentation	3.3.2 3.3.5 3.3.6	YES-3	
3/4.3.3	Monitoring Instrumentation			
3/4.3.3.1	Radiation Monitoring Instrumentation			
Instrument 1.A.i	Area Monitors - Upper Containment	Deleted	NO	Deleted, see Radiation Monitoring Instrumentation technical change discussion in the Discussion of Changes for CTS 3/4.3.3.1.
Instrument 1.A.ii	Area Monitors - Containment High Range	3.3.3	YES-3	
Instrument 1.B	Process Monitors	3.4.15	YES-1	
Instrument 1.C	Noble Gas Effluent Monitors	Deleted	NO	Deleted, see Radiation Monitoring Instrumentation technical change discussion in the Discussion of Changes for CTS 3/4.3.3.1.
Instrument 2.A/B	MODE 6 Monitors	3.3.6	YES-3	
Instrument 3.A	Spent Fuel Storage	Deleted	NO	Deleted, see Radiation Monitoring Instrumentation technical change discussion in the Discussion of Changes for CTS 3/4.3.3.1.
3/4.3.3.2	Movable Incore Detectors	Relocated	NO	See Appendix A, Page 11.
3/4.3.3.3	Seismic Instrumentation	Relocated	NO	See Appendix A, Page 12.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.3.3.4	Meteorological Instrumentation	Relocated	NO	See Appendix A, Page 13.
3/4.3.3.5	Remote Shutdown Instrumentation	3.3.4	YES-4	
3/4.3.3.5.1	Appendix R Remote Shutdown Instrumentation	Relocated	NO	See Appendix A, Page 14.
3/4.3.3.6 (Unit 1); 3/4.3.3.7 (Unit 2)	Deleted by Amendments 120 (Unit 1) and 82 (Unit 2)	NA	NA	
3/4.3.3.7 (Unit 1); 3/4.3.3.8 (Unit 2)	Deleted by Amendments 208 (Unit 1) and 192 (Unit 2)	NA	NA	
3/4.3.3.8 (Unit 1); 3/4.3.3.6 (Unit 2)	Post-Accident Instrumentation	3.3.3	YES-3	See Appendix A, Page 15. Instrumentation that does not monitor Regulatory Guide 1.97 Type A or Category 1 variables has been relocated in accordance with the guidance provided in NUREG-1431, Revision 2.
3/4.3.3.9	Explosive Gas Monitoring Instrumentation	Relocated	NO	See Appendix A, Page 17.
3/4.4	**REACTOR COOLANT SYSTEM**	3.4		
3/4.4.1	Reactor Coolant Loops and Coolant Circulation			
3/4.4.1.1	Startup and Power Operation	3.4.4	YES-2	
3/4.4.1.2	Hot Standby	3.4.5	YES-3	
3/4.4.1.3	Hot Shutdown	3.4.6	YES-3	
3/4.4.1.4	Cold Shutdown - Loops Filled	3.4.7 3.4.12	YES-4 YES-2	
3/4.4.1.5	Cold Shutdown - Loops Not Filled	3.4.8	YES-4	
3/4.4.2	Safety Valves - Shutdown	3.4.10	YES-3	
3/4.4.3	Safety Valves - Operating	3.4.10	YES-3	
3/4.4.4	Pressurizer	3.4.9	YES-2	
3/4.4.5	Steam Generators	3.4.13 5.5	YES-2 YES	This Specification has been incorporated as a Surveillance Requirement (SR 3.4.13.2) in ITS 3.4.13 and a program in ITS 5.5.
3/4.4.6	Reactor Coolant System Leakage			
3/4.4.6.1	Leakage Detection Systems	3.4.15	YES-1	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.4.6.2	Operational Leakage	3.4.13 3.4.14 3.5.5	YES-2	
3/4.4.7	Chemistry	Relocated	NO	See Appendix A, Page 18.
3/4.4.8	Specific Activity	3.4.16	YES-2	
3/4.4.9	Pressure/Temperature Limits			
3/4.4.9.1	Reactor Coolant System	3.4.3	YES-2	
3/4.4.9.2	Pressurizer	Relocated	NO	See Appendix A, Page 19.
3/4.4.9.3	Overpressure Protection Systems	3.4.12	YES-2	
3/4.4.10	Structural Integrity			
3/4.4.10.1	ASME Code Class 1, 2 and 3 Components	Relocated	NO	See Appendix A, Page 20. The Reactor Coolant Pump Flywheel Surveillance is being retained as a Program in ITS 5.5
3/4.4.11	Relief Valves - Operating	3.4.11	YES-3	
3/4.4.12	Reactor Coolant Vent System			
3/4.4.12.1	Reactor Vessel Head Vents	Relocated	NO	See Appendix A, Page 22.
3/4.4.12.2	Pressurizer Steam Space Vents	Relocated	NO	See Appendix A, Page 22.
3/4.5	**EMERGENCY CORE COOLING SYSTEMS**	3.5		
3/4.5.1	Accumulators	3.5.1	YES-3	
3/4.5.2	ECCS Subsystems - Tavg ≥ 350 F	3.5.2	YES-3	
3/4.5.3	ECCS Subsystems - Tavg <350 F	3.5.3 3.4.12	YES-3 YES-2	
3/4.5.4	Deleted by Amendments 158 (Unit 1) and 142 (Unit 2)			
3/4.5.5	Refueling Water Storage Tank	3.5.4	YES-3	
3/4.6	**CONTAINMENT SYSTEMS**	3.6		
3/4.6.1	Primary Containment			
3/4.6.1.1	Containment Integrity	3.6.1	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.6.1.2	Containment Leakage	3.6.1	YES-3	Containment leakage is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1 and a program in ITS 5.5.16.
3/4.6.1.3	Containment Air Locks	3.6.2	YES-3	
3/4.6.1.4	Internal Pressure	3.6.4	YES-2	
3/4.6.1.5	Air Temperature	3.6.5	YES-2	
3/4.6.1.6	Containment Structural Integrity	3.6.1	YES-3.	Containment vessel structural integrity is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1.
3/4.6.1.7	Containment Ventilation System	3.6.3	YES-3	Containment purge valves are being retained as a Surveillance Requirement (SR 3.6.3.1) in ITS 3.6.3.
3/4.6.2	Depressurization and Cooling Systems			
3/4.6.2.1	Containment Spray System	3.6.6	YES-3	
3/4.6.2.2	Spray Additive Systems	3.6.7	YES-3	
3/4.6.3	Containment Isolation Valves	3.6.3	YES-3	
3/4.6.4	Combustible Gas Control			
3/4.6.4.1	Hydrogen Analyzers	3.3.3	YES-3	
3/4.6.4.2	Electric Hydrogen Recombiners-W	3.6.8	YES-3	
3/4.6.4.3	Distributed Ignition System	3.6.9	YES-4	
3/4.6.5	Ice Condenser			
3/4.6.5.1	Ice Bed	3.6.11	YES-3	
3/4.6.5.2	Ice Bed Temperature Monitoring System	Relocated	NO	See Appendix A, Page 23.
3/4.6.5.3	Ice Condenser Doors	3.6.12	YES-3	
3/4.6.5.4	Inlet Door Position Monitoring System	Relocated	NO	See Appendix A, Page 24.
3/4.6.5.5	Divider Barrier Personnel Access Doors and Equipment Hatches	3.6.13	YES-3	
3/4.6.5.6	Containment Air Recirculation Systems	3.6.10	YES-3	
3/4.6.5.7	Floor Drains	3.6.14	YES-3	
3/4.6.5.8	Refueling Canal Drains	3.6.14	YES-3	
3/4.6.5.9	Divider Barrier Seal	3.6.13	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.7	**PLANT SYSTEMS**	3.7		
3/4.7.1	Turbine Cycle			
3/4.7.1.1	Safety Valves	3.7.1	YES-3	
3/4.7.1.2	Auxiliary Feedwater System	3.7.5	YES-3	
3/4.7.1.3	Condensate Storage System	3.7.6	YES-2, 3	
3/4.7.1.4	Activity	3.7.17	YES-2	
3/4.7.1.5	Steam Generator Stop Valves	3.7.2	YES-3	
3/4.7.2	Steam Generator Pressure/Temperature Limitation	Relocated	NO	See Appendix A, Page 25.
3/4.7.3	Component Cooling Water System	3.7.7	YES-3	
3/4.7.4	Essential Service Water System	3.7.8	YES-3	
3/4.7.5.1	Control Room Emergency Ventilation System	3.7.10 3.3.7	YES-3	
3/4.7.5.2	Control Room Air Conditioning System	3.7.11	YES-3	
3/4.7.6	ESF Ventilation System	3.7.12	YES-3	
3/4.7.7 (Unit 1); 3/4.7.8 (Unit 2)	Sealed Source Contamination	Relocated	NO	See Appendix A, Page 26.
3/4.7.8 (Unit 1); 3/4.7.7 (Unit 2)	Snubbers	Deleted	NO	Deleted, see Snubbers technical change discussion in the Discussion of Changes for CTS 3/4.7.8 (Unit 1) and 3/4.7.7 (Unit 2).
3/4.8	**ELECTRICAL POWER SYSTEM**	3.8		
3/4.8.1	A.C. Sources			
3/4.8.1.1	Operating	3.8.1 3.8.3	YES-3	
3/4.8.1.2	Shutdown	3.8.2 3.8.3	YES-3	
3/4.8.2	Onsite Power Distribution Systems			
3/4.8.2.1	A.C. Distribution - Operating	3.8.7 3.8.9	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.8.2.2	A.C. Distribution - Shutdown	3.8.8 3.8.10	YES-3	
3/4.8.2.3	D.C. Distribution - Operating	3.8.4 3.8.6 3.8.9	YES-3	
3/4.8.2.4	D.C. Distribution - Shutdown	3.8.5 3.8.6 3.8.10	YES-3	
3/4.8.2.5	D.C. Distribution - Operating - Train N Battery System	3.8.4 3.8.6 3.8.9	YES-3	
3/4.8.3	Alternative A.C. Power Sources	Deleted	NO	Deleted, see Alternative AC Power Sources technical change discussion in the Discussion of Changes for CTS 3/4.8.3.1.
3/4.9	**REFUELING OPERATIONS**	**3.9**		
3/4.9.1	Boron Concentration	3.9.1	YES-2	
3/4.9.2	Instrumentation	3.9.2	YES-3	
3/4.9.3	Decay Time	Deleted	NO	Deleted, see Decay Time technical change discussion in the Discussion of Changes for CTS 3/4.9.3.
3/4.9.4	Containment Building Penetrations	3.9.3	YES-3	
3/4.9.5	Communications	Relocated	NO	See Appendix A, Page 27.
3/4.9.6	Deleted by Amendments 267 (Unit 1) and 248 (Unit 2)			
3/4.9.7	Deleted by Amendments 267 (Unit 1) and 248 (Unit 2)			
3/4.9.8	Residual Heat Removal and Coolant Circulation			
3/4.9.8.1	High Water Level	3.9.4 3.9.5	YES-4	
3/4.9.8.2	Low Water Level	3.9.5	YES-4	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.9.9	Containment Purge and Exhaust Isolation System	3.3.6 3.9.3	YES-3	
3/4.9.10	Water Level - Reactor Vessel	3.9.6	YES-2	
3/4.9.11	Storage Pool Water Level	3.7.14	YES-2, 3	
3/4.9.12	Storage Pool Ventilation System	3.7.13	YES-3	
3/4.9.13	Spent Fuel Cask Movement	Deleted	NO	Deleted, see Spent Fuel Cask Movement technical change discussion in the Discussion of Changes for CTS 3/4.9.13.
3/4.9.14	Spent Fuel Cask Drop Protection System	Deleted	NO	Deleted, see Spent Fuel Cask Drop Protection System technical change discussion in the Discussion of Changes for CTS 3/4.9.14.
3/4.9.15	Storage Pool Boron Concentration	3.7.15	YES-2	
3/4.10	**SPECIAL TEST EXCEPTIONS**	**NA**		
3/4.10.1	Shutdown Margin	Deleted	NO	This Specification is provided to allow relaxation of the SDM LCO under certain specific conditions for testing. Direct application of the Technical Specification selection criteria is not appropriate. However, this special test exception is not necessary in the ITS since the SDM limit is in the COLR.
3/4.10.2	Group Height, Insertion and Power Distribution Limits	Deleted	NO	This Specification is provided to allow relaxation of certain LCOs under certain specific conditions for testing. Direct application of the Technical Specification selection criteria is not appropriate. However, this special test exception is not necessary in the ITS since the specific testing has been completed.
3/4.10.3 (Unit 1 only)	Pressure/Temperature Limitation - Reactor Criticality	Deleted	NO	Deleted, see Pressure/Temperature Limitation - Reactor Criticality technical change discussion in the Discussion of Changes for CTS 3/4.10.3.
3/4.10.4 (Unit 1); 3/4.10.3 (Unit 2)	Physics Tests	3.1.8	YES	This Specification is provided to allow relaxation of certain LCOs under certain specific conditions for testing. Direct application of the Technical Specification selection criteria is not appropriate. However, this special test exception is not necessary in the ITS since the specific testing has been completed.
3/4.10.5 (Unit 1); 3/4.10.4 (Unit 2)	Natural Circulation Tests (Unit 1); Reactor Coolant Loops (Unit 2)	Deleted	NO	Same as above.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.10.5 (Unit 2 only)	Deleted by Amendment 194	NA	NA	
3/4.11	**RADIOACTIVE EFFLUENTS**	NA		
3/4.11.1	Liquid Holdup Tanks	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons; dated October 25, 1993.
3/4.11.2	Gaseous Effluents			
3/4.11.2.1	Explosive Gas Mixture	5.5.10	YES	Same as above.
3/4.11.2.2	Gas Storage Tanks	5.5.10	YES	Same as above.
5.0	**DESIGN FEATURES**	3.7.16 4.0	YES	Application of Technical Specification selection criteria is not appropriate. However, specific portions of Design Features will be included in Technical Specifications as required by 10 CFR 50.36.
6.0	**ADMINISTRATIVE CONTROLS**	5.0	YES	Application of Technical Specification selection criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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APPENDIX A

JUSTIFICATION FOR SPECIFICATION RELOCATION

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.1 FLOW PATHS - SHUTDOWN
3/4.1.2.5 BORIC ACID TRANSFER PUMPS - SHUTDOWN

LCO STATEMENT:

3/4.1.2.1

As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

3/4.1.2.5

At least one boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.1 FLOW PATHS - SHUTDOWN
3/4.1.2.5 BORIC ACID TRANSFER PUMPS - SHUTDOWN (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Flow Paths - Shutdown LCO and Surveillances and the Boric Acid Transfer Pumps - Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.2 FLOW PATHS - OPERATING
3/4.1.2.6 BORIC ACID TRANSFER PUMPS - OPERATING

LCO STATEMENT:

3/4.1.2.2

Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

3/4.1.2.6

At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.2 FLOW PATHS - OPERATING
3/4.1.2.6 BORIC ACID TRANSFER PUMPS - OPERATING (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Flow Paths - Operating LCO and Surveillances and the Boric Acid Transfer Pumps - Operating LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.3 CHARGING PUMP - SHUTDOWN

LCO STATEMENT:

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. It should be noted that this LCO (part b) has requirements associated with the safe shutdown requirements of 10 CFR 50, Appendix R, and a requirement concerning the maximum number of charging and safety injection pumps that can be OPERABLE. These requirements are not covered by this discussion.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Charging Pump - Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.4 CHARGING PUMPS - OPERATING

LCO STATEMENT:

At least two charging pumps shall be OPERABLE.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Charging Pumps - Operating LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.7 BORATED WATER SOURCES - SHUTDOWN

LCO STATEMENT:

As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

   1. A minimum usable borated water volume of 5000 gallons,

   2. Between 6,550 and 6,990 ppm of boron, and

   3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

   1. A minimum usable borated water volume of 90,000 gallons,

   2. A minimum boron concentration of 2400 ppm, and

   3. A minimum solution temperature of 70°F.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-10) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.7 BORATED WATER SOURCES - SHUTDOWN (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Borated Water Sources - Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.8 BORATED WATER SOURCES - OPERATIONS (UNIT 1)/OPERATING (UNIT 2)

LCO STATEMENT:

Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

   1. A minimum contained borated water volume of 8,500 gallons,

   2. Between 6,550 and 6,990 ppm of boron, and

   3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

   1. A minimum contained borated water volume of 375,500 gallons of water,

   2. Between 2400 and 2600 ppm of boron, and

   3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-10) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

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3/4.1.2.8 BORATED WATER SOURCES - OPERATIONS (UNIT 1)/OPERATING (UNIT 2) (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Borated Water Sources - Operations/Operating LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.2 MOVABLE INCORE DETECTORS

LCO STATEMENT:

The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

DISCUSSION:

This Specification ensures the OPERABILITY of Movable Incore Detector Instrumentation when required to monitor the flux distribution within the core. The System is used for periodic Surveillance of the reactor core power distribution, and calibration of the excore neutron flux detectors, but is not assumed in any DBA analysis and does not mitigate an accident.

COMPARISON TO SCREENING CRITERIA:

1. This system is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. This system is not a process variable that is an initial condition in a DBA or transient analyses.

3. This system does not act as a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-12) and summarized in Table 1 of WCAP-11618, the loss of Movable Incore Detectors was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Movable Incore Detectors LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.3 SEISMIC INSTRUMENTATION

LCO STATEMENT:

The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

DISCUSSION:

In the event of an earthquake, seismic instrumentation is required to permit comparison of the measured response to that used in the design basis of the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR 100. Since this is determined after the event has occurred, it has no bearing on the mitigation of any DBA.

COMPARISON TO SCREENING CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. These instruments do not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-22), and summarized in Table 1 of WCAP-11618, the loss of seismic monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Seismic Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

LCO STATEMENT:

The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

DISCUSSION:

Meteorological instrumentation is used to measure environmental parameters that may affect distribution of fission products and gases following a design basis accident (DBA), but it is not an input assumption for any DBA analysis and does not mitigate the accident. Meteorological information is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

COMPARISON TO SCREENING CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. These instruments do not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-23), and summarized in Table 1 of WCAP-11618, the loss of meteorological monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Meteorological Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

LCO STATEMENT:

The Appendix R remote shutdown instrumentation channels shown in Table 3.3-9A shall be OPERABLE with an opposite unit power supply available and with read out capability at the LSI panels.

DISCUSSION:

The Appendix R Remote Shutdown Instrumentation is used to ensure that a fire will not preclude achieving safe shutdown. This instrumentation is independent of areas where a fire could damage systems normally used to shutdown the reactor. However, the instrumentation is not used to detect a degradation of the reactor coolant pressure boundary, and is not assumed to mitigate a design basis accident (DBA) or transient event. The Appendix R Remote Shutdown Instrumentation capability is consistent with the requirements of 10 CFR 50, Appendix R. The acceptability of the relocation of the Appendix R Technical Specification requirements from the plant Technical Specifications has already been endorsed by the NRC as indicated in Generic Letter 86-10.

COMPARISON TO SCREENING CRITERIA:

1. The Appendix R Remote Shutdown Instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Appendix R Remote Shutdown Instrumentation does not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The Appendix R Remote Shutdown Instrumentation is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Appendix B, page 1, I&M found the loss of the Appendix R Remote Shutdown Instrumentation to be a non-significant risk contributor to core damage frequency and offsite releases.

CONCLUSION:

Since the screening criteria have not been satisfied, the Appendix R Remote Shutdown Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.8 (Unit 1); POST-ACCIDENT INSTRUMENTATION
3/4.3.3.6 (Unit 2)

LCO STATEMENT:

The post-accident monitoring instrumentation channels shown in Table 3.3-11 (Unit 1) and 3.3-10 (Unit 2) shall be OPERABLE.

DISCUSSION:

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to ensure sufficient information is available following an accident to allow an operator to verify the response of automatic safety systems, and to take preplanned manual actions to accomplish a safe shutdown of the plant.

The NRC position on application of the deterministic screening criteria to post-accident monitoring instrumentation is documented in letter dated May 9, 1988 from T.E. Murley (NRC) to W.S. Wilgus (NRC Split Report to Owners Groups). The position taken was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the CNP Units 1 and 2 Regulatory Guide 1.97 instruments. Those instruments meeting these criteria have remained in Technical Specifications. The instruments not meeting this criteria will be relocated from the Technical Specifications to plant controlled documents.

A review of the CNP Units 1 and 2 UFSAR and the NRC Regulatory Guide 1.97 Safety Evaluation for CNP Units 1 and 2 shows that the following Unit 1 CTS Tables 3.3-11 and 4.3-7 and Unit 2 CTS Tables 3.3-10 and 4.3-10 Instruments do not meet Category 1 or Type A requirements.

Instrument 9 Boric Acid Tank Solution Level
Instrument 12 PORV Position Indicator - Limit Switches
Instrument 13 PORV Block Valve Position Indicator - Limit Switches
Instrument 14 Safety Valve Position Indicator - Acoustic Monitor
Instrument 17 Containment Sump Level

COMPARISON TO SCREENING CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The monitored parameters are not process variables, design features, or operating restrictions that are initial conditions of a DBA or transient.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-25) and summarized in Table 1 of WCAP-11618, the loss of the (above listed) instruments were found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.8 (Unit 1); POST-ACCIDENT INSTRUMENTATION (continued)
3/4.3.3.6 (Unit 2)

CONCLUSION:

Since the screening criteria have not been satisfied for instruments which do not meet Regulatory Guide 1.97 Type A variable requirements or Category 1 variable requirements, their associated LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.9 EXPLOSIVE GAS MONITORING INSTRUMENTATION

LCO STATEMENT:

The explosive gas monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded.

DISCUSSION:

The explosive gas monitor Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous waste processing system is adequately monitored, which will help ensure that the concentration is maintained below the flammability limit. However, the system is designed to contain detonations, and detonations would not affect the function of any safety related equipment. The concentration of oxygen in the gaseous Waste Processing System is not an initial assumption of any design basis accident (DBA) or transient analysis.

COMPARISON TO SCREENING CRITERIA:

1. The explosive gas monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The explosive gas monitoring instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient. In addition, excessive system oxygen is not an indication of a DBA or transient.

3. The explosive gas monitoring instrumentation is not part of a primary success path in the mitigation of a DBA or transient. In addition, excessive oxygen discharge is not part of a primary success path in mitigating a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the loss of the explosive gas monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Explosive Gas Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.4.7 CHEMISTRY

LCO STATEMENT:

The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

DISCUSSION:

Poor coolant water chemistry contributes to the long term degradation of system materials of construction, and thus is not of immediate importance to the unit operator. Reactor coolant water chemistry is monitored for a variety of reasons. One reason is to reduce the possibility of failures in the Reactor Coolant System pressure boundary caused by corrosion. However, the chemistry monitoring activity is of a long term preventative purpose rather than mitigative.

COMPARISON TO SCREENING CRITERIA:

1. Reactor coolant water chemistry is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor coolant water chemistry is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Reactor coolant water chemistry is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-40) and summarized in Table 1 of WCAP-11618, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Chemistry LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.4.9.2 PRESSURIZER

LCO STATEMENT:

The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,
b. A maximum cooldown of 200°F in any one hour period, and
c. A maximum spray water temperature differential of 320°F.

DISCUSSION:

The heatup and cooldown rate limits and spray water differential limit are placed on the pressurizer to prevent non-ductile failure and assure compatibility of operation with the fatigue analysis performed. The limits meet the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. These limitations are consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Statement the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

COMPARISON TO SCREENING CRITERIA:

1. Pressurizer heatup and cooldown limits and spray water differential limit are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Pressurizer heatup and cooldown limits and spray water differential limit are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Pressurizer heatup and cooldown limits and spray water differential limit are not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-41) and summarized in Table 1 of WCAP-11618, the pressurizer heatup and cooldown limits and spray water differential limit were found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Pressurizer LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.4.10.1 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LCO STATEMENT:

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

DISCUSSION:

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the life of the component. ASME Code Class 1, 2, and 3 components are monitored so that the possibility of component structural failure does not degrade the safety function of the system. The monitoring activity is of a preventive nature rather than a mitigative action. Other Technical Specifications require important systems to be OPERABLE (for example, Emergency Core Cooling Systems) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this Specification to ensure immediate OPERABILITY of safety systems.

Further, this Technical Specification prescribes inspection requirements that are performed during plant shutdown. It is, therefore, not directly important for responding to design basis accidents.

COMPARISON TO SCREENING CRITERIA:

1. The inspections stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary during operations prior to a design basis accident (DBA).

2. The inspections stipulated by this Specification are not a process variable, design feature, or operating restriction that is an initial assumption in a DBA or transient.

3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification only specifies inspection requirements for these components, and these inspections can only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.

4. As discussed in Section 4.0 (Appendix A, page A-43) and summarized in Table 1 of WCAP-11618 the assurance of operability of the entire system as verified in the system operability Specification dominates the risk contribution of the system. The lack of a long term assurance of structural integrity as stipulated by this Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Structural Integrity - ASME Code Class 1, 2, and 3 Components LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. In addition, surveillances, except for the reactor coolant pump (RCP) flywheel inspection, are already required by regulations in 10 CFR 50.55a to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.4.10.1 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2 AND 3 COMPONENTS (continued)

The RCP flywheel inspection requirement is not covered by other regulatory requirements and is needed for safe operation of the plant; therefore, this requirement will be maintained in the CNP Units 1 and 2 Improved Technical Specifications. Chapter 5.0 of the CNP Units 1 and 2 Improved Technical Specifications will contain a section which provides a programmatic approach to the requirements relating to the structural integrity of ASME Code Class 1, 2, and 3 components.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.4.12.1 REACTOR VESSEL HEAD VENTS
3/4.4.12.2 PRESSURIZER STEAM SPACE VENTS

LCO STATEMENT:

3/4.4.12.1

At least one of the Reactor Vessel head vent paths, consisting of two remotely operated valves in series, powered from Class 1E DC busses, shall be OPERABLE and closed.

3/4.4.12.2

At least one of the pressurizer steam space vent paths, each consisting of two remotely operated valves in series, powered from Class 1E DC busses, shall be OPERABLE and closed.

DISCUSSION:

The reactor vessel head and pressurizer steam space vents are provided to exhaust noncondensible gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a loss-of-coolant accident (LOCA). Their function, capabilities, and testing requirements are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," however, the operation of reactor vessel head vents is not part of the primary success path. The operation of these vents is an operator action after the event has occurred, and is only required when there is indication that natural circulation is not occurring.

COMPARISON TO SCREENING CRITERIA:

1. Reactor vessel head and pressurizer steam space vents are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor vessel head and pressurizer steam space vents are not process variables, design features, or operating restrictions that are an initial condition of a DBA or transient.

3. Reactor vessel head and pressurizer steam space vents are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-44) and summarized in Table 1 of WCAP-11618, the reactor vessel head and pressurizer steam space vents were found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Reactor Vessel Head Vents LCO and Surveillances and Pressurizer Steam Space Vents LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

LCO STATEMENT:

The ice bed temperature monitoring system shall be OPERABLE with at least 2 OPERABLE RTD channels in the ice bed at elevations 652' 2 1/4", 672' 5 1/4" and 696' 2 1/4" for each one third of the ice condenser.

DISCUSSION:

The Ice Bed Temperature Monitoring System monitors the temperature of the ice bed to ensure that the ice bed temperature does not increase above the required limits undetected. However, the Ice Bed Temperature Monitoring System is not required to ensure the ice bed temperature is maintained within limits. Another Technical Specification (that is being retained) will continue to ensure that temperature is maintained within the required limits.

COMPARISON TO SCREENING CRITERIA:

1. The Ice Bed Temperature Monitoring System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).
2. The Ice Bed Temperature Monitoring System is are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.
3. The Ice Bed Temperature Monitoring System is not part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Section 4.0 (Appendix A, page A-78) and summarized in Table 1 of WCAP-11618, the Ice Bed Temperature Monitoring System was found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Ice Bed Temperature Monitoring System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

LCO STATEMENT:

The inlet door position monitoring system shall be OPERABLE.

DISCUSSION:

The Inlet Door Position Monitoring System monitors the position of the ice bed inlet doors during normal operation to ensure that the ice bed inlet doors do not open (which could allow the ice bed temperature to increase above the required limits). However, the Inlet Door Position Monitoring System is not required to ensure the inlet doors remain closed and ice bed temperature is maintained within limits. Other Technical Specifications (that are being retained) will continue to ensure that the inlet doors remain closed and temperature is maintained within the required limits.

COMPARISON TO SCREENING CRITERIA:

1. The Inlet Door Position Monitoring System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The Inlet Door Position Monitoring System is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The Inlet Door Position Monitoring System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-78) and summarized in Table 1 of WCAP-11618, the Inlet Door Position Monitoring System was found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Inlet Door Position Monitoring System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LCO STATEMENT:

The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

DISCUSSION:

The limitation on steam generator pressures and temperatures ensures that pressure-induced stresses on the steam generators do not exceed the maximum allowable fracture toughness limits. These pressure and temperature limits are based on maintaining a steam generator RT_{NDT} sufficient to prevent brittle fracture. As such, the Technical Specification places limits on variables consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Statement the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

COMPARISON TO SCREENING CRITERIA:

1. The steam generator P/T limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The steam generator P/T limits are not process variables, design features, or operating restrictions that are an initial condition of a DBA or transient.

3. The steam generator P/T limits are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-55) and summarized in Table 1 of WCAP-11618, the steam generator P/T limits were found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Steam Generator P/T Limitation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.7.7 (Unit 1); SEALED SOURCE CONTAMINATION
3/4.7.8 (Unit 2)

LCO STATEMENT:

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of 0.005 microcuries of removable contamination.

DISCUSSION:

The limitations on sealed source contamination are intended to ensure that the total body and individual organ irradiation doses do not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

COMPARISON TO SCREENING CRITERIA:

1. Sealed source contamination is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Sealed source contamination is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Sealed source contamination is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-59) and summarized in Table 1 of WCAP-11618, the sealed source contamination being not within limits was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Sealed Source Contamination LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.9.5 COMMUNICATIONS

LCO STATEMENT:

Direct communications shall be maintained between the control room and personnel at the refueling station.

DISCUSSION:

Communication between the control room personnel and personnel performing CORE ALTERATIONS is maintained to ensure that personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and containment personnel. The prompt notification of the control room of a fuel handling accident is not an assumption in the fuel handling accident analysis. While notification is necessary to ensure the control room is isolated to meet the control room operator dose limits in General Design Criteria 19, the fuel handling accident analysis does not take credit for direct communications between the refueling station and the control room (30 minutes is assumed before control room operators isolate the control room).

COMPARISON TO SCREENING CRITERIA:

1. Communications during refueling operations is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Communications during refueling operations is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Communication during refueling operations is not a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-67) and summarized in Table 1 of WCAP-11618, the loss of communications was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Communications LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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APPENDIX B

COOK NUCLEAR PLANT UNITS 1 AND 2
SPECIFIC RISK SIGNIFICANT EVALUATIONS

APPENDIX B
COOK NUCLEAR PLANT UNITS 1 AND 2
SPECIFIC RISK SIGNIFICANT EVALUATIONS

TECHNICAL SPECIFICATION 3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

DESCRIPTION OF REQUIREMENT:

The Appendix R remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown of the facility to a COLD SHUTDOWN condition in the event of a fire in the main control room. This Specification maintains this requirement.

POTENTIAL EFFECT:

Loss of capability to place a unit in COLD SHUTDOWN as a result of a fire in the main control room.

REFERENCED DOCUMENTS UTILIZED:

1. CNP Updated Final Safety Analysis Report, Section 7.7, Operating Control Stations.
2. CNP Technical Specifications, Facility Operating Licenses DPR-58 and DPR-74.
3. CNP Probabilistic Risk Assessment Final Report, Volume 11.
4. 10 CFR Part 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.
5. NRC Generic Letter 86-10, Implementation of Fire Protection Requirements.

COMMENTS:

Although the Appendix R remote shutdown instrumentation has not been specifically evaluated for risk significance either generically or on a plant specific basis, insight based on a review of the referenced documents indicates that the instrumentation is not risk dominant with regards to core damage frequency or off-site health effects. Furthermore, Generic Letter 86-10 identifies conditions under which fire protection related Technical Specifications may be relocated to other administratively controlled documents, subject to the provisions of 10 CFR 50.59, without a significant increase to public health and safety.

CONCLUSION:

Based on a thorough review of the listed references, it is recommended that CNP Technical Specification 3/4.3.3.5.1 be relocated from Technical Specifications to a licensee controlled document.

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SUMMARY OF CHANGES
GENERIC DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT

Change Description	Affected Pages
A self-identified change for Less Restrictive Discussion of Change (L DOC) Category 13 has been made. This change deletes the L DOC Category 13 discussion, since L DOC Category 13 is no longer used in the ITS submittal.	Page 34 of 37.

VOLUME 2

CNP UNITS 1 AND 2 IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

GENERIC DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT

Revision 1

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION FOR ADMINISTRATIVE CHANGES

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve reformatting, renumbering, and rewording of Current Technical Specifications (CTS) with no change in intent. These changes, since they do not involve technical changes to the CTS, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the CTS. These changes also include non-technical modifications of requirements to conform to NEI 01-03, "Writer's Guide for the Improved Standard Technical Specifications," or provide consistency with the Improved Standard Technical Specifications in NUREG-1431. Administrative changes are not intended to add, delete, or relocate any technical requirements of the CTS.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves reformatting, renumbering, and rewording the CTS. The reformatting, renumbering, and rewording process involves no technical changes to the CTS. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION FOR MORE RESTRICTIVE CHANGES

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve adding more restrictive requirements to the Current Technical Specifications (CTS) by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional requirements that decrease allowed outage times, increase the Frequency of Surveillances, impose additional Surveillances, increase the scope of Specifications to include additional plant equipment, increase the Applicability of Specifications, or provide additional actions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides more stringent Technical Specification requirements for the facility. These more stringent requirements do not result in operations that significantly increase the probability of initiating an analyzed event, and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different Technical Specification requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
RELOCATED SPECIFICATIONS

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve relocating Current Technical Specification (CTS) Limiting Conditions for Operations (LCOs) to licensee controlled documents.

The Company has evaluated the CTS using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the ITS. These specifications have been relocated from the CTS to the Technical Requirements Manual, which is incorporated into the Updated Final Safety Analysis Report (UFSAR).

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relocates requirements and Surveillances for structures, systems, components, or variables that do not meet the criteria of 10 CFR 50.36 (c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Donald C. Cook Nuclear Plant Units 1 and 2 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and Surveillances for these affected structures, systems, components, or variables will be relocated from the CTS to an appropriate administratively controlled document which will be incorporated into the UFSAR, thus it will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components, or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59, and are subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose or

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eliminate any requirements, and adequate control of existing requirements will be maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention. In addition, the relocated requirements are moved without change, and any future changes to these requirements will be evaluated per 10 CFR 50.59.

   NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety which can be evaluated. However, the proposed change is consistent with NUREG-1431, issued by the NRC, which allows revising the CTS to relocate these requirements and Surveillances to a licensee controlled document controlled by 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
REMOVED DETAIL CHANGES

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve moving details out of the Current Technical Specifications (CTS) and into the Technical Specifications Bases, the Updated Final Safety Analysis Report (UFSAR), the Technical Requirements Manual (TRM), or other documents under regulatory control such as the CORE OPERATING LIMITS REPORT (COLR), Offsite Dose Calculation Manual (ODCM), Quality Assurance Program Description (QAPD), and Inservice Inspection Program (IIP). The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1431 for format and content.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relocates certain details from the CTS to other documents under regulatory control. The Technical Specification Bases and TRM will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.59 or 10 CFR 50.71(e). Other documents are subject to controls imposed by ITS or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change will not reduce a margin of safety because it has no effect on any assumption of the safety analyses. In addition, the details to be moved from the CTS to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.90 requirement for NRC review and approval of future changes to the relocated details. Not including these details in the Technical Specifications is consistent with NUREG-1431, issued by the NRC, which allows revising the Technical Specifications to relocate these requirements and Surveillances to a licensee controlled document controlled by 10 CFR 50.59, 10 CFR 50.71(e), or other Technical Specification controlled or regulation controlled documents. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 1
RELAXATION OF LCO REQUIREMENTS

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve relaxation of the Current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the ISTS approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis, and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will significantly increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 2
RELAXATION OF APPLICABILITY

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve relaxation of the applicability of Current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

Reactor operating conditions are used in CTS to define when the LCO features are required to be OPERABLE. CTS Applicabilities can be specific defined terms of reactor conditions or more general such as, "all MODES" or "any operating MODE." Generalized applicability conditions are not contained in ITS, therefore the ITS eliminates CTS requirements such as "all MODES" or "any operating MODE," replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analyses assumptions for OPERABILITY of the required features.

CTS requirements may also be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the ITS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analyses, and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is generally made to conform with NUREG-1431, and has been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will significantly increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 3
RELAXATION OF COMPLETION TIME

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the Current Technical Specifications (CTS).

Upon discovery of a failure to meet a Limiting Condition for Operation (LCO), the ITS specifies times for completing Required Actions of the associated ITS Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times (referred to as Allowed Outage Times (AOTs) in the CTS). These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a Design Basis Accident (DBA) occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated, and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing AOT. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

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The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 4 RELAXATION OF REQUIRED ACTION

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve relaxation of the Required Actions in the Current Technical Specifications (CTS).

Upon discovery of a failure to meet a Limiting Condition for Operation (LCO), the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from the ISTS is acceptable because the Required Actions take into account the OPERABILITY status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated, and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the OPERABILITY status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a Design Basis Accident (DBA) occurring during the repair period. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve deletion of Surveillance Requirements in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice, and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 6
RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify OPERABILITY under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements that when combined, include OPERABILITY verification of all components required in the LCO for the features specified in the CTS. Adopting this format preference in the ITS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes which provide exceptions to Surveillance Requirements to provide for variations that do not affect the results of the test are also included in this category. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 7
RELAXATION OF SURVEILLANCE FREQUENCY, NON-24 MONTH TYPE CHANGE

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the relaxation of Surveillance Frequencies in the Current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing Surveillance tests. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain Surveillance Frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other ITS requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing can not be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced, and reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice, or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the Surveillance Frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required that are established predictors of system performance. *Surveillance Frequency extensions can be based on NRC-approved topical reports.* The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected, and the

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing any accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 8
DELETION OF REPORTING REQUIREMENTS

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the deletion of requirements in the Current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1431, and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator of any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 9
SURVEILLANCE FREQUENCY CHANGE USING GL 91-04 GUIDELINES,
NON-24 MONTH TYPE CHANGE

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the relaxation of 7 day, 31 day, or 92 day Surveillance Frequencies for Surveillances in the Current Technical Specifications (CTS).

This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. While the Generic Letter provided guidance for extending Surveillance Frequencies from 18 months to 24 months, the guidelines in the Generic Letter provide a basis for extending other Surveillance Frequencies based on historical data. Reviews of historical maintenance and Surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Based on the inherent system and component reliability and the routine testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical Surveillance data also demonstrates that there have been no failures that would invalidate this conclusion. In addition, the proposed 184 day Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (230 days) does not invalidate any assumptions in the plant licensing basis.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves a change in the Surveillance Frequencies from 7 days, 31 days, or 92 days to 184 days. The proposed change does not physically impact the plant, and does not impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analyses. The proposed change does not impact the Surveillance Requirements themselves, and do not change the methods used for performing Surveillances. Additionally, the proposed change does not introduce any new accident initiators, because no accidents previously evaluated have as their initiators anything related to the Frequency of Surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident,

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of Surveillance test results indicates that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change involves a change in the Surveillance Frequencies from 7 days, 31 days, or 92 days to 184 days. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of Surveillance test results indicates that there is no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change involves a change in the Surveillance Frequencies from 7 days, 31 days, or 92 days to 184 days. Although the proposed change will result in an increase in the interval between Surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 10
18 TO 24 MONTH SURVEILLANCE FREQUENCY CHANGE, NON-CHANNEL CALIBRATION TYPE

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the extension of 18 month Surveillance Frequencies for non-CHANNEL CALIBRATION type Surveillances in the Current Technical Specifications (CTS) to 24 months.

This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and Surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Based on the inherent system and component reliability and the routine testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical Surveillance data also demonstrates that there have been no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves a change in the Surveillance Frequencies from 18 months to 24 months. The proposed change does not physically impact the plant, and does not impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analyses. The proposed change does not impact the Surveillance Requirements themselves, and does not change the methods used for performing the Surveillances. Additionally, the proposed change does not introduce any new accident initiators, because no accidents previously evaluated have as their initiators anything related to the Frequency of Surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident, because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of Surveillance test results indicates that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change involves a change in the Surveillance Frequencies from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of Surveillance test results indicates no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   Although the proposed change will result in an increase in the interval between Surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 11
18 TO 24 MONTH SURVEILLANCE FREQUENCY CHANGE, CHANNEL CALIBRATION TYPE

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the extension of 18 month Surveillance Frequencies for CHANNEL CALIBRATION type Surveillances in the Current Technical Specifications (CTS) to 24 months.

This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Furthermore, the impacted instrumentation has been evaluated based on make, manufacturer, and model number to determine that the actual drift of the instrumentation falls within the design allowance in the associated setpoint calculation. Based on the design of the instrumentation and the drift evaluations, it is concluded that the impact, if any, from this change on system availability is minimal. A review of the Surveillance test history was performed to validate the above conclusion. This review demonstrates that there have been no failures that would invalidate the conclusion that the impact, if any, on system availability from this change is minimal. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by ITS SR 3.0.2 (30 months), does not invalidate any assumptions in the plant licensing basis.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves a change in the instrumentation CHANNEL CALIBRATION Surveillance Frequencies from 18 months to 24 months. The proposed change does not physically impact the plant, and does not impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analyses. The proposed change does not impact the Surveillance Requirements themselves, and does not change the methods for performing the Surveillances. Additionally, the proposed change does not introduce any new accident initiators, because no accidents previously evaluated have as their initiators anything related to the Frequency of Surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident, because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of Surveillance test results indicates that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicates no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change involves a change in the instrumentation CHANNEL CALIBRATION Surveillance Frequencies from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of Surveillance test results indicates no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   Although the proposed change will result in an increase in the interval between Surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 12
DELETION OF SURVEILLANCE REQUIREMENT SHUTDOWN PERFORMANCE REQUIREMENTS

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes involve the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE periodically. The CTS requires these Surveillances to be performed with the unit in a specified condition, usually in a MODE outside the Applicability of the Limiting Condition for Operation (LCO). The ITS Surveillance does not include the restriction on unit conditions. The control of the unit conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to be unnecessary as an ITS restriction. As indicated in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, allowing this control is consistent with the vast majority of other Technical Specification Surveillances that do no dictate unit conditions for the Surveillance. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

**DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES**

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition, but does not change the method of performance. The appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures to assure the possibility of a new or different kind of accident are not created. The control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition. However, the appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures. The control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 13
ADDITION OF LCO 3.0.4 EXCEPTION

Not Used.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 14
CHANGING INSTRUMENTATION ALLOWABLE VALUES

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Rev. 2. Some of the proposed changes to the Current Technical Specifications (CTS) involve a change to the Allowable Values for Technical Specification instrumentation.

The proposed changes in selected Allowable Values for the instrumentation included in Section 3.3 of the ITS are the result of application of the AEP Instrument Setpoint Methodology (EG-IC-004, "Instrument Setpoint Uncertainty," Rev. 4). This methodology incorporates the guidance of ANSI/ISA S67.04-Part I-1994 and RP67.04-Part II-1994. Application of this methodology results in instrumentation selected Allowable Values that more accurately reflect total instrumentation loop accuracy as well as that of test equipment and setpoint drift between Surveillances.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves the change in selected Allowable Values for the instrumentation included in Section 3.3 of the ITS. The proposed changes will not result in any hardware changes. The instrumentation included in the proposed Section 3.3 of the ITS is not assumed to be an initiator of any analyzed event. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to this proposed change. As a result, the proposed change will not result in unnecessary plant transients. The role of the instrumentation included in Section 3.3 of the ITS is in mitigating and thereby limiting the consequences of accidents. The Allowable Values have been developed to ensure that the design and safety analyses limits will be satisfied. The methodology used for the development of the Allowable Values ensures that the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses, and that the results and consequences described in the safety analyses remain bounding. Additionally, the proposed change does not alter the ability of the instrumentation and associated systems and components to detect and mitigate events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

The proposed changes are the result of application of the AEP Instrumentation Setpoint Methodology, and do not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the fact that the method and manner of plant operation is unchanged. The use of the proposed Allowable Values does not impact safe operation of the plant, in that the safety analyses limits will be maintained. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). These Allowable Values were developed using a methodology to ensure the affected instrumentation and associated systems and components remain capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation, except that setpoint may be changed. Since operational methods remain unchanged, and the existing operating parameters have been evaluated to maintain the unit within existing design basis criteria, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change does not involve a reduction in a margin of safety. The proposed changes have been developed using a methodology to ensure safety analyses limits are not exceeded. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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ENVIRONMENTAL ASSESSMENT

I&M has evaluated this license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this license amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in the generic and specific Determination of No Significant Hazards Considerations, this proposed amendment does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

No new effluents or effluent release paths are created by the proposed amendment.

Therefore, pursuant to 10 CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

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SUMMARY OF CHANGES
ITS CHAPTER 1.0

Change Description	Affected Pages
A self-identified change has been made to correct a grammatical error in the definition of DOSE EQUIVALENT I-131, second sentence, by changing "page 192-212" to "pages 192-212."	Page 65 of 105.

VOLUME 3

CNP UNITS 1 AND 2
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 1.0
USE AND APPLICATION

Revision 1

LIST OF ATTACHMENTS

1. ITS Chapter 1.0

ATTACHMENT 1

ITS Chapter 1.0, Use and Application

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

1.0 USE AND APPLICATION

A.1

1

1.0 DEFINITIONS

1.1 DEFINED TERMS - NOTE:

A.1

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications. and Bases

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER (RTP)

RTP

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3304 MWt.

A.1

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1. -1 with fuel in the reactor vessel

moved from Table 1.1

A.1

INSERT 1 A.2

ACTION S ,

S

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

INSERT 2

A.1

OPERABLE - OPERABILITY

safety A.3

,

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling and or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

and when

A.1

, , and al

or A.4

specified safety A.3

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A.2

INSERT 1

, and reactor vessel head closure bolt tensioning

A.1

INSERT 2

that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times

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A.1

ITS

DEFINITIONS

~~REPORTABLE EVENT~~

~~1.7 A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.~~

~~CONTAINMENT INTEGRITY~~

~~1.8 CONTAINMENT INTEGRITY shall exist when:~~

A.5

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.1

1.8.2 All equipment hatches are closed and sealed.

See ITS 3.6.1

1.8.3 Each air lock is in compliance with the requirements of Specification 3.6.1.3.

See ITS 3.6.2

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2

See ITS 3.6.1

1.1 CHANNEL CALIBRATION

that in

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps ~~such that the entire channel is calibrated~~.

A.1

INSERT 3

A.6

means of

A.1

,

A.1

CHANNEL CHECK

, ,

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

A.1

to

A.1

A.6

INSERT 3

all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

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A.1

ITS

DEFINITIONS

1.1

OPERATIONAL (COT) COT

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions. INSERT 4 — A.1, A.7

or actual — L.1

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position. INSERT 5 — L.2, A.8

s — A.1

SHUTDOWN MARGIN (SDM)

SDM subcritical control (RCCAs)

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn. — A.1

: a. RCCA ,

INSERT 6 — A.9

LEAKAGE

IDENTIFIED LEAKAGE

a. 1.14 IDENTIFIED LEAKAGE shall be: — A.10

that from

1\. a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or — A.1

is s INSERT 7

collection systems or — A.10

2\. b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or — A.1

(RCS) (SG)

3\. c. Reactor Coolant System leakage through a steam generator to the Secondary System. ;

A.10

b. UNIDENTIFIED LEAKAGE

INSERT 8

All

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE. — A.1

; and

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A.7

INSERT 4

of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.

L.2

INSERT 5

fuel, sources, or reactivity control components,

A.9

INSERT 6

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 MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

A.10

INSERT 7

(except reactor coolant pump (RCP) seal water injection or leakoff),

A.10

INSERT 8

(except RCP seal water injection or leakoff) that

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ITS Chapter 1.0

A.1

ITS

1.1

DEFINITIONS

c.

~~PRESSURE BOUNDARY~~ LEAKAGE

A.10

SG

~~1.16~~ ~~PRESSURE BOUNDARY~~ LEAKAGE ~~shall be leakage~~ (except ~~steam generator tube~~ ~~leakage~~) through a non~~i~~solable fault in a ~~Reactor Coolant System~~ component body, pipe wall, or vessel wall.

, n RCS

A.1

~~CONTROLLED LEAKAGE~~

~~1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~

A.10

(QPTR)

QUADRANT POWER TILT RATIO

QPTR

A.1

~~1.18~~ ~~QUADRANT POWER TILT RATIO~~ shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

See ITS 3.2.4

DOSE EQUIVALENT I-131

microcuries

A.1

~~1.19~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (~~μCi~~/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 those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," ~~or in NRC~~ Regulatory Guide 1.109, Rev 1, ~~October~~ 1977.

that

INSERT 8a AEC, 1962, those listed in Table E-7 of , . NRC,

A.1

STAGGERED TEST BASIS

L.4

~~1.20~~ A STAGGERED TEST BASIS shall consist of~~:~~

INSERT 9

~~a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,~~

~~b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.~~

A.11

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L.4

INSERT 8a

, or those listed in ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

A.11

INSERT 9

the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

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ITS

A.1

DEFINITIONS

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2. A.12

(RTS)

REACTOR TRIP SYSTEM RESPONSE TIME

RTS

that

1.1 1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. INSERT 10 A.1 A.13 L.3

(ESF)

ENGINEERED SAFETY FEATURE RESPONSE TIME

ESF

1.23 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). , Times shall include diesel generator starting and sequence loading delays where applicable. INSERT 11 A.1 A.13 L.3

(AFD)

AXIAL FLUX DIFFERENCE

AFD

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector. A.1

These tests are:

, Initial Tests and Operation,

U

PHYSICS TESTS

b.

. 1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission. A.1

a. ; c. ; Nuclear Regulatory

Ē - AVERAGE DISINTEGRATION ENERGY

1.26 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant. A.1

>

SOURCE CHECK

1.27 A SOURCE CHECK shall be the qualitative assessment of Channel response when the Channel sensor is exposed to a radioactive source A.5

INSERT 10

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

A.13

L.3

INSERT 11

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

A.13

L.3

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A.1

ITS

DEFINITIONS

PROCESS CONTROL PROGRAM (PCP)

1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that 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 Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

See CTS 6.0

~~1.29 Deleted.~~

A.1

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

See ITS 5.5

~~GASEOUS RADWASTE TREATMENT SYSTEM~~

~~1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

A.5

~~VENTILATION EXHAUST TREATMENT SYSTEM~~

~~1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~

A.5

~~PURGE-PURGING~~

~~1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~

A.5

~~VENTING~~

~~1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~

A.5

ITS

A.1

1.0 DEFINITIONS

~~MEMBER(S) OF THE PUBLIC~~

~~1.35 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 its 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.~~ A.5

~~SITE BOUNDARY~~

~~1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.~~ A.5

~~UNRESTRICTED AREA~~

~~1.37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access 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 industrial, commercial, institutional and/or recreational purposes.~~ A.5

~~ALLOWABLE POWER LEVEL (APL)~~

~~1.38 ALLOWABLE POWER LEVEL (APL) is that maximum calculated power level at which power distribution limits are satisfied.~~ A.5

CORE OPERATING LIMITS REPORT (COLR)

cycle specific parameter

1.1 ~~1.39~~ The COLR 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 ~~6.9.1.11~~. Unit operation within these ~~operating~~ limits is addressed in individual ~~Specifications~~. A.1

5.6.5 parameter

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) INSERT 12 A.14

TADOT

~~1.40~~ A ~~TRIP ACTUATING DEVICE OPERATIONAL TEST~~ shall consist of operating the ~~Trip Actuating~~ ~~Device and verifying~~ OPERABILITY of ~~alarm, interlock, and/or trip functions~~. The ~~TRIP ACTUATING DEVICE OPERATIONAL TEST~~ shall include adjustment, as necessary, of the ~~Trip Actuating Device~~ ~~such~~ that it actuates at the required setpoint within the ~~required~~ accuracy. A.1

the TADOT so

necessary INSERT 13 A.14

INSERT 14 A.15

A.14

INSERT 12

all devices in the channel required for trip actuating device OPERABILITY

A.14

INSERT 13

The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

A.15

INSERT 14

ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

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A.1

ITS

Table 1.1-1

TABLE 1.1

OPERATIONAL MODES

MODE	TITLE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER* (a)	REACTOR AVERAGE COOLANT TEMPERATURE (°F)
1.	POWER OPERATION	≥ 0.99	> 5%	NA ≥ 350°F
2.	STARTUP	≥ 0.99	≤ 5%	NA ≥ 350°F
3.	HOT STANDBY	< 0.99	NA 0	≥ 350°F
4.	HOT SHUTDOWN (b)	< 0.99	NA 0	350°F > T_{avg} > 200°F
5.	COLD SHUTDOWN (b)	< 0.99	NA 0	≤ 200°F
6.	REFUELING** (c)	NA ≤ 0.95	NA 0	NA ≤ 140°F

(a) * Excluding decay heat.

(c) ** One or more Reactor vessel head closure bolts less than fully tensioned unbolted or removed and fuel in the vessel.

(b) All reactor vessel head closure bolts fully tensioned.

A.1

A.1

A.1

A.16

A.1

A.16

A.1

A.2

A.1

A.16

LA.1

A.2

Moved to definition of MODE, page 1 of 35

A.2

A.1

ITS

1.0 DEFINITIONS

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
2 Months	At least once per 62 days
SA	At least once per 184 days.
R	At least once per 549 days.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not Applicable.

A.12

Add proposed ITS Sections
1.2 - Logical Connectors
1.3 - Completion Times
1.4 - Frequency

A.17

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ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4. See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

1.1 a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. See ITS 3.1.4 If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). A.9

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5. See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5. See ITS 3.1.1

*See Special Test Exception 3.10.1. See ITS 3.1.1

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A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - TAVG LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% Delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% Delta k/k:

1.1 a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS 3.1.4

A.9

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration
6. Samarium concentration, and
7. Boron penalty.

See ITS 3.1.1

A.1

ITS

1.0 USE AND APPLICATION

A.1

1

1.0 DEFINITIONS

1.1 DEFINED TERMS NOTE:

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases A.1

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER (RTP) RTP

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3468 MWt. A.1

OPERATIONAL MODE A.1

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level , and average reactor coolant temperature specified in Table 1.1 INSERT 1 A.1 -1 with fuel in the reactor vessel moved from Table 1.1 A.2

ACTION S

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications. INSERT 2 A.1

OPERABLE - OPERABILITY , safety A.3

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that and when all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s). and A.1

, , and specified safety or A.4 A.3

A.2 INSERT 1

, and reactor vessel head closure bolt tensioning

A.1 INSERT 2

that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times

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ITS

A.1

DEFINITIONS

~~REPORTABLE EVENT~~

~~1.7 A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.~~

A.5

~~CONTAINMENT INTEGRITY~~

~~1.8 CONTAINMENT INTEGRITY shall exist when:~~

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.1

1.8.2 All equipment hatches are closed and sealed, — See ITS 3.6.1

1.8.3 Each air lock is in compliance with the requirements of Specification 3.6.1.3, — See ITS 3.6.2

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and — See ITS 3.6.1

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE. — See ITS 3.6.1

1.1 CHANNEL CALIBRATION

~~1.9~~ A CHANNEL CALIBRATION shall be that the adjustment, as necessary, of the channel output such that it responds in with the necessary range and accuracy to known values of the parameter ~~which~~ the channel monitors. The CHANNEL CALIBRATION shall ~~encompass~~ the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. INSERT 3 The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps ~~such that the entire channel is calibrated~~.

A.1 A.6 A.1

CHANNEL CHECK

~~1.10~~ A CHANNEL CHECK shall be the qualitative assessment, ~~by observation~~, of channel behavior during operation ~~by observation~~. This determination shall include, where possible, comparison of the channel indication ~~and/or~~ status ~~with~~ to other indications ~~and/or~~ status derived from independent instrument channels measuring the same parameter.

A.1

A.6 INSERT 3

all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel

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ITS Chapter 1.0

A.1

ITS

DEFINITIONS

1.1

CHANNEL FUNCTIONAL TEST — OPERATIONAL (COT)

1.11 A CHANNEL FUNCTIONAL TEST (COT) shall be:

a. Analog channels - the injection of a simulated (or actual) signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions. INSERT 4

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

L.1 A.1 A.7 A.18

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION(S) shall not preclude completion of movement of a component to a safe conservative position.

INSERT 5 L.2 A.8 A.1

SHUTDOWN MARGIN (SDM)

1.13 SHUTDOWN MARGIN (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 full length rod cluster assemblies (RCCAs) (shutdown and control) are fully inserted except for the single rod cluster assembly (RCCA) of highest reactivity worth, which is assumed to be fully withdrawn. INSERT 6

A.1 A.9

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. 1. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to collection systems or a sump or collecting tank, or INSERT 7

2. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

3. Reactor coolant system (RCS) leakage through a steam generator (SG) to the secondary system;

A.10 A.1 A.10 A.1

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A.7 INSERT 4

of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.

L.2 INSERT 5

fuel, sources, or reactivity control components,

A.9 INSERT 6

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 MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

A.10 INSERT 7

(except reactor coolant pump (RCP) seal water injection or leakoff),

A.1

ITS

DEFINITIONS

1.1 b. UNIDENTIFIED LEAKAGE A.10

INSERT 8

All

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE. A.1

; and

c. PRESSURE BOUNDARY LEAKAGE A.10

SG

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall. A.1

, n RCS

CONTROLLED LEAKAGE A.10

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

(QPTR)

QUADRANT POWER TILT RATIO A.1

QPTR

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average. See ITS 3.2.4

DOSE EQUIVALENT I-131 A.1

microcuries

that

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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 those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," or in NRC Regulatory Guide 1.109, Rev. 1, October 1977. A.1

INSERT 8a AEC, 1962, those listed in Table E-7 of , NRC,

L.4

A.10 INSERT 8

(except RCP seal water injection or leakoff) that

L.4 INSERT 8a

, or those listed in ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

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A.1

ITS

DEFINITIONS

1.1

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of: INSERT 9

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the begining of each subinterval.

A.11

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

A.12

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM (RTS) RESPONSE TIME shall be the (that) time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. INSERT 10

A.13

L.3

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

1.23 The ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. INSERT 11

A.1

A.13

L.3

AXIAL FLUX DIFFERENCE (AFD)

1.24 AXIAL FLUX DIFFERENCE (AFD) shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

A.1

A.11 INSERT 9

the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

INSERT 10

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

A.13

L.3

INSERT 11

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

A.13

L.3

A.1

ITS

DEFINITIONS

1.1 PHYSICS TESTS

These tests are: , Initial Tests and Operation, U

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

. a. b. c. ; ; Nuclear Regulatory A.1

Ē - AVERAGE DISINTEGRATION ENERGY

1.26 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant. A.1

>

~~SOURCE CHECK~~

~~1.27 A SOURCE CHECK shall be the qualitative assessment of Channel response when the Channel sensor is exposed to a radioactive source.~~ A.5

PROCESS CONTROL PROGRAM (PCP)

1.28 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that 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 Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste. See CTS 6.0

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A.1

ITS

DEFINITIONS

1/29 ~~Deleted.~~

A.1

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.7.

See ITS 5.5

~~GASEOUS RADWASTE TREATMENT SYSTEM~~

~~1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

A.5

~~VENTILATION EXHAUST TREATMENT SYSTEM~~

~~1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~

A.5

~~PURGE-PURGING~~

~~1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~

A.5

~~VENTING~~

~~1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~

A.5

A.1

ITS

1.0 DEFINITIONS

~~MEMBER(S) OF THE PUBLIC~~

~~1.35 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 its 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.~~ A.5

~~SITE BOUNDARY~~

~~1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.~~ A.5

~~UNRESTRICTED AREA~~

~~1.37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access 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 industrial, commercial, institutional and/or recreational purposes.~~ A.5

~~ALLOWABLE POWER LEVEL (APL)~~

~~1.38 ALLOWABLE POWER LEVEL (APL) is that maximum calculated power level at which power distribution limits are satisfied.~~ A.5

CORE OPERATING LIMITS REPORT (COLR)

1.1 ~~1.39~~ The COLR is the unit specific document that provides ~~core operating~~ cycle specific parameter limits for the current ~~operating~~ reload cycle. These cycle specific ~~core operating~~ parameter limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.11~~ 5.6.5. Unit operation within these ~~operating~~ limits is addressed in individual specifications. A.1

TADOT

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) INSERT 12 A.14

~~1.40~~ A ~~TRIP ACTUATING DEVICE OPERATIONAL TEST~~ TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of ~~alarm, interlock, and/or trip functions~~. The ~~TRIP ACTUATING DEVICE OPERATIONAL TEST~~ TADOT shall include adjustment, as necessary, of the trip actuating device ~~such~~ so that it actuates at the required setpoint within the ~~required~~ necessary accuracy. INSERT 13 A.1 A.14

the

INSERT 14 A.15

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A.14

INSERT 12

all devices in the channel required for trip actuating device OPERABILITY

A.14

INSERT 13

The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

A.15

INSERT 14

ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

ITS Chapter 1.0

A.1

ITS

Table 1.1-1

TABLE 1.1

OPERATIONAL MODES

MODE	TITLE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER (a)	REACTOR AVERAGE COOLANT TEMPERATURE (°F)
1.	POWER OPERATION	≥ 0.99	> 5%	≥ 350°F NA
2.	STARTUP	≥ 0.99	≤ 5%	≥ 350°F NA
3.	HOT STANDBY	< 0.99	0 NA	≥ 350°F
4.	HOT SHUTDOWN (b)	< 0.99	0 NA	350°F > T_{avg} > 200°F
5.	COLD SHUTDOWN (b)	< 0.99	0 NA	≤ 200°F
6.	REFUELING (c)	≤ 0.95 NA	0 NA	≤ 140°F NA

A.1, A.1, A.16, A.1, A.16, A.1, A.2, A.1, A.16, LA.1

(a) ~~* Excluding decay heat.~~

(c) ~~**~~ One or more Reactor vessel head ~~unbolted or removed and~~ closure bolts less than fully tensioned ~~fuel in the vessel~~.

Moved to definition of MODE, page 18 of 35

A.2

(b) All reactor vessel head closure bolts fully tensioned.

A.2

A.1

ITS

1.0 DEFINITIONS

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
2 Months	At least once per 62 days
SA	At least once per 184 days
R	At least once per 549 days
S/U	Prior to each reactor start-up
P	Completed prior to each release
N.A.	Not Applicable

A.12

Add proposed ITS Sections
1.2 - Logical Connectors
1.3 - Completion Times
1.4 - Frequency

A.17

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{AVG} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4. See ITS 3.1.1

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

1.1 a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. See ITS 3.1.4 If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). A.9

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6. See ITS 3.1.1

*See Special Test Exception 3.10.1. See ITS 3.1.1

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{AVG} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% Delta k/k. (See ITS 3.1.1)

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% Delta k/k:

1.1 a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. (See ITS 3.1.4) If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). (A.9)

b. At least once per 24 hours by consideration of the following factors: (See ITS 3.1.1)

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and
7. Boron penalty.

DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS Section 1.0 and Table 1.1, "OPERATIONAL MODES," provide a description of the MODES. ITS Section 1.1 and Table 1.1-1, "MODES," changes the CTS MODE definitions in several ways:

- The phrase "Reactor vessel head unbolted or removed" in CTS Table 1.1 Note ** is replaced with "One or more reactor vessel head closure bolts less than fully tensioned" in ITS Table 1.1-1 Note c.

  This change is acceptable because the revised phrase is consistent with the current interpretation and usage. MODE 6 is currently declared when the first vessel head closure bolt is detensioned. This change also eliminates a redundant phrase. The reactor vessel head cannot be removed unless the reactor vessel head closure bolts are unbolted. Since "reactor vessel head unbolted" is already specified in the CTS Note, including "or removed" is unnecessary.

- The CTS Table 1.1 Note ** condition "fuel in the vessel" is moved to the ITS MODE definition.

  This change is acceptable because it moves information within the Technical Specifications with no change in intent. Each MODE in the Table includes fuel in the vessel.

- ITS Table 1.1-1 contains a new Note b, which applies to MODES 4 and 5. Note b states "All reactor vessel head closure bolts fully tensioned." This Note is the opposite of CTS Note ** and ITS Table 1.1-1 Note c.

  This change is acceptable because it avoids a conflict between the definition of MODE 6 and the other MODES should RCS temperature increase above the CTS MODE 6 temperature limit while a reactor vessel head closure bolt is less than fully tensioned. This ITS Note is included only for clarity. It is consistent with the current use of MODES 4 and 5 and does not result in any technical change to the application of the MODES.

- For consistency with the Notes in ITS Table 1.1-1, the ITS definition of MODE adds "reactor vessel head closure bolt tensioning" to the list of characteristics that define a MODE. Currently, the CTS definition does not include this clarification.

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DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

This change is acceptable because the definition of MODE should be consistent with the MODE table in order to avoid confusion. This change is made only for consistency and results in no technical changes to the Technical Specifications.

These changes are designated as administrative because they clarify the application of the MODES and no technical changes to the MODE definitions are made. The clarifications are consistent with the current use and application of the MODES.

A.3 The CTS Section 1.0 definition of OPERABLE-OPERABILITY requires a system, subsystem, train, component or device to be capable of performing its "specified function(s)" and all necessary support systems to also be capable of performing their "function(s)." The ITS Section 1.1 definition of OPERABLE-OPERABILITY requires the system, subsystem, train, component, or device to be capable of performing the "specified safety function(s)," and requires all necessary support systems that are required for the system, subsystem, train, component, or device to perform its "specified safety function(s)" to also be capable of performing their related support functions. This changes the CTS by altering the requirement to be able to perform "functions" to a requirement to be able to perform "safety functions."

The purpose of the CTS and ITS definitions of OPERABLE-OPERABILITY is to ensure that the safety analysis assumptions regarding equipment and variables are valid. This change is acceptable because the intent of both the CTS and ITS definitions is to address the safety function(s) assumed in the accident analysis and not encompass other non-safety functions a system may also perform. These non-safety functions are not assumed in the safety analysis and are not needed in order to protect the public health and safety. This change is consistent with the current interpretation and use of the terms OPERABLE and OPERABILITY. This change is designated as administrative as it does not change the current use and application of the Technical Specifications.

A.4 The CTS Section 1.0 definition of OPERABLE-OPERABILITY requires that all necessary normal and emergency electrical power sources be available for the system, subsystem, train, component, or device to be OPERABLE. The ITS Section 1.1 definition of OPERABLE-OPERABILITY will replace the phrase "normal and emergency electrical power sources" with "normal or emergency electrical power sources." This changes the CTS definition of OPERABLE-OPERABILITY by allowing a device to be considered OPERABLE with either normal or emergency power available.

The OPERABILITY requirements for normal and emergency power sources are clearly addressed in CTS 3.0.5. These requirements allow only the normal or the emergency electrical power source to be OPERABLE, provided its redundant system(s), subsystem(s), train(s), component(s), and device(s) (redundant to the systems, subsystems, trains, components, and devices with an inoperable power source) are OPERABLE. This effectively changes the current "and" to an "or." The existing requirements (CTS 3.0.5) are incorporated into ITS 3.8.1 ACTIONS for when a normal (offsite) or emergency (diesel generator) power source is inoperable. Therefore, the ITS definition now uses the word "or" instead of the

DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

current word "and." In ITS 3.8.1, new times are provided to perform the determination of OPERABILITY of the redundant systems, et. al. This change is discussed in the Discussion of Changes (DOCs) for ITS 3.8.1. This change is designated administrative since the ITS definition is effectively the same as the CTS definition.

A.5 CTS Section 1.0 includes the following definitions:

- ALLOWABLE POWER LEVEL
- CONTAINMENT INTEGRITY
- GASEOUS RADWASTE TREATMENT SYSTEM
- MEMBER(S) OF THE PUBLIC
- PURGE - PURGING
- REPORTABLE EVENT
- SITE BOUNDARY
- SOURCE CHECK
- UNRESTRICTED AREA
- VENTILATION EXHAUST TREATMENT SYSTEM
- VENTING

The ITS does not use this terminology and ITS Section 1.1 does not contain these definitions.

These changes are acceptable because the terms are not used as defined terms in the ITS. Discussions of any technical changes related to the deletion of these terms are included in the DOCs for the CTS sections in which the terms are used. These changes are designated as administrative because they eliminate defined terms that are no longer used.

A.6 The CTS defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated." ITS defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps." This results in a number of changes to the CTS.

- The CTS definition states, "The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions." The ITS states, "The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY."

This change is acceptable because the statements are equivalent in that both require that all needed portions of the channel be tested. The ITS definition reflects the CTS understanding that the CHANNEL CALIBRATION includes only those portions of the channel needed to perform the safety function.

- The CTS states that the CHANNEL CALIBRATION "shall include the CHANNEL FUNCTIONAL TEST." The ITS does not include this statement.

  This change is acceptable because the eliminated CTS statement does not add any requirements. In both the CTS and the ITS, performance of a single test that fully meets the requirements of other tests can be credited for satisfying the other tests.

- The ITS adds the statement, "Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel." The purpose of a CHANNEL CALIBRATION is to adjust the channel output so that the channel responds within the necessary range and accuracy to known values of the parameters that the channel monitors.

  This change is acceptable because resistance temperature detectors and thermocouples are designed such that they have a fixed input/output response, which cannot be adjusted or changed once installed. Calibration of a channel containing an RTD or thermocouple is performed by applying the RTD or thermocouple fixed input/output relationship to the remainder of the channel, and making the necessary adjustments to the adjustable devices in the remainder of the channel to obtain the necessary output range and accuracy. Therefore, unlike other sensors, an RTD or thermocouple is not actually calibrated. The ITS CHANNEL CALIBRATION allowance for channels containing RTDs and thermocouples is consistent with the CTS calibration practices of these channels. This information is included in the ITS to avoid confusion, but does not change the current CHANNEL CALIBRATION practices for these types of channels.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A.7 CTS Section 1.0 defines CHANNEL FUNCTIONAL TEST as "the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions." ITS Section 1.1 renames the CTS definition to CHANNEL OPERATIONAL TEST (COT), and defines it as "the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps." The addition of use of an actual signal is discussed in DOC L.1. This changes the CTS by stating that the COT shall include adjustments, as necessary, of the devices in the channel so that the setpoints are within the required range and accuracy, changes the example list of devices contained in the definition, and states that the test may be performed by means of any series of sequential, overlapping, or total channel steps.

- The CTS definition states that the CHANNEL FUNCTIONAL TEST shall verify that the channel is OPERABLE "including alarm and/or trip functions." The ITS states that the COT shall verify OPERABILITY of "all devices in the channel required for channel OPERABILITY."

  This change is acceptable because the statements are equivalent in that both require that the channel be verified to be OPERABLE. The CTS and the ITS use different examples of what is included in a channel, but this does not change the intent of the requirement. The ITS use of the phrase "all devices in the channel required for channel OPERABILITY" reflects the CTS understanding that the test includes only those portions of the channel needed to perform the safety function.

- The ITS states "The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy."

  This change is acceptable because it clarifies that adjustments performed during a COT do not invalidate the test. This is consistent with the current implementation of the CHANNEL FUNCTIONAL TEST and does not result in a technical change to the Technical Specifications.

- The ITS states "The COT may be performed by means of any series of sequential, overlapping, or total channel steps."

  This change is acceptable because it states current Industry practice. This is consistent with the current implementation of the CHANNEL FUNCTIONAL TEST and does not result in a technical change to the Technical Specifications.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A.8 CTS Section 1.0 provides a definition of CORE ALTERATION. The ITS Section 1.1 definition of CORE ALTERATION revises the CTS definition to eliminate two redundant phrases.

DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

The CTS definition includes "movement or manipulation" of any component within the reactor pressure vessel. The ITS definition of CORE ALTERATION will only include "movement" of components, not "manipulation."

This change is acceptable because the eliminated phrase adds no value. In the context of this definition, any manipulation of a component will involve its movement, so stating "movement or manipulation" is redundant and potentially confusing.

- The CTS definition does not preclude completion of movement of a component to a "safe conservative" position. The ITS definition specifies only a "safe" position.

  This change is acceptable because the eliminated phrase adds no value. The Technical Specifications provide no basis for determining whether a movement is conservative, so it is assumed that the word "conservative" is used in the definition to mean "safe." Therefore, stating "safe conservative" is repetitious and potentially confusing.

  These changes are designated administrative because they represent the elimination of redundant words and phrases without changing the intent of the definition.

A.9 CTS Section 1.0 provides a definition of SHUTDOWN MARGIN (SDM). CTS 4.1.1.1.1.a and CTS 4.1.1.2.a provide an exception to the SDM definition, such that if a control rod is inoperable due to being immovable or untrippable, the SDM is modified (increased) by the worth of the inoperable rod. The ITS Section 1.1 definition of SDM contains two differences from the CTS definition.

- The CTS definition is changed to indicate that the worth of any Rod Control Cluster Assemblies (RCCAs) which are not capable of being fully inserted must be accounted for in the determination of the SDM. Currently, this requirement is not in the CTS.

  This change is acceptable because it is consistent with the existing SDM requirements in CTS 3.1.1.1 and 3.1.1.2.

- The CTS definition is clarified to include a description of the reactor fuel and moderator temperature conditions (i.e., nominal zero power level) at which the SDM is calculated when in MODE 1 or 2.

  This change is acceptable because including this information is not a technical change. SDM calculations are currently performed for nominal zero power conditions.

These changes are designated as administrative because they do not represent a technical change to the Technical Specifications.

A.10 CTS Section 1.0 provides definitions for CONTROLLED LEAKAGE, IDENTIFIED LEAKAGE, PRESSURE BOUNDARY LEAKAGE, and UNIDENTIFIED

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LEAKAGE. ITS Section 1.1 includes these requirements in one definition called LEAKAGE (which includes three categories: identified LEAKAGE, unidentified LEAKAGE, and pressure boundary LEAKAGE). This changes the CTS by incorporating the definitions into the ITS LEAKAGE definition with no technical changes. The CTS term CONTROLLED LEAKAGE, which is the seal water flow supplied to the reactor coolant pump seals, is no longer considered leakage and has its own specification titled "Seal Injection Flow" as ITS 3.5.5. Since seal injection flow is no longer considered leakage, it appears as an exception in the CTS definitions of IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE. As a result, the ITS will not contain a defined term, "CONTROLLED LEAKAGE."

This change is acceptable because it results in no technical changes to the Technical Specifications. This change is designated an administrative change in that it rearranges existing definitions, with no change in intent.

A.11 The CTS Section 1.0 definition of STAGGERED TEST BASIS states, "A STAGGERED TEST BASIS shall consist of: a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval." The ITS Section 1.1 definition states, "A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function." This changes the CTS to specify the frequency of a Surveillance on one system, subsystem, train, or other designated component in the Frequency column of the ITS instead of specifying the frequency in which all systems, subsystems, trains, or other designated components must be tested.

This change is acceptable because the testing frequency of components on a STAGGERED TEST BASIS is not changed. Unlike the CTS definition, the ITS definition allows the Surveillance interval for one subsystem to be specified in the Frequency column of the applicable Surveillance Requirements, independent of the number of subsystems. As an example, consider a three channel system tested on a STAGGERED TEST BASIS. The CTS would specify testing every three months on a STAGGERED TEST BASIS, which results in one channel being tested each month (three equal subintervals). Under the ITS definition, the Surveillance Frequency would be monthly on a STAGGERED TEST BASIS and, one channel would also be tested each month. In both the CTS and ITS definitions, all channels are tested every three months. Each test under the CTS definition would be performed at the beginning of the subinterval. Under the ITS definition, each Surveillance Frequency starts at the beginning of the CTS definition subinterval. Thus, there are no net changes in the testing interval. This change represents an editorial preference in the ITS. This change is designated as administrative as no technical changes are made to the Technical Specifications.

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A.12 CTS Section 1.0 provides a definition of FREQUENCY NOTATION and includes CTS Table 1.2, which lists these notations. The ITS will not contain this information in Section 1.1, but will state the requirements in each Surveillance.

This change is acceptable because each ITS Surveillance Requirement (SR) provides the specific frequency without relying on a notation (e.g., "31 days" versus "M"). Providing the specific frequencies in the Surveillance Requirements eliminates the need for the FREQUENCY NOTATION definition and CTS Table 1.2. Any Surveillance Frequencies altered by the elimination of the definition and table will be addressed in a DOC for the affected section. This change is designated as administrative because it does not change any SR frequencies.

A.13 CTS Section 1.0 provides definitions of ENGINEERED SAFETY FEATURE RESPONSE TIME and REACTOR TRIP SYSTEM RESPONSE TIME. ITS Section 1.1 modifies the definitions to more fully describe how the tests are performed. The ITS states that the "response time test may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured." Currently, the CTS does not describe this manner of testing.

This change is acceptable because the ITS definitions are consistent with current plant practices. Also, the definitions are consistent with the guidance provided in IEEE 338-1977, Section 6.3.4, "Response Time Verification Tests," although CNP is not committed to this standard. The results of the test are unaffected by this allowance. This change is designated as administrative as it does not result in a technical change to the response time tests.

A.14 The CTS defines TRIP ACTUATING DEVICE OPERATIONAL TEST as "A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock, and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy." ITS defines TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) as "A TADOT shall consist of operating the trip actuating device and verifying OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device such that it actuates at the required setpoint within the required accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps." This results in a number of changes to the CTS.

- The CTS definition states that the TRIP ACTUATING DEVICE OPERATIONAL TEST shall "verify OPERABILITY of alarm, interlock, and/or trip functions." The ITS states that the TADOT shall "verify the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY."

  This change is acceptable because the statements are equivalent in that both require that all needed portions of the channel to be tested. The ITS definition reflects the CTS understanding that the TRIP ACTUATING

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DEVICE OPERATIONAL TEST includes only those portions of the channel needed to perform the safety function.

- The ITS states, "The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps." Currently, the CTS does not describe this manner of testing.

  This change is acceptable because it states current Industry practice. This is consistent with the current implementation of the TADOT.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A.15 ITS Section 1.1 provides definitions of ACTUATION LOGIC TEST, MASTER RELAY TEST, and SLAVE RELAY TEST. These terms are used as defined terms in the ITS but do not appear in the CTS.

This change is acceptable because these changes do not impose any new requirements or alter existing requirements. Any technical changes due to the addition of these terms and definitions will be addressed in the DOCs for the sections of the Technical Specifications in which the terms are used. These changes are designated as administrative as they add defined terms which involve no technical change to the Technical Specifications.

A.16 CTS Table 1.1, OPERATIONAL MODES, is revised. The corresponding table in ITS Section 1.1 is Table 1.1-1, MODES. The changes to the CTS are:

- The CTS Table 1.1 minimum average reactor coolant temperature for MODES 1 and 2 is changed from ≥ 350°F to "NA" (not applicable) in ITS Table 1.1-1.

  This change is acceptable because ITS LCO 3.4.2, RCS Minimum Temperature for Criticality, provides the minimum reactor coolant temperature limits for MODES 1 and 2. Therefore, the 350°F minimum temperature does not provide any useful information in ITS Table 1.1-1, and is deleted from the CTS.

- The CTS Table 1.1 MODE 6 upper limit on average reactor coolant temperature (≤ 140°F) is removed. In ITS Table 1.1-1, the MODE 6 average reactor coolant temperature limit is specified as "NA" (not applicable).

  This change is acceptable because it eliminates a conflict in the CTS MODE Table. If the average coolant temperature exceeds the upper limit with the reactor vessel head closure bolts less than fully tensioned, the CTS Table could be misinterpreted as no MODE being applicable. This is not the intent of the CTS or ITS MODE 6 definitions. By removing the temperature reference, this ambiguity is eliminated.

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- The CTS Table 1.1 % RATED THERMAL POWER limit of 0% for MODES 3, 4, 5, and 6 is changed in ITS Table 1.1-1 to "NA" (not applicable).

  This change is acceptable because the reactivity and plant equipment limitations in MODES 3, 4, 5, and 6 do not allow power operation. Therefore, it is not necessary to have these restrictions in the MODE Table.

These changes are designated as administrative because they result in no technical changes to the Technical Specifications.

A.17 ITS Sections 1.2, 1.3, and 1.4 contain information that is not in the CTS. This change to the CTS adds explanatory information on ITS usage that is not applicable to the CTS. The added sections are:

- <u>Section 1.2 - Logical Connectors</u>

  Section 1.2 provides specific examples of the logical connectors "<u>AND</u>" and "<u>OR</u>" and the numbering sequence associated with their use.

- <u>Section 1.3 - Completion Times</u>

  Section 1.3 provides guidance on the proper use and interpretation of Completion Times. The section also provides specific examples that aid in the use and understanding of Completion Times.

- <u>Section 1.4 - Frequency</u>

  Section 1.4 provides guidance on the proper use and interpretation of Surveillance Frequencies. The section also provides specific examples that aid in the use and understanding of Surveillance Frequency.

This change is acceptable because it aids in the understanding and use of the format and presentation style of the ITS. The addition of these sections does not add or delete technical requirements, and will be discussed specifically in those Technical Specifications where application of the added sections results in a change. This change is designated as administrative because it does not result in a technical change to the Technical Specifications.

A.18 Unit 2 CTS Section 1.0 includes a CHANNEL FUNCTIONAL TEST definition for bistable channels. The definition of CHANNEL FUNCTIONAL TEST for bistable channels requires "the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions." However, this CTS definition is essentially duplicative of the TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) definition. Additionally, this part of the CHANNEL FUNCTIONAL TEST definition is not included in the Unit 1 CTS. ITS Section 1.1 does not include this definition, since the requirements for bistable channels are covered by the TADOT definition.

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This change is acceptable because the TADOT definition adequately covers bistable channels, and does not impose any new requirements or alter any existing requirements. This change is categorized as administrative because the bistable portion of the definition is duplicative of the TADOT definition.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 1.1, "OPERATIONAL MODES," states that MODE 6 is restricted to reactivity conditions with $k_{eff} \leq 0.95$. ITS Table 1.1-1, "MODES," does not contain this restriction.

This change is acceptable because the core reactivity requirements for MODE 6 are covered in ITS 3.9.1, "Boron Concentration," by requiring the boron concentration in the Reactor Coolant System to be maintained within the limits specified in the COLR. The LCO section of the 3.9.1 Bases states "The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations." Moving this detail from the MODE Table to the LCO 3.9.1 Bases eliminates the potential to misinterpret the MODE table and not apply the MODE 6 requirements if the reactor vessel head closure bolts are less than fully tensioned, fuel is in the reactor vessel, and core reactivity exceeds a k_{eff} of 0.95. ITS LCO 3.9.1 will ensure that the appropriate reactivity conditions are maintained in MODE 6, so it is not necessary to have this restriction in the MODE Table in order to provide adequate protection of the public health and safety. Once moved to the Bases, any changes to the core reactivity requirement will be controlled by the Technical Specifications Bases Control Program described in Chapter 5 of the ITS. This change is designated a less restrictive movement of detail because it moves information from the Technical Specifications to the Bases.

LESS RESTRICTIVE CHANGES

L.1 The CTS Section 1.0 definition of CHANNEL FUNCTIONAL TEST requires the use of a simulated signal when performing the test. ITS Section 1.1 renames the CTS definition to CHANNEL OPERATIONAL TEST (COT) as discussed in DOC A.7. The ITS Section 1.1 COT definition allows the use of an actual or simulated signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance if sufficient information is collected to satisfy the surveillance test requirements.

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This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because it allows an actual signal to be credited for a Surveillance where only a simulated signal was previously allowed.

L.2 The CTS Section 1.0 definition of CORE ALTERATION applies to the movement or manipulation of any component in the reactor vessel with the vessel head removed and fuel in the vessel. The ITS Section 1.1 definition of CORE ALTERATION will only apply to the movement of fuel, sources, or reactivity control components in the reactor vessel. This changes the CTS by eliminating from the definition of CORE ALTERATION the movement of any components in the reactor vessel that are not fuel, sources, or reactivity control components. The elimination of "or manipulation" from the definition is discussed in DOC A.8.

The defined term CORE ALTERATION in the ITS is used to prevent a core reactivity excursion. Other accidents which can occur during refueling conditions, such as a fuel handling accident or boron dilution accident, are addressed in the ITS by prohibitions on the movement of irradiated fuel or prohibitions on positive reactivity additions. This change is acceptable because the ITS definition of CORE ALTERATION controls the movement of components such as fuel, sources, and reactivity control components that can affect core reactivity. The CTS definition also prohibits the movement of other equipment such as cameras, thimble plugs, and core internals that have little, if any, effect on core reactivity. Therefore, controlling the movement of those items under the definition of CORE ALTERATION is not necessary. This change is designated as less restrictive because the ITS definition applies in fewer circumstances than does the CTS definition.

L.3 The CTS Section 1.0 definitions of ENGINEERED SAFETY FEATURE RESPONSE TIME and REACTOR TRIP SYSTEM RESPONSE TIME require measurement of the response time from the sensor through the actuated equipment. The ITS definitions of ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME and REACTOR TRIP SYSTEM (RTS) RESPONSE TIME are modified to state "In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC." This changes the CTS by eliminating the requirement to include all components in a response time test.

The purpose of response time testing is to ensure that the system response time, from measurement of a parameter to actuation of the appropriate device, is consistent with the assumptions in the safety analyses. WCAP-13632-P-A, Rev. 2, "Elimination of Pressure Sensor Response Time Testing Requirements," dated January 1996, justified the elimination of the pressure sensor response time testing requirements and allows the response time for selected components to be verified instead of measured. WCAP-14036-P-A, Rev. 1, "Elimination of Periodic Protection Channel Response Time Tests," provides the basis for using allocated signal processing actuation logic response times in the overall verification of the protection system channel response time. This change is acceptable because the cited Topical Reports have demonstrated that modified

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response time tests will continue to provide assurance that the systems will perform their functions as assumed in the safety analysis. In addition, the Topical Reports have been determined to be applicable to the specific components for which CNP is requesting this allowance.

WCAP-13632-P-A, Rev. 2, contains the technical basis and methodology for eliminating response time testing requirements for pressure and differential pressure sensors identified in the WCAP. The program described in the WCAP utilizes the methods contained in EPRI Report NP-7243, Rev. 1, "Investigation of Response Time Testing Requirements," for justifying elimination of response time testing surveillance requirements for certain pressure and differential pressure sensors. The EPRI report justifies the elimination of response time testing based on Failure Modes and Effects Analyses (FMEA) that show that component degradation that impacts pressure and differential pressure sensor response time can be detected in other routine tests such as calibration tests. The report concludes that response time testing of pressure and differential pressure sensors is redundant to other surveillance requirements such as sensor calibrations. The EPRI report only applied to those specific sensors included in the FMEA.

To address other sensors installed in Westinghouse designed plants, the WCAP contains a similarity analysis to sensors in the EPRI report or a specific FMEA to provide justification for elimination of response time testing requirements for those other sensors. Each pressure and differential pressure sensor that is identified as a candidate for elimination of periodic response time testing requirements is listed in Table 9-1 of the WCAP.

WCAP-13632-P-A, Rev. 2, has been reviewed and evaluated against the actual RTS and Engineered Safety Feature Actuation System (ESFAS) pressure and differential pressure sensors used at CNP to determine applicability. Sensors for the following RTS Functions (as shown in ITS Table 3.3.1-1) have been confirmed to be specifically addressed by WCAP-13632-P-A, and are proposed to have their response times optionally verified in lieu of measurement using the WCAP-13632-P-A methodology:

RTS Function (ITS Table 3.3.1-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
6. Overtemperature ΔT (Pressurizer Pressure Input)	1-PT-455, 457, 458 2-PT-455, 457, 458	Foxboro N-E11GM-HIE2-AL
6. Overtemperature ΔT (Pressurizer Pressure Input)	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2
8.a. Pressurizer Pressure – Low	1-PT-455, 457, 458 2-PT-455, 457, 458	Foxboro N-E11GM-HIE2-AL
8.a. Pressurizer Pressure – Low	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2
8.b. Pressurizer Pressure – High	1-PT-455, 457, 458 2-PT-455, 457, 458	Foxboro N-E11GM-HIE2-AL

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RTS Function (ITS Table 3.3.1-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
8.b. Pressurizer Pressure – High	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2
9. Pressurizer Water Level – High	1-LT-459 2-LT-459	Foxboro N-E13DH-HIH2-AL
9. Pressurizer Water Level – High	1-LT-460, 461 2-LT-460, 461	Foxboro N-E13DH-HIH2
10. Reactor Coolant Flow - Low	1-FT-414, 415, 416 1-FT-424, 425, 426 1-FT-434, 435, 436 1-FT-444, 445, 446 2-FT-414, 415, 416 2-FT-424, 425, 426 2-FT-434, 435, 436 2-FT-444, 445, 446	Foxboro E13DH
14. Steam Generator (SG) Water Level – Low Low	1-LT-517, 519 1-LT-527, 528, 529 1-LT-537, 538, 539 1-LT-547, 548, 549 2-LT-517, 518, 519 2-LT-529 2-LT-538, 539 2-LT-547, 548, 549	Foxboro N-E13DM-H1M2-BL
14. Steam Generator (SG) Water Level – Low Low	1-LT-518	Foxboro N-E13DM-H1M2-AL
14. Steam Generator (SG) Water Level – Low Low	2-LT-527, 528 2-LT-537	Foxboro N-E13DM-H1M2

Sensors for the following ESFAS Functions (as shown in ITS Table 3.3.2-1) have been confirmed to be specifically addressed by WCAP-13632-P-A, and are proposed to have their response times optionally verified in lieu of measurement using the WCAP-13632-P-A methodology:

ESFAS Function (ITS Table 3.3.2-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
1.c. Safety Injection, Containment Pressure - High	1-PT-934, 935, 936 2-PT-934, 935, 936	Foxboro E11GM-HSAA1
1.d. Safety Injection, Pressurizer Pressure - Low	1-PT-455, 457 2-PT-455, 457	Foxboro N-E11GM-HIE2-AL
1.d. Safety Injection, Pressurizer Pressure - Low	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2

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ESFAS Function (ITS Table 3.3.2-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
1.e.(1)Safety Injection, Steam Line Pressure - Low	1-PT-514, 525, 536, 546 2-PT-514, 525, 536, 546	Foxboro N-E11GM-HIE2-B
2.c. Containment Spray, Containment Pressure – High High	1-PT-934, 935, 936, 937 2-PT-934, 935, 936, 937	Foxboro E11GM-HSAA1
3.a.(3)Containment Isolation, Phase A, SI Input from ESFAS, Containment Pressure - High	1-PT-934, 935, 936 2-PT-934, 935, 936	Foxboro E11GM-HSAA1
3.a.(3)Containment Isolation, Phase A, SI Input from ESFAS, Pressurizer Pressure - Low	1-PT-455, 457 2-PT-455, 457	Foxboro N-E11GM-HIE2-AL
3.a.(3)Containment Isolation, Phase A, SI Input from ESFAS, Pressurizer Pressure - Low	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2
3.a.(3)Containment Isolation, Phase A, SI Input from ESFAS, Steam Line Pressure - Low	1-PT-514, 525, 536, 546 2-PT-514, 525, 536, 546	Foxboro N-E11GM-HIE2-B
3.b.(3)Containment Isolation, Phase B, Containment Pressure – High High	1-PT-934, 935, 936, 937 2-PT-934, 935, 936, 937	Foxboro E11GM-HSAA1
4.c. Steam Line Isolation, Containment Pressure – High High	1-PT-934, 935, 936, 937 2-PT-934, 935, 936, 937	Foxboro E11GM-HSAA1
4.d. Steam Line Isolation, Steam Line Pressure - Low	1-PT-514, 525, 536, 546 2-PT-514, 525, 536, 546	Foxboro N-E11GM-HIE2-B

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ESFAS Function (ITS Table 3.3.2-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
5.b. Turbine Trip and Feedwater Isolation, Steam Generator (SG) Water Level – High High	1-LT-517, 519 1-LT-527, 528, 529 1-LT-537, 538, 539 1-LT-547, 548, 549 2-LT-517, 518, 519 2-LT-529 2-LT-538, 539 2-LT-547, 548, 549	Foxboro N-E13DM-H1M2-BL
5.b. Turbine Trip and Feedwater Isolation, Steam Generator (SG) Water Level – High High	1-LT-518	Foxboro N-E13DM-H1M2-AL
5.b. Turbine Trip and Feedwater Isolation, Steam Generator (SG) Water Level – High High	2-LT-527, 528 2-LT-537	Foxboro N-E13DM-H1M2
5.c. Turbine Trip and Feedwater Isolation, SI Input from ESFAS, Containment Pressure - High	1-PT-934, 935, 936 2-PT-934, 935, 936	Foxboro E11GM-HSAA1
5.c. Turbine Trip and Feedwater Isolation, SI Input from ESFAS, Pressurizer Pressure - Low	1-PT-455, 457 2-PT-455, 457	Foxboro N-E11GM-HIE2-AL
5.c. Turbine Trip and Feedwater Isolation, SI Input from ESFAS, Pressurizer Pressure - Low	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2
5.c. Turbine Trip and Feedwater Isolation, SI Input from ESFAS, Steam Line Pressure - Low	1-PT-514, 525, 536, 546 2-PT-514, 525, 536, 546	Foxboro N-E11GM-HIE2-B
6.c. Auxiliary Feedwater, Steam Generator (SG) Water Level – Low Low	1-LT-517, 519 1-LT-527, 528, 529 1-LT-537, 538, 539 1-LT-547, 548, 549 2-LT-517, 518, 519 2-LT-529 2-LT-538, 539 2-LT-547, 548, 549	Foxboro N-E13DM-H1M2-BL
6.c. Auxiliary Feedwater, Steam Generator (SG) Water Level – Low Low	1-LT-518	Foxboro N-E13DM-H1M2-AL

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ESFAS Function (ITS Table 3.3.2-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
6.c. Auxiliary Feedwater, Steam Generator (SG) Water Level – Low Low	2-LT-527, 528 2-LT-537	Foxboro N-E13DM-H1M2
6.d. Auxiliary Feedwater, SI Input from ESFAS, Containment Pressure - High	1-PT-934, 935, 936 2-PT-934, 935, 936	Foxboro E11GM-HSAA1
6.d. Auxiliary Feedwater, SI Input from ESFAS, Pressurizer Pressure - Low	1-PT-455, 457 2-PT-455, 457	Foxboro N-E11GM-HIE2-AL
6.d. Auxiliary Feedwater, SI Input from ESFAS, Pressurizer Pressure - Low	1-PT-456 2-PT-456	Foxboro N-E11GM-HIE2
6.d. Auxiliary Feedwater, SI Input from ESFAS, Steam Line Pressure - Low	1-PT-514, 525, 536, 546 2-PT-514, 525, 536, 546	Foxboro N-E11GM-HIE2-B
7.c. Containment Air Recirculation/Hydrogen Skimmer (CEQ) System, Containment Pressure - High	1-PT-934, 935, 936 2-PT-934, 935, 936	Foxboro E11GM-HSAA1

The response time to be allocated in place of response times obtained through actual measurement during the period of verification may be obtained according to the methodology described in WCAP-13632-P-A, Rev. 2. As described in the Bases for ITS SR 3.3.1.19 (RTS RESPONSE TIME Surveillance) and ITS SR 3.3.2.13 (ESFAS RESPONSE TIME Surveillance), these verified response times may be chosen from historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests); in place, onsite, or offsite (e.g., vendor) test measurements; or utilizing vendor engineering specifications.

The NRC Safety Evaluation Report (SER) for WCAP-13632-P-A, Rev. 2, requires confirmation by the licensee that the generic analysis in the WCAP is applicable to their plant, and that the licensee commit to the following actions:

a. Perform a hydraulic response time test prior to installation of a new transmitter/switch or following refurbishment of the transmitter/switch (e.g., sensor cell or variable damping components) to determine an initial sensor-specific response time value.

b. For transmitter and switches that use capillary tubes, perform a response time test after initial installation and after any maintenance or modification activity that could damage the capillary tubes.

c. If variable damping is used, implement a method to assure that the potentiometer is at the required setting and cannot be inadvertently changed, or perform a hydraulic response time test of the sensor following each calibration.

d. Perform periodic drift monitoring of all Model 1151, 1152, 1153, and 1154 Rosemount pressure and differential pressure transmitters, for which response time testing elimination is proposed, in accordance with the guidance contained in Rosemount Technical Bulletin No. 4 and continue to remain in full compliance with any prior commitments to Bulletin 90-01, Supplement 1, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount." As an alternative to performing periodic drift monitoring of Rosemount transmitters, licensees may complete the following actions: (a) ensure that operators and technicians are aware of the Rosemount transmitter loss of fill-oil issue and make provisions to ensure that technicians monitor for sensor response time degradation during the performance of calibrations and functional tests of these transmitters; and (b) review and revise surveillance testing procedures, if necessary, to ensure that calibrations are being performed using equipment designed to provide a step function or fast ramp in the process variable and that calibrations and functional tests are being performed in a manner that allows simultaneous monitoring of both the input and output response of the transmitter under test, thus allowing, with reasonable assurance, the recognition of significant response time degradation.

To comply with the requirements of the WCAP-13632-P-A, Rev. 2, SER, CNP commits to the following:

a. The applicable plant procedures will include requirements that stipulate that pressure and differential pressure sensor response times must be verified by performance of an appropriate response time test prior to placing a sensor into operational service, and re-verified following maintenance that may adversely affect sensor response time.

b. The applicable plant procedures, and/or other appropriate administrative controls, will include requirements that stipulate that pressure and differential pressure sensors (transmitters and switches) utilizing capillary tubes (e.g., containment pressure), shall be subjected to response time testing after initial installation and following any maintenance or modification activity that could damage the transmitter capillary tubes. The only transmitters that use capillary tubes at CNP, and are being proposed to have their response times optionally verified in lieu of measurement using the WCAP-13632-P-A methodology, are shown in the table below:

RTS Function (ITS Table 3.3.1-1)	Unit 1 and Unit 2 Instruments	Manufacturer and Model Number
9. Pressurizer Water Level – High	1-LT-459 2-LT-459	Foxboro N-E13DH-HIH2-AL
9. Pressurizer Water Level – High	1-LT-460, 461 2-LT-460, 461	Foxboro N-E13DH-HIH2

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These transmitters for Pressurizer Water Level have filled capillary lines for the reference side of the instrument.

c. CNP has no pressure or differential pressure transmitters with variable damping installed in any RTS or ESFAS application that are being proposed to have their response times optionally verified in lieu of measurement using the WCAP-13632-P-A methodology. However, modifications may be performed in the future to install transmitters with variable damping capability for one or more of the applicable pressure or differential pressure sensors. If this occurs, then the applicable plant procedures, and/or other appropriate administrative controls, will be developed or revised to implement a method to assure that the potentiometer is at the required setting and cannot be inadvertently changed, or that a hydraulic response time test of the sensor is performed following each calibration.

d. I&M responded to NRC Bulletins 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," on May 24, 1990, and its supplement (Supplement 1) on March 1, 1993. In these responses, I&M specified that there were no Rosemount transmitters installed in safety-related systems at CNP, and the NRC determined that this confirmation provided an adequate basis to consider NRC's review of the I&M response complete as documented in letters dated December 11, 1990, and April 16, 1993, respectively. No further reviews have been conducted by the NRC regarding the concerns identified in NRC Bulletin 90-01, including Supplement 1, and the concerns identified have been resolved for CNP. In addition, there are still no Rosemount transmitters installed in safety-related systems at CNP. However, periodic technician training is conducted that addresses awareness of this issue, and technicians are trained to monitor for sluggish response of pressure and differential pressure sensors during maintenance and testing activities. Based on the current status of this issue, no further actions are required.

Based on this evaluation, the change to eliminate response time testing requirements for the specific pressure and differential pressure sensors identified in the two tables above is acceptable because the analysis presented in WCAP-13632-P-A, Rev. 2, has been determined to be applicable to CNP, and I&M has committed to the additional actions required by the NRC SER approving this Topical Report.

WCAP-14036-P-A, Rev. 1, contains the technical basis and methodology for eliminating response time testing requirements for signal processing and actuation logic components of the RTS and ESFAS protection channels identified in the WCAP. The justification for this elimination is based on a Failure Modes and Effects Analysis (FMEA) that either determined that individual component degradation had no response time impact; or identified components that may contribute to RTS or ESFAS response time degradation. Where potential response time impact was identified, testing was conducted to determine the magnitude of the response time degradation, or a bounding response time limit for the system or component was identified. As described in the Bases for ITS SR 3.3.1.19 and ITS SR 3.3.2.13, the allocations for sensor, signal conditioning,

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and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance work that may adversely affect response time. For the identified signal processing and actuation logic components, bounding response time allocation will be derived from design response time specifications for the component.

The NRC Safety Evaluation Report (SER) for WCAP-14036-P-A, Rev. 1, requires confirmation by the licensee that the FMEA in the WCAP is applicable to the equipment actually installed in the facility, and that the analysis is valid for the versions of the boards used in the facility protection system.

WCAP-14036-P-A has been reviewed and evaluated against the actual RTS and ESFAS signal processing and actuation logic used at CNP to determine applicability. At CNP, signal processing of most of the RTS and ESFAS sensor inputs is performed using Foxboro Spec 200 and Foxboro Spec 200μ signal conditioning racks. This signal processing equipment is not included in the specific equipment evaluated in the WCAP. Therefore, I&M will continue to measure the response time of this equipment instead of using allocated response times.

For neutron flux RTS protection channels, signal processing is performed by the Westinghouse Nuclear Instrumentation System (NIS), and the Westinghouse Solid State Protection System (SSPS) is used for the protection channel actuation logic. Neutron detectors are exempted from response time testing as shown in proposed ITS SR 3.3.1.19. For the other RTS and ESFAS protection channels using either Foxboro Spec 200 or Foxboro Spec 200μ signal processing, and for the reactor coolant pump undervoltage and underfrequency RTS and ESFAS protection channels, the Westinghouse SSPS is used for the protection channel actuation logic. Sections 4.6 and 4.8 of WCAP-14036-P-A describe the results of the FMEA for the NIS and SSPS used at CNP, respectively, and I&M has verified that the FMEA is applicable to the NIS and SSPS equipment actually installed at CNP. As described in WCAP-14036-P-A, the FMEA alone was used for the NIS to establish response time degradation limits that are not detectable by other periodic surveillance tests. For the SSPS, response time degradation limits are based on the response time of relays, since the relays are the limiting response time component in this system. In both cases, testing was not required to determine the magnitude of the response time degradation. Therefore, the results of the NIS FMEA and evaluation of SSPS relay response times in the WCAP, and confirmation that the specific equipment used at CNP is addressed by these evaluations in the WCAP, demonstrate the acceptability of eliminating response time testing requirements for components of these two systems.

Signal processing components and actuation logic components for the following RTS Functions (as shown in ITS Table 3.3.1-1) have been confirmed to be specifically addressed by WCAP-14036-P-A, and are proposed to have their response times optionally verified in lieu of measurement using the WCAP-14036-P-A methodology:

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ITS CHAPTER 1.0, USE AND APPLICATION

RTS Function (ITS Table 3.3.1-1)		Signal Processing System	Actuation Logic System
2.a.	Power Range Neutron Flux - High	Westinghouse NIS	Westinghouse SSPS
2.b.	Power Range Neutron Flux - Low	Westinghouse NIS	Westinghouse SSPS
6.	Overtemperature ΔT	Note(1)	Westinghouse SSPS
7.	Overpower ΔT	Note(1)	Westinghouse SSPS
8.a.	Pressurizer Pressure - Low	Note(1)	Westinghouse SSPS
8.b.	Pressurizer Pressure - High	Note(1)	Westinghouse SSPS
10.	Reactor Coolant Flow - Low	Note(1)	Westinghouse SSPS
12.	Undervoltage RCPs	Note(1)	Westinghouse SSPS
13.	Underfrequency RCPs	Note(1)	Westinghouse SSPS
14.	Steam Generator (SG) Water Level – Low Low	Note(1)	Westinghouse SSPS
17.	Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	Note(1)	Westinghouse SSPS

(1) RTS RESPONSE TIME will continue to be measured.

Signal processing components and actuation logic components for the following ESFAS Functions (as shown in ITS Table 3.3.2-1) have been confirmed to be specifically addressed by WCAP-14036-P-A, and are proposed to have their response times optionally verified in lieu of measurement using the WCAP-14036-P-A methodology:

ESFAS Function (ITS Table 3.3.2-1)		Signal Processing System	Actuation Logic System
1.c.	Safety Injection, Containment Pressure - High	Note(2)	Westinghouse SSPS
1.d.	Safety Injection, Pressurizer Pressure - Low	Note(2)	Westinghouse SSPS

DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

ESFAS Function (ITS Table 3.3.2-1)	Signal Processing System	Actuation Logic System
1.e.(1)Safety Injection, Steam Line Pressure - Low	Note(2)	Westinghouse SSPS
2.c. Containment Spray, Containment Pressure – High High	Note(2)	Westinghouse SSPS
4.c. Steam Line Isolation, Containment Pressure – High High	Note(2)	Westinghouse SSPS
4.d. Steam Line Isolation, Steam Line Pressure - Low	Note(2)	Westinghouse SSPS
5.b. Turbine Trip and Feedwater Isolation, Steam Generator (SG) Water Level – High High	Note(2)	Westinghouse SSPS
6.c. Auxiliary Feedwater, Steam Generator (SG) Water Level – Low Low	Note(2)	Westinghouse SSPS
6.f. Auxiliary Feedwater, Undervoltage Reactor Coolant Pump	Note(2)	Westinghouse SSPS
7.c. Containment Air Recirculation/Hydrogen Skimmer (CEQ) System, Containment Pressure - High	Note(2)	Westinghouse SSPS

(2) ESFAS RESPONSE TIME will continue to be measured.

The response time to be allocated in place of response times obtained through actual measurement during the period of verification may be obtained according to the methodology described in WCAP-14036-P-A, Rev. 1, as described in the Bases for ITS SR 3.3.1.19 and ITS SR 3.3.2.13.

Based on this evaluation, the change to eliminate response time testing requirements for the specific signal processing and actuation logic components of the RTS and ESFAS protection channels described above is acceptable because the analysis presented in WCAP-14036-P-A, Rev. 1, has been determined to be applicable to CNP, as required to be confirmed by the NRC SER approving this Topical Report.

This change is designated as less restrictive because some components which must be response time tested under the CTS will not require response time testing under the ITS.

L.4 The CTS Section 1.0 definition of DOSE EQUIVALENT I-131 requires that the DOSE EQUIVALENT I-131 be calculated using either the thyroid dose conversion factors found in Table III of TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Regulatory Guide

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ITS CHAPTER 1.0, USE AND APPLICATION

(RG) 1.109, Rev. 1 (Table E-7). The ITS allows DOSE EQUIVALENT I-131 to be calculated using any one of three thyroid dose conversion factors: TID-14844 (1962); Table E-7 of RG 1.109, Rev. 1 (1977); or ICRP 30, Supplement to Part 1, page 192-212, Table Titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity." This changes the CTS by allowing a third method, ICRP 30, Supplement to Part 1, to be used to calculate DOSE EQUIVALENT I-131.

The purpose of the defined term is to provide acceptable methods for computing DOSE EQUIVALENT I-131. Using thyroid dose conversion factors other than those given in TID-14844 results in lower doses and higher allowable activity but is justified by the discussion given in the Federal Register (FR page 23360 VI 56 No 98 May 21, 1991). This discussion accompanied the final rulemaking on 10 CFR 20 by the NRC. In that discussion, the NRC stated that they were incorporating modifications to existing concepts and recommendations of the ICRP and NCRP into NRC regulations. Incorporation of the methodology of ICRP 30 into the 10 CFR 20 revision was specifically mentioned with the explanation that changes being made result from changes in the scientific techniques and parameters used in calculating dose. In a response to a specific question as to whether or not the ICRP 30 dose parameters should be used, the NRC stated "Appropriate parameters for calculating organ doses can be found in ICRP 30 and its supplements..." Lastly, Commissioner Curtis provided additional views of the revised 10 CFR 20 with respect to the backfit rule. In that discussion, he stated that the AEC, when they issued the original 10 CFR 20, had emphasized that the standards were subject to change with the development of new knowledge and experience. He went on to say that the limits given in the revised 10 CFR 20 were based on up-to-date metabolic models and dose factors. This Federal Register entry shows clearly that, in general, the NRC was updating 10 CFR 20 to incorporate ICRP-30 recommendations and data. Given this discussion, it is concluded that using ICRP thyroid dose conversion factors to calculate DOSE EQUIVALENT I-131 is acceptable. In addition, RG 1.109 was developed by the NRC for the purpose of evaluating compliance with 10 CFR 50, Appendix I. The RG 1.109 thyroid dose conversion factors are higher than the ICRP 30 thyroid dose conversion factors for all five iodine isotopes in question. Therefore, using RG 1.109 thyroid dose conversion factors to calculate DOSE EQUIVALENT I-131 is more conservative than ICRP 30 and is therefore acceptable.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

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1.0 USE AND APPLICATION

1.1 Definitions

Section 1.0

- NOTE -

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

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Section 1.0

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Section 1.0

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
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 5.6.5. ~~Plant~~ Unit (5) operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that 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 those listed in ~~[~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," ~~or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in~~ Target Organs or Tissues per Intake of Unit Activity"~~]~~ (1) (4) those listed in (4) (5) (1) (4)
Ē - AVERAGE DISINTEGRATION ENERGY	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > ~~[~~15~~]~~ minutes, making up at least 95% of the total noniodine activity in the coolant. (1)

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Section 1.0

1.1 Definitions

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and ⑥

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c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

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1.1 Definitions

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MASTER RELAY TEST — A MASTER RELAY TEST shall consist of energizing ~~each required~~ all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps. (4)

Section 1.0

MODE — A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY — A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS — PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter ~~14~~ (3), Initial Test~~Program~~s and Operation (1) of the (U) FSAR; (5) (6)

b. Authorized under the provisions of 10 CFR 50.59; or (6)

c. Otherwise approved by the Nuclear Regulatory Commission.

~~**PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)** — The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the LTOP arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating~~ (2)

<TSTF-419 not shown>

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1.1 Definitions

~~PRESSURE AND TEMPERATURE LIMITS REPORT (continued)~~	~~limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~ ⟨TSTF-419 Not shown⟩ 2
Section 1.0 QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[3411]~~ 2893 MWt for Unit 1 and 3468 MWt for Unit 2. 1
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
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, 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 3 6 b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the ~~[~~nominal zero power design level~~]~~. 1

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1.1 Definitions

DOC A.15	SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing ~~each required~~ all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.	4
Section 1.0	STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.	
	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	
	TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.	

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Table 1.1

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)	
1	Power Operation	≥ 0.99	> 5	NA	
2	Startup	≥ 0.99	≤ 5	NA	
3	Hot Standby	< 0.99	NA	≥ ~~[~~350~~]~~	1
4	Hot Shutdown[(b)]	< 0.99	NA	~~[~~350~~]~~ > T_{avg} > ~~[~~200~~]~~	1
5	Cold Shutdown[(b)]	< 0.99	NA	≤ ~~[~~200~~]~~	1
6	Refueling[(c)]	NA	NA	NA	

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

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1.0 USE AND APPLICATION

1.2 Logical Connectors

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PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

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1.2 Logical Connectors

EXAMPLES (continued)

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EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

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1.2 Logical Connectors

EXAMPLES (continued)

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EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

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1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability, and ; (6)

CTS

1.3 Completion Times

DESCRIPTION (continued)

DOC A.17

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours ~~or~~ ; 6

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

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1.3 Completion Times

DOC A.17.

EXAMPLES (continued)

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

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1.3 Completion Times

EXAMPLES (continued)

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EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

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1.3 Completion Times

EXAMPLES (continued)

DOC A.17

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

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CTS

1.3 Completion Times

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

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CTS

1.3 Completion Times

DOC A.17

EXAMPLES (continued)

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

CTS

1.3 Completion Times

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

CTS

1.3 Completion Times

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.3-5

ACTIONS

- NOTE -
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

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1.3 Completion Times

EXAMPLES (continued)

DOC A.17

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

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CTS

1.3 Completion Times

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

CTS

1.3 Completion Times

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

CTS

1.3 Completion Times

DOC A.17

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

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CTS

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.

JOC A.17

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. 4

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. 4

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction. 4

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied: 4

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CTS

DESCRIPTION (continued)

Doc A.17

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; (5) ⑥

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed. (t) (4)

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discusses these special situations. (4)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

CTS

1.4 Frequency

DOC A.17

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

CTS

1.4 Frequency

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

CTS

1.4 Frequency

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
- NOTE - Not required to be performed until 12 hours after ≥ 25% RTP.	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

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Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

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CTS

1.4 Frequency

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
- NOTE - Only required to be met in MODE 1.	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were was (4) not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were was (4) not met), SR 3.0.4 would require satisfying the SR.

CTS

1.4 Frequency

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
- NOTE - Only required to be performed in MODE 1.	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, (3) or 3 (the assumed Applicability of the associated LCO) between performances. (4)

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance ~~were~~ was not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1. (4)

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance ~~were~~ was not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply. (4)

CTS

1.4 Frequency

EXAMPLES (continued)

DOC A.17

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
- NOTE - Not required to be met in MODE 3.	
Verify parameter is within limits.	24 hours

Example 1.4-~~6~~ specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES ~~1,~~ 2 and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance ~~were~~ was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency ~~were~~ was not met), SR 3.0.4 would require satisfying the SR.

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JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

1. The brackets are removed and the proper plant specific information/value is provided.

2. CNP does not propose to use a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) and will not relocate the Pressure and Temperature limits from the Technical Specifications. The current limits will be retained in the ITS. Therefore, the definition of PTLR was not incorporated in the ITS.

3. The ISTS SHUTDOWN MARGIN definition includes an exception to not assume a stuck rod if all rods can be verified inserted by two independent means. The CNP plant design does not provide two independent means to verify a rod is fully inserted. Therefore, the allowance cannot be used and is removed to avoid confusion.

4. Typographical/grammatical error corrected.

5. The proper plant specific information/nomenclature/value is provided.

6. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.1

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

The CTS Section 1.0 definition of CHANNEL FUNCTIONAL TEST requires the use of a simulated signal when performing the test. ITS Section 1.1 renames the CTS definition to CHANNEL OPERATIONAL TEST (COT) as discussed in DOC A.7. The ITS Section 1.1 COT definition allows the use of an actual or simulated signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance if sufficient information is collected to satisfy the surveillance test requirements.

This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because it allows an actual signal to be credited for a Surveillance where only a simulated signal was previously allowed.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the COT. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal, and the proposed requirement does not change the technical content or validity of the test. This change will not affect the probability of an accident. The source of the signal sent to components during a Surveillance is not assumed to be an initiator of any analyzed event. The consequence of an accident is not affected by this change. The results of the testing, and, therefore, the likelihood of discovering an inoperable component, are unaffected. As a result, the assurance that equipment will be available to mitigate the consequences of an accident is unaffected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the COT. This change will not physically alter the plant (no new or different type of equipment will be installed). The change also does not require any new or revised operator actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the COT. The margin of safety is not affected by this change. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal. As a result, the proposed requirement does not change the technical content or validity of the test. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.2

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

The CTS Section 1.0 definition of CORE ALTERATION applies to the movement or manipulation of any component in the reactor vessel with the vessel head removed and fuel in the vessel. The ITS Section 1.1 definition of CORE ALTERATION will only apply to the movement of fuel, sources, or reactivity control components in the reactor vessel. This changes the CTS by eliminating from the definition of CORE ALTERATION the movement of any components in the reactor vessel that are not fuel, sources, or reactivity control components. The elimination of "or manipulation" from the definition is discussed in DOC A.8.

The defined term CORE ALTERATION in the ITS is used to prevent a core reactivity excursion. Other accidents which can occur during refueling conditions, such as a fuel handling accident or boron dilution accident, are addressed in the ITS by prohibitions on the movement of irradiated fuel or prohibitions on positive reactivity additions. This change is acceptable because the ITS definition of CORE ALTERATION controls the movement of components such as fuel, sources, and reactivity control components that can affect core reactivity. The CTS definition also prohibits the movement of other equipment such as cameras, thimble plugs, and core internals that have little, if any, effect on core reactivity. Therefore, controlling the movement of those items under the definition of CORE ALTERATION is not necessary. This change is designated as less restrictive because the ITS definition applies in fewer circumstances than does the CTS definition.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change revises the definition of CORE ALTERATION to be the movement of fuel, sources, or reactivity control components within the reactor vessel rather than the movement of any component within the reactor vessel. This change will not affect the probability of an accident. The only component within the reactor vessel assumed to be an initiator of an event previously evaluated is an irradiated fuel assembly when it is dropped. None of the other components are initiators of any analyzed event. As fuel is retained in the list of components which, when moved, constitute a CORE ALTERATION, the probability of a fuel handling accident is not affected. Also, this change has no effect on the probability of a boron dilution event because a boron dilution event is not initiated by movement of components in the reactor vessel. The consequences of an accident are not affected by this change as movement of the components being excluded from the definition of CORE ALTERATION do not act to mitigate the consequences of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change revises the definition of CORE ALTERATION to be the movement of fuel, sources, or reactivity control components within the reactor vessel rather than the movement of any component within the reactor vessel. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change revises the definition of CORE ALTERATION to be the movement of fuel, sources, or reactivity control components within the reactor vessel rather than the movement of any component within the reactor vessel. The margin of safety is not affected by this change because the safety analysis assumptions are not affected. The safety analyses do not address the movement of components within the reactor vessel other than fuel, sources, and reactivity control components. Fuel continues to be included in the CORE ALTERATION definition. Also, the shutdown margin is unaffected by the movement of components other than fuel, sources, and reactivity control components because the movement of other components will not significantly change core reactivity. No change is being proposed in the application of the definition to the movement of components which are factors in the design basis analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.3

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

The CTS Section 1.0 definitions of ENGINEERED SAFETY FEATURE RESPONSE TIME and REACTOR TRIP SYSTEM RESPONSE TIME require measurement of the response time from the sensor through the actuated equipment. The ITS definitions of ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME and REACTOR TRIP SYSTEM (RTS) RESPONSE TIME are modified to state "In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC." This changes the CTS by eliminating the requirement to include all components in a response time test.

The purpose of response time testing is to ensure that the system response time, from measurement of a parameter to actuation of the appropriate device, is consistent with the assumptions in the safety analyses. WCAP-13632-P-A, Rev. 2, "Elimination of Pressure Sensor Response Time Testing Requirements," dated January 1996, justified the elimination of the pressure sensor response time testing requirements and allows the response time for selected components to be verified instead of measured. WCAP-14036-P-A, Rev. 1, "Elimination of Periodic Protection Channel Response Time Tests," provides the basis for using allocated signal processing actuation logic response times in the overall verification of the protection system channel response time. This change is acceptable because the cited Topical Reports have demonstrated that modified response time tests will continue to provide assurance that the systems will perform their functions as assumed in the safety analysis. In addition, the Topical Reports have been determined to be applicable to the specific components for which CNP is requesting this allowance, as described in the Discussion of Change. This change is designated as less restrictive because some components which must be response time tested under the CTS will not require response time testing under the ITS.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change allows some devices to be assigned an allocated response time, instead of a measured response time, when performing response time testing of the RTS and ESFAS protection channels. This change does not alter the design, material, and construction standards that were applicable prior to the change. The same RTS and ESFAS instrumentation is being used, and the time response allocations and modeling assumption in the safety and accident analyses as described in Chapter 14 of the CNP Updated Final Safety Analysis Report (UFSAR) remain the same, with only the method of verifying time response changed. The proposed change does not modify any system interface, and could not increase the probability of an accident because these events are independent of this change. The proposed change does not change, degrade, or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the CNP UFSAR. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change allows some devices to be assigned an allocated response time, instead of a measured response time, when performing response time testing of the RTS and ESFAS protection channels. This change does not alter the performance of the pressure and differential pressure transmitters and switches, signal processing components, or actuation logic components used in the RTS and ESFAS protection systems. All applicable pressure and differential pressure sensors, signal processing components, and actuation logic components of the RTS and ESFAS protection systems will still have response time verified by test before placing the sensor in operational service and after any maintenance that could affect response time. Changing the method of periodically verifying response for certain components of the RTS and ESFAS protection systems (assuring component operability) from time response testing to calibration and channel checks does not create any new accident initiators or scenarios. Periodic surveillance of these components will continue, and may be used to detect significant degradation in the response characteristic that may cause the total response time allowance of the RTS and ESFAS protection systems to be exceeded. The total time response allowance for each RTS and ESFAS protection function bounds all degradation that cannot be detected by periodic surveillance. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change allows some devices to be assigned an allocated response time, instead of a measured response time, when performing response time testing of the RTS and ESFAS protection channels. The change does not affect the total system response times assumed in the safety analyses. The periodic system response time verification method for selected pressure and differential pressure sensors, signal processing components, and actuation logic components is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

within that defined in the safety analyses. Periodic surveillance of these components will continue, and may be used to detect significant degradation in the response characteristic that may cause the total response time allowance of the RTS and ESFAS protection systems to be exceeded. The total time response allowance for each RTS and ESFAS protection function bounds all degradation that cannot be detected by periodic surveillance. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.4

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

The CTS Section 1.0 definition of DOSE EQUIVALENT I-131 requires that the DOSE EQUIVALENT I-131 be calculated using either the thyroid dose conversion factors found in Table III of TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Regulatory Guide (RG) 1.109, Rev. 1 (Table E-7). The ITS allows DOSE EQUIVALENT I-131 to be calculated using any one of three thyroid dose conversion factors: TID-14844 (1962); Table E-7 of RG 1.109, Rev. 1 (1977); or ICRP 30, Supplement to Part 1, page 192-212, Table Titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity." This changes the CTS by allowing a third method, ICRP 30, Supplement to Part 1, to be used to calculate DOSE EQUIVALENT I-131.

The purpose of the defined term is to provide acceptable methods for computing DOSE EQUIVALENT I-131. Using thyroid dose conversion factors other than those given in TID-14844 results in lower doses and higher allowable activity but is justified by the discussion given in the Federal Register (FR page 23360 VI 56 No 98 May 21, 1991). This discussion accompanied the final rulemaking on 10 CFR 20 by the NRC. In that discussion, the NRC stated that they were incorporating modifications to existing concepts and recommendations of the ICRP and NCRP into NRC regulations. Incorporation of the methodology of ICRP 30 into the 10 CFR 20 revision was specifically mentioned with the explanation that changes being made result from changes in the scientific techniques and parameters used in calculating dose. In a response to a specific question as to whether or not the ICRP 30 dose parameters should be used, the NRC stated "Appropriate parameters for calculating organ doses can be found in ICRP 30 and its supplements..." Lastly, Commissioner Curtis provided additional views of the revised 10 CFR 20 with respect to the backfit rule. In that discussion, he stated that the AEC, when they issued the original 10 CFR 20, had emphasized that the standards were subject to change with the development of new knowledge and experience. He went on to say that the limits given in the revised 10 CFR 20 were based on up-to-date metabolic models and dose factors. This Federal Register entry shows clearly that, in general, the NRC was updating 10 CFR 20 to incorporate ICRP-30 recommendations and data. Given this discussion, it is concluded that using ICRP thyroid dose conversion factors to calculate DOSE EQUIVALENT I-131 is acceptable. In addition, RG 1.109 was developed by the NRC for the purpose of evaluating compliance with 10 CFR 50, Appendix I. The RG 1.109 thyroid dose conversion factors are higher than the ICRP 30 thyroid dose conversion factors for all five iodine isotopes in question. Therefore, using RG 1.109 thyroid dose conversion factors to calculate DOSE EQUIVALENT I-131 is more conservative than ICRP 30 and is therefore acceptable.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

    Response: No.

    The proposed use of ICRP 30 thyroid dose conversion factors to calculate DOSE EQUIVALENT I-131 is a change in analysis methodology which does not include a physical change to the plant, a new mode of plant operation, or a change in surveillance frequency. Therefore, the probability of a previously analyzed accident would not increase. If ICRP 30 thyroid dose conversion factors are used to calculate maximum dose equivalent iodine specific activity, the total iodine activity (in units of μCi/gm) will increase and this activity is used to calculate the doses resulting from a Main Steam Line Break (MSLB) or other analyzed accident. The calculated thyroid doses resulting from a MSLB or other analyzed accident would not increase as the same dose conversion factors used to calculate the DOSE EQUIVALENT I-131 thyroid activity would also be used to calculate the offsite thyroid doses. However, these dose conversion factors would be less than TID-14844 thyroid dose conversion factors used to calculate doses given in the UFSAR. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

    Response: No.

    The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not introduce a new mode of plant operation and does not require physical modification of the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

    Response: No.

    The proposed change only refines the method of calculating thyroid doses and DOSE EQUIVALENT I-131 activity. Using this method would not result in the thyroid doses changing significantly, since the same dose factors would be used to calculate the thyroid doses and DOSE EQUIVALENT I-131 activity. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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SUMMARY OF CHANGES
ITS CHAPTER 2.0

Change Description	Affected Pages
The change described in the response to Question 200405261358 for ITS 2.1.1.1 has been made. This change moves the Departure from Nucleate Boiling Ratio (DNBR) limits originally proposed to be in the Bases to the actual Specification consistent with NUREG-1431, Revision 2 Improved Standard Technical Specifications (ISTS) 2.1.1.1.	Pages 29 and 30 of 46.
The changes described in the response to Question 200404271521 for ITS 2.2.1 has been made. This change revises ITS 2.0 Justification for Deviations (JFD) 2 to provide additional justification for deleting the "restore compliance" statement from ISTS 2.2.1.	Page 31 of 46.
A self-identified change for ITS 2.1.2 Bases Applicable Safety Analyses Section, third paragraph, second sentence, has been made. This change replaces "reactor high pressure trip" with "pressurizer high pressure trip."	Page 41 of 46.

VOLUME 4

CNP UNITS 1 AND 2
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Chapter 2.0, Safety Limits

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ A.1

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in ~~Figure 2.1-1~~ ~~for 4 loop operation~~. A.2

the COLR

Add proposed SL 2.1.1.1 and SL 2.1.1.2 LA.1

APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 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.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 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

2.2.2.1 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.

MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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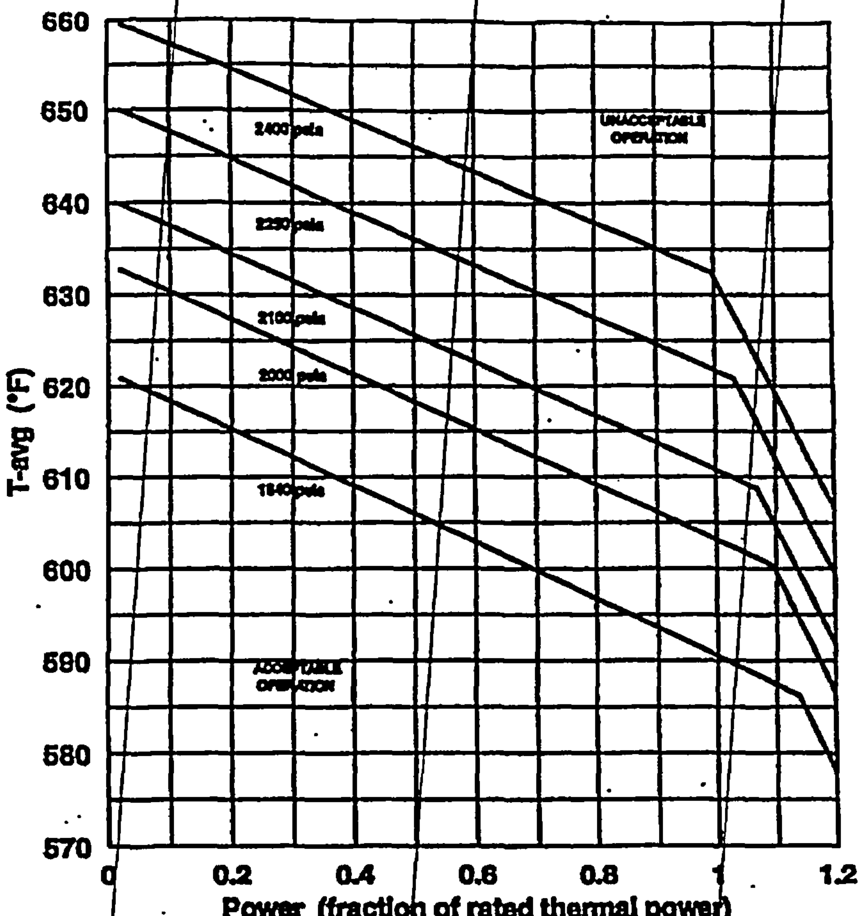
ITS

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ A.1

LA.1



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T-AVG IN °F)		
1840	(0.02, 620.86),	(1.136, 586.17),	(1.2, 577.94)
2000	(0.02, 632.79),	(1.094, 600.31),	(1.2, 586.52)
2100	(0.02, 639.85),	(1.068, 606.72),	(1.2, 591.77)
2250	(0.02, 649.96),	(1.031, 620.83),	(1.2, 599.40)
2400	(0.02, 659.52),	(0.996, 632.42),	(1.2, 606.63)

Figure 2.1-1 Reactor Core Safety Limits

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ITS

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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ITS

A.1

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

See ITS 3.3.1

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

ITS Chapter 2.0

ITS

A.1

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~

A.1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level - - High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

See ITS 3.3.1

*Design flow is 1/4 Reactor Coolant System total flow rate from Table 3.2.-1.

A.1

ITS

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13.	Steam Generator Water Level -- Low-Low	Greater than or equal to 17% of narrow range instrument span - each steam generator	Greater than or equal to 16% of narrow range instrument span - each steam generator
14.	Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 0.71 x 10^6 lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 0.73 x 10^6 lb/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15.	Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2750 volts - each bus	Greater than or equal to 2725 volts - each bus
16.	Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz each bus
17.	Turbine Trip		
	A. Low Fluid Oil Pressure	Greater than or equal to 800 psig	Greater than or equal to 750 psig
	B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18.	Safety Injection Input from ESF	Not Applicable	Not Applicable
19.	Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

See ITS 3.3.1

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 s}{1+\tau_2 s}\right] (T-T') + K_3 (P-P') - f_1(\Delta I)\right]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER (≤ 574.0 °F)
	P	=	Pressurizer pressure, psig
	P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig)
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.
	S	=	Laplace transform operator

See ITS 3.3.1

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A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = 1.17$
$K_2 = 0.0230$
$K_3 = 0.00110$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +3 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

See ITS 3.3.1

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A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_0 \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2(\Delta I)\right]$.

Where:	ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER (≤ 562.1 °F)
	K_4	=	1.083
	K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constants utilized in the rate lag controller for T_{avg} $\tau_3 =$ 10 secs.
	S	=	Laplace transform operator
	$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

See ITS 3.3.1

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ITS Chapter 2.0

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73.

b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President.

See CTS Chapter 6.0

6.7 SAFETY LIMIT VIOLATION

2.2 6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Chairman of the NSRB shall be notified within 24 hours. A.3 LA.2

b. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems or structures; and (3) corrective action taken to prevent recurrence. A.3

c. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NSRB and the Senior Vice President – Nuclear Operations within 14 days of the violation. LA.2

d. Operation of the unit shall not be resumed until authorized by the Commission. A.3

A.1

ITS

2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ A.1

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (T_{avg}) shall not exceed the limits shown in ~~Figure 2.1-1~~ the COLR ~~for 4 loop operation~~. A.2 LA.1

Add proposed SL 2.1.1.1 and SL 2.1.1.2

APPLICABILITY: MODES 1 and 2.

ACTION:

2.2.1 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.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 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

2.2.2.1 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.

MODES 3, 4 and 5

2.2.2.2 Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

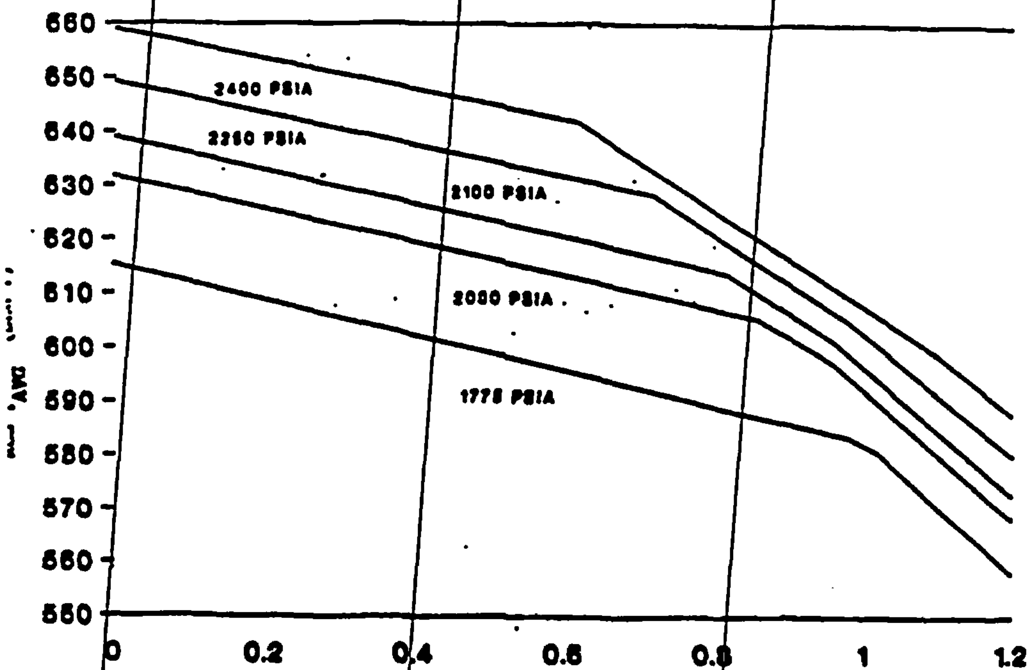
ITS

A.1

DESIGN FLOW = 91,600 GPM/LOOP

DESCRIPTION OF SAFETY LIMITS

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.4	0.98	583.8	1.02	580.9	1.2	558.1
2000	0.00	631.8	0.86	605.8	0.96	597.5	1.2	568.5
2100	0.00	639.1	0.82	614.0	0.96	601.6	1.2	573.1
2250	0.00	649.2	0.72	628.6	0.98	603.2	1.2	580.4
2400	0.00	659.0	0.62	642.0	1.1	599.0	1.2	588.1



LA.1

Figure 2.1-1 Reactor Core Safety Limits Four Loops in Operation

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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A.1

ITS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

See ITS 3.3.1

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ITS Chapter 2.0

ITS

A.1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER
	High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5. Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3 x 10^5 counts per second
7. Overtemperature Delta T	See Note 1	See Note 3
8. Overpower Delta T	See Note 2	See Note 4
9. Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10. Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11. Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12. Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

* Design flow is 91,600 gpm per loop.

See ITS 3.3.1

COOK NUCLEAR PLANT - UNIT 2 2-5 AMENDMENT NO. 82, 134

A.1

ITS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	Greater than or equal to 21% of narrow range instrument span - each steam generator	Greater than or equal to 19.2% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Less than or equal to 1.47 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator	Less than or equal to 1.56 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	Greater than or equal to 2905 volts - each bus	Greater than or equal to 2870 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	Greater than or equal to 57.5 Hz - each bus	Greater than or equal to 57.4 Hz - each bus
17. Turbine Trip		
A. Low Fluid Oil Pressure	Greater than or equal to 58 psig	Greater than or equal to 57 psig
B. Turbine Stop Valve Closure	Greater than or equal to 1% open	Greater than or equal to 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

See ITS 3.3.1

COOK NUCLEAR PLANT - UNIT 2 2-6 AMENDMENT NO. 83, 134, 151

A.1

ITS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1:
Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2[(1 + \tau_1 S)/(1 + \tau_2 S)](T - T') + K_3(P - P') - f_1(\Delta I)]$

Where:	ΔT_o	= Indicated ΔT at RATED THERMAL POWER
	T	= Average temperature, °F
	T'	= Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576.0 °F
	P	= Pressurizer Pressure, psig
	P'	= 2235 psig (indicated RCS nominal operating pressure)
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	= The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	= Time constants utilized in the lead-lag controller for T_{avg}; τ_1 = 28 secs, τ_2 = 4 secs.
	S	= Laplace transform operator

See ITS 3.3.1

A.1

ITS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

4 Loops in Operation

K1 = 1.09

K2 = 0.01331

K3 = 0.00058

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

See ITS 3.3.1

(i) for $q_t - q_b$ between -33 percent and +6 percent, $f_1(\Delta I)$=0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -33 percent, the ΔT trip setpoint shall be automatically reduced by 3.5 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 1.0 percent of its value at RATED THERMAL POWER.

COOK NUCLEAR PLANT - UNIT 2 2-8 AMENDMENT NO. 82, 134

ITS

A.1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \{K_4 - K_5[\tau_3 S/(1+\tau_3 S)]T - K_6[T - T''] - f_2(\Delta I)\}$

Where:

ΔT_o	= Indicated ΔT at rated power
T	= Average temperature, °F
T"	= Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576.0 °F
K_4	= 1.08
K_5	= 0.02/°F for increasing average temperature and 0 for decreasing average temperature
K_6	= 0.00197 for T greater than T"; K_6 = 0 for T less than or equal to T"
$\tau_3 S/(1+\tau_3 S)$	= The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	= Time constant utilized in the rate lag controller for T_{avg}: τ_3 = 10 secs.
S	= Laplace transform operator
$f_2(\Delta I)$	= 0.0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.3 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent ΔT span.

See ITS 3.3.1

A.1

ITS

6.0 ADMINISTRATIVE CONTROLS

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73.

b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NSRB and the Site Vice President.

See CTS Chapter 6.0

6.7 SAFETY LIMIT VIOLATION

2.2 6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Chairman of the NSRB shall be notified within 24 hours. A.3 LA.2

b. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the PORC. The report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems or structures; and (3) corrective action taken to prevent recurrence. A.3

c. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NSRB and the Senior Vice President – Nuclear Operations within 14 days of the violation. LA.2

d. Operation of the unit shall not be resumed until authorized by the Commission. A.3

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 2.1.1 references a curve providing limits on THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature (Tavg) "for 4 loop operation." ITS 2.1.1 does not contain this amplifying information.

This change is acceptable because the requirements have not changed. Both the ITS (ITS 3.4.4) and the CTS (CTS 3/4.4.1.1) require all four loops to be in operation in the applicable MODES (MODES 1 and 2). This change is designated as administrative because it eliminates redundant information in the CTS.

A.3 In the event that a safety limit is violated, CTS 6.7.1.a requires the NRC Operations Center to be notified by telephone within one hour, CTS 6.7.1.b requires a Safety Limit Violation Report to be prepared and specifies the information the report must contain, CTS 6.7.1.c requires the report to be submitted to the NRC, and CTS 6.7.1.d precludes resumption of operation of the unit until authorized by the NRC. The ITS does not specify any of these requirements.

These deletions are acceptable since the actual requirements are not being changed. These CTS requirements are duplicative of those currently located in 10 CFR 50.36(c)(1). Since CNP is required by the Operating License to comply with 10 CFR 50, the deletion of these requirements from the Technical Specifications is acceptable. The changes are designated as administrative since they are duplicative of regulations.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 2.1.1 requires that the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant average temperature not exceed the limits in Figure 2.1-1.

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

ITS 2.1.1 states that the combination of THERMAL POWER, RCS highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR and provides specific limits on DNBR and peak fuel centerline temperature. This changes the CTS by relocating limits that must be confirmed on a cycle specific basis to the COLR. The limiting Safety Limit parameters are retained in the SL.

The removal of these cycle specific parameter limits from the Technical Specifications to the COLR and the retention of the limiting Safety Limits in the Technical Specifications is acceptable because the cycle specific limits are developed or utilized under NRC-approved methodologies that ensure the Safety Limits are met. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the Safety Limits. NRC-approved Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," determined that the specific values for these parameters may be relocated to the COLR provided the limiting Safety Limits continue to appear in the Technical Specifications. The methodologies used to develop the parameters in the COLR were approved by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits of the safety analysis are met (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits). This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* In the event that a Safety Limit is violated, CTS 6.7.1.a requires the Chairman of the NSRB to be notified within 24 hours, CTS 6.7.1.b requires the Safety Limit Violation Report to be reviewed by the PORC, and CTS 6.7.1.c requires the report to be submitted to the Chairman of the NSRB and the Senior Vice President - Nuclear Operations within 14 days of the violation. The ITS does not include these requirements; they have been relocated to the Quality Assurance Program Description (QAPD).

The removal of these details for making notifications/reports from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The notification occurs following the Safety Limit violation and the reports are after-the-fact reports, thus they are not necessary to assure safe operation of the unit. The ITS still requires the unit to be shut down, and 10 CFR 50.36(c)(1) provides NRC reporting requirements and requires the NRC's permission to be obtained prior to restarting the unit. Also, this change is acceptable because these types of details will be adequately controlled in the QAPD. The QAPD is controlled under 10 CFR 50.54 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of

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DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

detail change because reporting requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 — 2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ [1].17 for the WRB-1 / WRB-2 DNB correlation[s]. INSERT 1
<Unit 1 only> <Unit 2 only>

2.1.1.2 The peak fuel centerline temperature shall be maintained < [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup]. ①

2.1.2 — 2.1.2 Reactor Coolant System Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ [2735] psig. ①

2.2 SAFETY LIMIT VIOLATIONS

2.1.1 Action — 2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour. ②

2.1.2 Action — 2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

1

INSERT 1

W-3 DNB correlation with a DNB correlation limit of ≥ 1.30 is used where the $\boxed{\text{WRB-1}}$ $\boxed{\text{WRB-2}}$ DNB correlation is not applicable.

{Unit 1 only}

{Unit 2 only}

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0, SAFETY LIMITS

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS 2.2.1 states that if SL 2.1.1 is violated to "restore compliance and be in MODE 3 within 1 hour." ISTS SL 2.1.1 is only applicable in MODES 1 and 2. CTS 2.1.1 is also applicable in MODES 1 and 2, and CTS 2.1.1 Action requires, whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, to be in HOT STANDBY within 1 hour. ITS 2.2.1 states that if SL 2.1.1 is violated to "be in MODE 3 within 1 hour." This changes ISTS 2.2.1 by deleting the "restore compliance" action.

   The "restore compliance" action is not necessary, since the remaining action to be in MODE 3 within 1 hour will restore compliance by placing the unit outside the Applicability of ITS 2.1.1. Being in MODE 3 is also likely to restore the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure to within the limits specified in the Core Operating Limits Report (COLR) as required by ITS SL 2.1.1, and is also likely to restore the departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature to within limits specified in ITS SL 2.1.1.1 and ITS SL 2.1.1.2, respectively, if these parameters were exceeded during MODES 1 and 2. In addition, this change is consistent with other specifications that do not require a "restore compliance" action when an action is included that requires exiting the Applicability of the Specification and the time frame is the same for either action, and is consistent with the CTS, which only requires being in MODE 3 within 1 hour.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

Reactor Core SLs
B 2.1.1

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES INSERT 1 (1)

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during ~~steady state operation, normal operational transients, and anticipated operational occurrences (AOOs)~~ INSERT 2. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature (Ref. 2). (2)

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

Trip T (2)

The proper functioning of the Reactor ~~Protection~~ System (R~~P~~S) and ~~steam generator~~ safety valves prevents violation of the reactor core SLs.

Main Steam (2)

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B 2.1.1

1 **INSERT 1**

Plant Specific Design Criterion (PSDC) 6

1 **INSERT 2**

all expected conditions of normal operation, with appropriate margins for uncertainties and specified transient situations that can be anticipated

BASES INSERT 3 (2)

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB, and (6)

b. The hot fuel pellet in the core must not experience centerline fuel melting. RTS (3)

The Reactor Trip System Setpoints Allowable Values (Ref. 2) (3), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and RCS Flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities. (2) (4)

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS (7) and the steam generator main steam safety valves. (2) (2) (7) (2)

The SLs represent a design requirement for establishing the RPS trip Setpoints Allowable Values (2) identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed (4) initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded. (3) (3) (2)

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB, and (6)

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

2 **INSERT 3**

operational transients and transient conditions arising from faults of moderate frequency

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BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10. INSERT 5
2. FSAR, Section 7.2. INSERT 6

2 **INSERT 4**

transient conditions arising from faults of moderate frequency

1 **INSERT 5**

UFSAR, Section 1.4.2.

2 **INSERT 6**

2. UFSAR, Section 3.5.3 (Unit 1) and Section 3.4.1 (Unit 2).

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to ~~10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).~~ INSERT 1 INSERT 2 ④ ① ~~Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.~~ INSERT 3

2485 psig ②

The design pressure of the RCS is ~~2500 psia~~. During normal operation and ~~AOOs~~ anticipated operational transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3). ②

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the ~~reactor~~ pressurizer high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded. ②

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control

1 **INSERT 1**

Plant Specific Design Criterion (PSDC) 9, "Reactor Coolant Pressure Boundary"

1 **INSERT 2**

shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. The RCS, in conjunction with its control and protective provisions, was designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation or anticipated system interactions, and to maintain the stresses within allowable code stress limits.

1 **INSERT 3**

PSDC 33, "Reactor Coolant Pressure Boundary Capability" (Ref. 1), the reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

BASES

APPLICABLE SAFETY ANALYSES (continued)

actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

anticipated operational transients

Allowable Values

determined

Allowable Value

The Reactor Trip System ~~setpoints~~ (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and ~~AOOs~~. The ~~reactor~~ high pressure trip ~~setpoint~~ is specifically ~~set~~ to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices. 2

pressurizer

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs); 6

b. ~~Steam line relief valve~~ Steam generator PORVs; 2

c. Steam Dump System; 6

d. Reactor Control System; 6

e. Pressurizer Level Control System; or

f. Pressurizer spray valve. 5 2

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 6) is 120% of design pressure. The ~~most~~ more limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig. 5 4

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

BASES

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.~~ INSERT 4 (1)

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR 100.

5. (U)FSAR, Section [7.2] (2)(5)

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

1 **INSERT 4**

UFSAR, Sections 1.4.2 and 1.4.6.

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JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0 BASES, SAFETY LIMITS

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Editorial correction made to the Bases.

4. Typographical/grammatical error corrected.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Attachment 1, Volume 4, Rev. 1, Page 44 of 46

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 2.0, SAFETY LIMITS

There are no specific NSHC discussions for this Chapter.

SUMMARY OF CHANGES
ITS SECTION 3.0

Change Description	Affected Pages
A self-identified change described in the response to Question 200406100852 for ITS LCO 3.0.4 and ITS SR 3.0.4 has been made. CTS Amendments 281 (Unit 1) and 265 (Unit 2) have been incorporated into the ITS submittal. This CTS change adopted the allowances of TSTF-359 and affects CTS 3.0.4, CTS 4.0.1, and CTS 4.0.4.	Pages 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 21, 23, 25, 28, 29, 30, 32, 33, 34, 35, 40, 41, 42, 43, 44, 45, 50, 51, 55, 56, 57, 58, 60, and 61 of 64.
A self-identified change for ITS SR 3.0.3 has been made. CTS Amendments 282 (Unit 1) and 266 (Unit 2) have been incorporated into the ITS submittal. This CTS change adopted the allowances of TSTF-358 and affects CTS 4.0.3.	Pages 7, 9, 13, 15, 26, and 64 of 64.
A self-identified change for ITS LCO 3.0.6 Bases has been made. This change corrects an error in the last sentence of the NUREG-1431, Revision 2 Improved Standard Technical Specifications (ISTS) LCO 3.0.6 Bases that has since been corrected in the NUREG-1431, Revision 3 ISTS LCO 3.0.6 Bases. Specifically, the last part of this sentence has been revised to state "...the appropriate LCO is the LCO for the support<u>ed</u> system."	Page 49 of 64.

VOLUME 5

CNP UNITS 1 AND 2
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS SECTION 3.0
LCO and SR APPLICABILITY

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Section 3.0, LCO and SR Applicability

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A.1

.0 (LCO)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.0 APPLICABILITY

A.1

LIMITING CONDITION FOR OPERATION

LCOs met

3.0.1 3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification, except as provided in Specification 3.0.6.

LCO 3.0.2 and LCO 3.0.7 in the Applicability A.2

3.0.2 3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification, except as provided in Specification 3.0.6. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

INSERT 1 A.3

an LCO and S

3.0.3 3.0.3 When a Limiting Condition for Operation is not met except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

INSERT 2 A.4

MODE 3 7 13 A.5

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

MODE 4 MODE 5 37

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

INSERT 3 A.6 this Specification A.1

an LCO

3.0.4 3.0.4 When a Limiting Condition for Operation is not met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

s ;

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

A.1

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied,

See ITS 3.8.1

A.3

INSERT 1

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

INSERT 2

are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. A.4 A.5

A.6

INSERT 3

in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

INSERT 4

Not Used

Insert Page 3/4 0-1

.0 (LCO)

3/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.0 APPLICABILITY

within 2 hours action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it as applicable in: — See ITS 3.8.1

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in MODES 5 or 6.

3.0.5 3.0.6 Equipment removed from service or declared inoperable to comply with ACTION requirements may be returned to service under administrative controls solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to Specifications 3.0.1 and 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY. — S, LCO, A.1, A.8, A.9, INSERT 5, INSERT 6, A.10

3.0 SURVEILLANCE REQUIREMENTS — SRs, LCOs, (SR) APPLICABILITY

SR 3.0.1 4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other specified conditions in the Applicability for individual Limiting Condition for Operations, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform Surveillance within the specified frequency shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits. — SR, LCO, a, SR 3.0.3, A.1

SR 3.0.2 4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval. — INSERT 8, A.12, M.1, L.3

SR 3.0.1 4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. — A.11

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. — LCO, Frequency, A.1

SR 3.0.3 If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be met. — LCO, Condition(s), is, entered

SR 3.0.3 When the Surveillance is performed within the delay period and the Surveillance criteria are not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be met. — LCO, Condition(s), A.1

SR 3.0.1 Surveillance requirements do not have to be performed on inoperable equipment. — A.11

SR 3.0.4 4.0.4 Entry into an OPERATIONAL MODE or other specified condition in the Applicability of a Limited Condition for Operation shall only be made when the Limiting Condition for Operation Surveillances have been met within their specified frequency, except as provided by Specification 4.0.3. When a Limiting Condition for Operation is not met due to Surveillances not having been met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4. — LCO's, n, LCO, n, SR 3.0.4, SR 3.0.3, A.1

A.9 **INSERT 5**

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

A.10 **INSERT 6**

LCO 3.0.7 Test Exception LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

Insert Page 3/4 0-2a

A.11 **INSERT 7**

Not Used

INSERT 8

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. A.12

For Frequencies specified as "once," the above interval extension does not apply. M.1

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. L.3

Exceptions to this Specification are stated in the individual Specifications. A.12

ITS

A.1

.0 (SR)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS A.1
3/4.0 APPLICABILITY

SR 3.0.4 This provision shall not prevent entry into OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. A.1

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.

SURVEILLANCE REQUIREMENTS

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

See ITS 5.5

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Deleted

4.0.7 Deleted

ITS

A.1

.0 (LCO)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS A.1
3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

LCOs met

3.0.1 3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification, except as provided in Specification 3.0.6. A.2

LCO 3.0.2 and LCO 3.0.7 in the Applicability

3.0.2 3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification, except as provided in Specification 3.0.6. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required. INSERT 1 A.3

an LCO and S A.4

3.0.3 3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated, to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in: INSERT 2

7 13 A.5

MODE 3 1. At least HOT STANDBY within the next 6 hours,
MODE 4 2. At least HOT SHUTDOWN within the following 6 hours, and
MODE 5 3. At least COLD SHUTDOWN within the subsequent 24 hours. 37

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications. INSERT 3 A.6

this Specification A.1

n LCO

3.0.4 3.0.4 When a Limiting Condition for Operation is not met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or A.1 :

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, See ITS 3.8.1

A.3

INSERT 1

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

INSERT 2

are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. A.4 A.5

A.6

INSERT 3

in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

INSERT 4

Not Used

Insert Page 3/4 0-1

(LCO)

.0

3/4 3/4.0 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS APPLICABILITY A.1

within 2 hours action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it as applicable in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in MODES 5 or 6.

See ITS 3.8.1

3.0.5 3.0.6 Equipment removed from service or declared inoperable to comply with ACTION requirements may be returned to service under administrative controls solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to Specifications 3.0.1 and 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

S LCO A.1 A.8 INSERT 5 A.9 INSERT 6 A.10

3.0 SURVEILLANCE REQUIREMENTS

SRs LCOs SR APPLICABILITY

SR 3.0.1 4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other specified conditions in the Applicability for individual Limiting Condition for Operations, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR LCO SR 3.0.3 A.1

SR 3.0.2 4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

INSERT 8 A.12 M.1 L.3

SR 3.0.1 4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification.

A.11

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

LCO Frequency A.1

SR 3.0.3 If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be met.

LCO entered Condition(s) is

SR 3.0.3 When the Surveillance is performed within the delay period and the Surveillance criteria are not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be met.

LCO Condition(s) A.1

SR 3.0.1 Surveillance requirements do not have to be performed on inoperable equipment. A.11

A.9 **INSERT 5**

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

A.10 **INSERT 6**

LCO 3.0.7 Test Exception LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

A.11

INSERT 7

Not Used

INSERT 8

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. A.12

For Frequencies specified as "once," the above interval extension does not apply. M.1

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. L.3

Exceptions to this Specification are stated in the individual Specifications. A.12

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A.1

ITS

.0

(SR)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.0 APPLICABILITY

LCO's n

SR 3.0.4 4.0.4 Entry into an OPERATIONAL MODE or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillances have been met within their specified frequency, except as provided by Specification 4.0.3. When a Limiting Condition for Operation is not met due to Surveillances not having been met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

LCO SR 3.0.3 LCO n SR 3.0.4 A.1

This provision shall not prevent entry into OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.

b. Surveillance Intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

See ITS 5.5

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Deleted

4.0.7 Deleted

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DISCUSSION OF CHANGES
ITS SECTION 3.0, LCO AND SR APPLICABILITY

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.0.1 states, "Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification, except as provided in Specification 3.0.6." ITS LCO 3.0.1 states, "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7." This results in several changes to the CTS.

- Certain phrases are revised to be consistent with the equivalent phrase used in the ITS. Specifically, "Limiting Conditions for Operation" is changed to "LCOs" and "OPERATIONAL MODES or other conditions specified" is changed to "MODES or other specified conditions" to be consistent with the ITS definition of MODE and the terminology used in the ITS.

  These changes are acceptable because they result in no change in the intent or application of the Technical Specification, but merely reflect editorial preferences used in the ITS.

- The phrase ". . . ACTION requirements shall be applicable during the OPERATIONAL MODES . . ." is moved from CTS 3.0.1 to ITS LCO 3.0.2 which states that when an LCO is not met, the Required Actions must be met.

  The change is acceptable because moving this information within the Technical Specifications results in no change in the intent or application of ACTIONS.

- The phrase "shall be applicable" is replaced in ITS LCO 3.0.1 with the phrase "shall be met." This change is made to be consistent with the ITS terminology and to clarify the concept of an LCO being met (i.e., being in compliance with the requirements of the LCO), versus the LCO being applicable or required (i.e., the requirements in the LCO apply).

  This change is acceptable because it is an editorial change that does not change the intent of the requirements.

- The phrase "except as provided in Specification 3.0.6" is replaced in ITS LCO 3.0.1 with the phrase "except as provided in LCO 3.0.2 and LCO 3.0.7." ITS LCO 3.0.2 describes the appropriate actions to be taken when

Attachment 1, Volume 5, Rev. 1, Page 17 of 64

DISCUSSION OF CHANGES
ITS SECTION 3.0, LCO AND SR APPLICABILITY

ITS LCO 3.0.1 is not met. LCO 3.0.7 describes Test Exception LCOs, which are exceptions to other LCOs. CTS 3.0.6 (ITS LCO 3.0.5) does not modify ITS LCO 3.0.1 since the ACTION requirements discussion that is in CTS 3.0.1 has been moved to ITS LCO 3.0.2, as described above.

This change is acceptable because adding the exception for LCO 3.0.2 and LCO 3.0.7 prevents a conflict within the Applicability section. This addition is needed for consistency in the ITS requirements and does not change the intent or application of the Technical Specifications.

These changes are designated administrative because they are editorial and result in no technical changes to the Technical Specifications.

A.3 CTS 3.0.2 states, "Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification, except as provided in Specification 3.0.6. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required." ITS LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated." This results in several changes to the CTS.

- The first sentence in CTS 3.0.2 states, in part, "Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION . . . shall constitute compliance with the specification." This requirement is divided into portions of ITS LCO 3.0.1, "LCOs shall be met" and ITS LCO 3.0.2, "Upon discovery of failure to meet an LCO, the Required Actions of the associated Conditions shall be met."

  This change is acceptable because the intent of the CTS requirement is preserved, but the aspects of LCO compliance and the performance of ACTIONS when the LCO is not met are separated.

- The CTS 3.0.2 term "Specification 3.0.6" has been changed in ITS LCO 3.0.2 to "LCO 3.0.5" due to renumbering and consistency with the terminology in the ITS.

  This change is acceptable because it results in no change in the intent or application of the Technical Specification, but merely reflects an editorial preference and renumbering.

- CTS 3.0.2 is revised to include an exception for ITS LCO 3.0.6. LCO 3.0.6 is a new allowance that takes exception to the ITS LCO 3.0.2 requirement to take the Required Actions when the associated LCO is not met. This exception is included in LCO 3.0.2 to avoid conflicts between the applicability requirements.

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This change is acceptable because it includes a reference to a new item in the ITS and results in no change to the CTS. Changes resulting from the incorporation of LCO 3.0.6 are discussed in Discussion of Change (DOC) A.9.

- The second sentence of CTS LCO 3.0.2 states "In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required." The sentence is replaced in ITS LCO 3.0.2 with "If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated."

  This change is acceptable because, while worded differently, both the CTS and ITS state that ACTIONS do not have to be completed once the LCO is met or is no longer applicable. ITS LCO 3.0.2 also adds the phrase, "unless otherwise stated." There are some ITS ACTIONS which must be completed, even if the LCO is met or is no longer applicable. This change is acceptable because it reflects a new feature in the ITS which did not exist in the CTS. The technical aspects of these changes are discussed in the appropriate ITS sections.

These changes are designated as administrative because they are editorial and do not result in technical changes to the Technical Specifications.

A.4 CTS LCO 3.0.3 is applicable "When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements." ITS LCO 3.0.3 expands those applicability requirements so that the requirement is applicable "When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS." This changes the CTS to add two new applicability conditions.

- ITS LCO 3.0.3 is applicable when the LCO is not met and there is no applicable ACTION to be taken.

  This change is acceptable because it is consistent with the current understanding and application of CTS 3.0.3.

- ITS LCO 3.0.3 is applicable when directed by the associated ACTIONS. The CTS do not contain requirements that direct entry into LCO 3.0.3. The ITS does contain such requirements. Any technical changes related to directing LCO 3.0.3 entry in an ACTION will be discussed in the affected Technical Specifications.

  This change is acceptable because referencing a new feature in the ITS is an editorial change.

These changes are designated as administrative because they do not result in any technical changes to the Technical Specifications.

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A.5 CTS 3.0.3 states the shutdown time limits in sequential order; i.e., each time limit is measured from the completion of the previous step. ITS 3.0.3 states the time limits (Completion Times) from the time the condition was entered. In addition, the MODE titles used in CTS 3.0.3 are replaced with the corresponding MODE numbers in ITS LCO 3.0.3. The stated times in CTS 3.0.3 and ITS LCO 3.0.3 are listed below:

Mode	Title	CTS Time to Enter Mode	ITS Time to Enter Mode
--	(Current Mode)	1 hour to begin action	1 hour to begin action
3	Hot Standby	within the next 6 hours	7 hours
4	Hot Shutdown	within the following 6 hours	13 hours
5	Cold Shutdown	within the subsequent 24 hours	37 hours

These changes are acceptable because the ITS times are the sum of the CTS times (e.g., the ITS Completion Time of 37 hours to enter MODE 5 is the same as the sum of the CTS allowance of 1 hour, 6 hours, 6 hours, and 24 hours.) This changes the CTS presentation only, and the time allowed to enter each MODE is unchanged. Using MODE numbers instead of the corresponding MODE titles is an editorial preference which results in no change to the requirements in the Technical Specifications. In addition, the CTS 3.0.3 statement "within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it" has been editorially reworded in ITS LCO 3.0.3 to "the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. ACTION shall be initiated within 1 hour to place the unit..." These changes are acceptable because they result in no change in the intent or application of the Technical Specification, but merely reflect editorial preferences used in the ITS.

These changes are designated as administrative as they implement the editorial conventions used in the ITS without resulting in technical changes to the Technical Specifications.

A.6 CTS 3.0.3 states "Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation." ITS LCO 3.0.3 states "Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required."

This change is acceptable because the changes to CTS 3.0.3 are editorial. Both the CTS and ITS state that LCO 3.0.3 can be exited if the LCO which led to the entry into LCO 3.0.3 is met, or if one of the ACTIONS of that LCO is applicable. The CTS requirement also specifies that the time to complete the ACTIONS in

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the LCO is based on the initial failure to meet the LCO. Reentering the LCO after exiting LCO 3.0.3 does not reset the ACTION statement time requirements. This information is not explicitly stated in ITS LCO 3.0.3 but is true under the multiple condition entry concept of the ITS. In addition, the sentence "LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4" is added to ITS LCO 3.0.3. CTS 3.0.3 and ITS LCO 3.0.3 require the unit to be placed only as low as COLD SHUTDOWN (MODE 5). Once the unit is in MODE 5, there are no further requirements. Thus, CTS 3.0.3 and ITS LCO 3.0.3 are effectively only applicable in MODES 1, 2, 3, and 4, and the addition of the sentence merely reflects editorial preferences used in the ITS.

These changes are designated as administrative because there is no change in the intent or application of the CTS 3.0.3 requirements.

A.7 Not used.

A.8 CTS 3.0.6 has a statement that CTS 3.0.6 is an exception to both CTS 3.0.1 and CTS 3.0.2. ITS LCO 3.0.5 includes only a statement that ITS LCO 3.0.5 is an exception to LCO 3.0.2. The statement that ITS LCO 3.0.5 is an exception to LCO 3.0.1 is not included.

This change is acceptable since ITS LCO 3.0.5 does not modify ITS LCO 3.0.1. The ACTION requirements discussion that is in CTS 3.0.1 has been moved to ITS LCO 3.0.2 (i.e., it is not included in ITS LCO 3.0.1). This change is designated as administrative since it does not result in any technical change to the Technical Specifications.

A.9 ITS LCO 3.0.6 is added to the CTS to provide guidance regarding the appropriate ACTIONS to be taken when a single inoperability (a support system) also results in the inoperability of one or more related systems (supported system(s)). LCO 3.0.6 states "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2." In the CTS, based on the intent and interpretation provided by the NRC over the years, there has been an ambiguous approach to the combined support/supported inoperability. Some of this history is summarized below:

- Guidance provided in the June 13, 1979, NRC memorandum from Brian K. Grimes (Assistant Director for Engineering and Projects) to Samuel E. Bryan (Assistant Director for Field Coordination) would indicate an

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ITS SECTION 3.0, LCO AND SR APPLICABILITY

intent/interpretation consistent with the proposed LCO 3.0.6, without the necessity of also requiring additional ACTIONS. That is, only the inoperable support system ACTIONS need be taken.

- Guidance provided by the NRC in their April 10, 1980, letter to all Licensees, regarding the definition of OPERABILITY and its impact as a support system on the remainder of the CTS, would indicate a similar philosophy of not taking ACTIONS for the inoperable supported equipment. However, in this case, additional actions (similar to the proposed Safety Function Determination Program actions) were addressed and required.

- Generic Letter 91-18 and a plain-English reading of the CTS provide an interpretation that inoperability, even as a result of a Technical Specification support system inoperability, requires all associated ACTIONS to be taken.

- Certain CTS contain ACTIONS such as "Declare the {supported system} inoperable and take the ACTIONS of {its Specification}." In many cases, the supported system would likely already be considered inoperable. The implication of this presentation is that the ACTIONS of the inoperable supported system would not have been taken without the specific direction to do so.

Considering the history of misunderstandings in this area, the WOG ISTS, NUREG-1431, Rev. 2, was developed with Industry input and approval of the NRC to include LCO 3.0.6 and a new program, Specification 5.5.13, "Safety Function Determination Program (SFDP)." This change is acceptable since its function is to clarify existing ambiguities and to maintain actions within the realm of previous interpretations. This change is designated as administrative because it does not technically change the Technical Specifications.

A.10 ITS LCO 3.0.7 is added to the CTS. LCO 3.0.7 states "Test Exception LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications."

This change is acceptable because the CTS contain test exception specifications which allow certain LCOs to not be met for the purpose of special tests and operations. However, the CTS does not contain the equivalent of ITS LCO 3.0.7. As a result, there could be confusion regarding which LCOs are applicable during special tests. LCO 3.0.7 was crafted to avoid that possible confusion. LCO 3.0.7 is consistent with the use and application of CTS test exception Specifications and does not provide any new restriction or allowance. This change is

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designated as administrative because it does not technically change the Technical Specifications.

A.11 The first sentence of CTS 4.0.3 states "Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification." The last sentence of CTS 4.0.3 states "Surveillance Requirements do not have to be performed on inoperable equipment." CTS 4.0.1 contains similar requirements, in that it states, in part, "Failure to perform surveillance within the specified frequency shall be failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3." Furthermore, CTS 4.0.1 states "Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." ITS SR 3.0.1 states "SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." The changes to the CTS are:

- The first sentence of CTS 4.0.3 states "Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification." This information is consistent with the current wording in CTS 4.0.1 and proposed ITS SR 3.0.1. ITS SR 3.0.1 states "Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3."

  This change is acceptable because it is consistent with the current use and application of the Technical Specifications and with previous NRC guidance, and moves information within the Technical Specifications with no change in intent. This change is designated as administrative because it clarifies the Technical Specifications with no change in intent.

- CTS 4.0.3 states, in part, "Surveillance requirements do not have to be performed on inoperable equipment." CTS 4.0.1 includes this allowance, but also states that Surveillances do not have to be performed on variables outside specified limits. ITS SR 3.0.1 states "Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." The allowance in CTS 4.0.3 is duplicative of the allowance in CTS 4.0.1. This changes the CTS by incorporating the allowance of CTS 4.0.3 into CTS 4.0.1 (ITS SR 3.0.1).

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This change is acceptable and is designated as administrative because it moves and clarifies information within the Technical Specifications with no change in intent.

A.12 CTS 4.0.2 states "Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as 'once,' the above interval extension does not apply. If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This results in several changes to the CTS.

- ITS SR 3.0.2 adds to the CTS "For Frequencies specified as 'once,' the above interval extension does not apply." This is described in DOC M.1.

- ITS SR 3.0.2 adds to the CTS "If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance." This is described in DOC L.3.

- CTS 4.0.2 states "Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states, in part, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency." This change is made to be consistent with the ITS terminology and to clarify the concept of the specified SR Frequency being met.

  The change is acceptable since it does not change the intent of the requirements.

- ITS SR 3.0.2 is more specific regarding the start of the Frequency by stating "as measured from the previous performance or as measured from the time a specified condition of the Frequency is met." This direction is consistent with the current use and application of the Technical Specifications.

  This change is acceptable because the ITS presentation has the same intent as the CTS requirement.

- ITS SR 3.0.2 adds to the CTS "Exceptions to this Specification are stated in the individual Specifications."

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ITS SECTION 3.0, LCO AND SR APPLICABILITY

This change is acceptable because it reflects practices used in the ITS that are not used in the CTS. Any changes to a Technical Specification, by inclusion of such an exception, will be addressed in the affected Technical Specification.

The changes are designated as administrative because they reflect presentation and usage rules of the ITS without making technical changes to the Technical Specifications.

A.13 Not used.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.0.2 states "Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as 'once,' the above interval extension does not apply. If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding "For Frequencies specified as 'once,' the above interval extension does not apply." The remaining changes to CTS 4.0.2 are discussed in DOC A.12 and DOC L.3.

The purpose of the 1.25 extension allowance to Surveillance Frequencies is to allow for flexibility in scheduling tests. This change is acceptable because Frequencies specified as "once" are typically condition-based Surveillances in which the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Frequency without extension in order to avoid operation in unacceptable conditions. This change is designated as more restrictive because an allowance to extend Frequencies by 25% is eliminated from some Surveillances.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 Not used.

DISCUSSION OF CHANGES
ITS SECTION 3.0, LCO AND SR APPLICABILITY

L.2 Not used.

L.3 CTS 4.0.2 states "Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as 'once,' the above interval extension does not apply. If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, "If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance." The remaining changes to CTS 4.0.2 are discussed in DOC A.12 and DOC M.1.

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions which must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time without extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because additional time is provided to perform some periodic Required Actions.

L.4 Not Used.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

3.0.1	LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.
3.0.2	LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
3.0.3	LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours; (3) b. MODE 4 within 13 hours; and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.
3.0.4	LCO 3.0.4	~~When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time.~~ INSERT 1 This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. ~~Exceptions to this Specification are stated in the individual Specifications.~~ TSTF-359 ~~LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.~~

TSTF-359

INSERT 1

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications~~, or~~ ;

3

c. When an allowance is stated in the individual value, parameter, or other Specification.

CTS

3.0 LCO Applicability

LCO 3.0.4 (continued)

~~- REVIEWER'S NOTE -
LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

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3.0.6 | LCO 3.0.5 | Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

DOC A.9 | LCO 3.0.6 | When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.~~15~~ 13, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. (4)

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

DOC A.10 | LCO 3.0.7 | Test Exception LCOs 3.1.8 ~~and 3.4.19~~ , "PHYSIC TESTS Exceptions - MODE 2," allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs (5)

CTS

Doc A.10

3.0 LCO Applicability

LCO 3.0.7 (continued)

is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

CTS

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

CTS		
4.0.1, 4.0.3	SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
4.0.2	SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
4.0.3	SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
4.0.4	SR 3.0.4	Only ~~Entry into a MODE or other~~ when specified condition in the Applicability of an LCO shall ~~not~~ be made ~~unless~~ the LCO's Surveillances have been met within their specified Frequency. INSERT 2 This provision shall not prevent entry into

TSTF-359

TSTF-359 **INSERT 2**

, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

CTS

3.0 SR Applicability

4.0.4 SR 3.0.4 (continued)

MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

~~SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.~~

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~~- REVIEWER'S NOTE -~~
~~SR 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, SR 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in SR 3.0.4 were previously applicable in all MODES. Before this version of SR 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

JUSTIFICATION FOR DEVIATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

1. Not used.
2. Not used.
3. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.
4. Change made to be consistent with a change made in another Specification.
5. The brackets have been removed and the proper plant specific information/value has been provided.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.9 and apply at all times, unless otherwise stated. (1)
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and (9)

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

BASES

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and either: 2

a. An associated Required Action and Completion Time is not met and no other Condition applies; or 9

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of

BASES

LCO 3.0.3 (continued)

Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant unit 7
upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met; 9

b. A Condition exists for which the Required Actions have now been performed; or 9

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

BASES

LCO 3.0.3 (continued)

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive ~~C~~ondition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. 2

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.~~15~~, "Fuel Storage Pool Water Level." LCO 3.7.~~15~~ has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.~~15~~ are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.~~15~~ "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications. 14 4 to 2

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It ~~precludes~~ allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when ~~the following exist~~: INSERT 1 TSTF-359

~~a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered and~~

TSTF-359 **INSERT 1**

unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c.

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BASES

LCO 3.0.4 (continued)

b. ~~Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.~~

INSERT 2

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

INSERT 3

TP

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

INSERT 4

~~Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.~~

INSERT 5

~~LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. [In some cases (e.g., ..) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]~~

TSTF-359 **INSERT 2**

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time.

TSTF-359 **INSERT 3**

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

LCO 3.0.4.b may be used with single or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components. 2

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

TSTF-359 **INSERT 3 (continued)**

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these system and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable. s 2

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., [Containment Air Temperature, Containment Pressure, ~~MCPR~~ Moderator Temperature Coefficient]), and may be applied to other Specifications based on NRC plant-specific approval. and 6

TSTF-359 **INSERT 4**

In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

TSTF-359 **INSERT 5**

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

BASES

LCO 3.0.4 (continued)

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, ~~changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with~~ LCO 3.0.4 ~~or where an exception to LCO 3.0.4 is stated~~ is not a violation of SR 3.0.1 or SR 3.0.4 for ~~those~~ Surveillances that ~~do not~~ have ~~to be~~ performed ~~due to the associated~~ inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

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TSTF-359

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate ~~either~~: (2)

a. The OPERABILITY of the equipment being returned to service ~~or~~ (9)

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions. (2)

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions. (2)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2. (4)

(13)

Specification 5.5.~~15~~, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required.

BASES

LCO 3.0.6 (continued)

The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and: 6

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1) 9

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2) or 9

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

EXAMPLE B 3.0.6-1 2

If System 2 of Train A is inoperable and System 5 of Train B is inoperable, a loss of safety function exists in supported System 5.

EXAMPLE B 3.0.6-2

If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

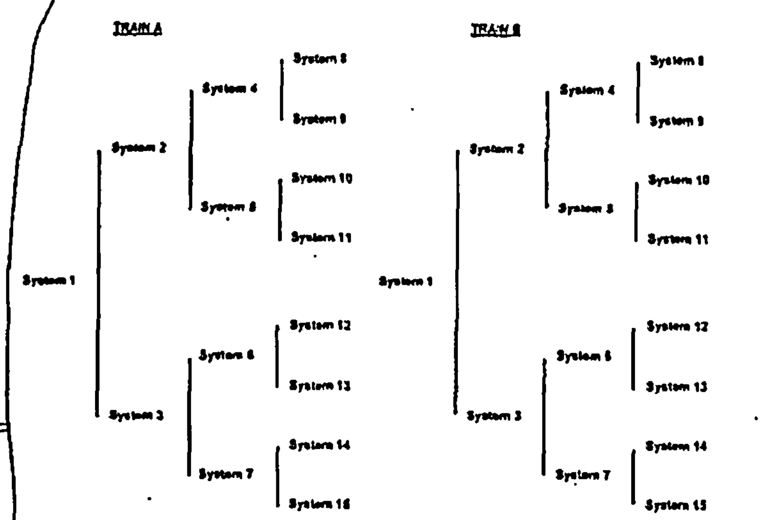
EXAMPLE B 3.0.6-3

If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11. 2 6

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

BASES

LCO 3.0.6 (continued)



move to end of Section

Figure B 3.0-1
Configuration of Trains and Systems

7

6

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY). are 2

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for

BASES

LCO 3.0.6 (continued)

the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the support system. ed ② ②

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.4.19 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

"PHYSICS TESTS Exceptions - MODE 2," ⑥

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

in Sections 3.1 through 3.9 ①

SRs — SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1 — SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. INSERT 6 TSTF-434

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. ~~This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.~~ ⑧

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current

TSTF-434

INSERT 6

Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

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BASES

SR 3.0.1 (continued)

MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

An ~~Some~~ example~~s~~ of this process ~~are~~ is:

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures ≥ ~~800~~ 850 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing. 9 4

~~b. High pressure safety injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.~~ 4

SR 3.0.2 SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

BASES

SR 3.0.2 (continued)

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. SR 3.0.3 provides a time limit for, and allowances for the performance of,

BASES

SR 3.0.3 (continued)

Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant unit (3)
configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC (3)
(1) Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance (2)
Activities at Nuclear Power Plants." This Regulatory Guide addresses (2)
consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including (3)
unit plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

BASES

SR 3.0.4 SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. TSTF-359

~~The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.~~

INSERT 7 TSTF-359

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. (2)

INSERT 8 TSTF-359

~~The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.~~ SR

INSERT 9

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

TSTF-359 **INSERT 7**

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to Surveillance not being met in accordance with LCO 3.0.4.

TSTF-359 **INSERT 8**

SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

TSTF-359 **INSERT 9**

In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

BASES

SR 3.0.4 (continued)

~~SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.~~

TSTF-359

JUSTIFICATION FOR DEVIATIONS
ITS SECTION 3.0 BASES, LCO AND SR APPLICABILITY

1. ITS LCO 3.0.1 and ITS SR 3.0.1 Applicabilities only apply to Specifications in ITS Sections 3.1 through 3.9; they do not apply to Specifications in Chapter 4.0 and Chapter 5.0 unless specifically stated in those Specifications. Therefore, this statement has been added for clarity.

2. Typographical/grammatical error corrected.

3. Changes have been made for consistency with similar discussions/terminology in the Bases.

4. The Bases are changed to reflect a change to the Specifications.

5. Not used.

6. The brackets have been removed and the proper plant specific information/value has been provided.

7. The Figure has been moved to the end of the Section, consistent with the format of the ITS.

8. The ITS SR 3.0.3 Bases allows credit to be taken for unplanned events that satisfy Surveillances. The Bases further states that this allowance also includes those SRs whose performance is normally precluded in a given MODE or other specified condition. This portion of the allowance has been deleted. As documented in Part 9900 of the NRC Inspection Manual, Technical Guidance - Licensee Technical Specifications Interpretations, and in the Bases Control Program (ITS 5.5.12), neither the Technical Specifications Bases nor Licensee generated interpretations can be used to change the Technical Specification requirements. Thus, if the Technical Specifications preclude performance of an SR in certain MODES (as is the case for some SRs in ITS Section 3.8), the Bases cannot change the Technical Specifications requirement and allow the SR to be credited for being performed in the restricted MODES, even if the performance is unplanned.

9. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.1

Not used.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.2

Not used.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.3

CNP is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

CTS 4.0.2 states "Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval." ITS SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as 'once,' the above interval extension does not apply. If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This changes the CTS by adding, "If a Completion Time requires periodic performance on a 'once per . . .' basis, the above Frequency extension applies to each performance after the initial performance." The remaining changes to CTS 4.0.2 are discussed in DOC A.12 and DOC M.1.

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions which must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time without extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because additional time is provided to perform some periodic Required Actions.

Indiana Michigan Power Company (I&M) has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. This change does not affect the probability of an accident. The length of time between performance of Required Actions is not an initiator to any accident previously evaluated. The consequences of any accident previously evaluated are the same during the Completion Time or during any extension of the Completion Time. As a result, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. This change will not physically alter the plant (no new or different type of equipment will be installed). Also, the change does not involve any new or revised operator actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. The 25% extension allowance is provided for scheduling convenience and is not expected to have a significant effect on the average time between Required Actions. As a result, the Required Actions will continue to provide appropriate compensatory measures for the subject Condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.4

Not used.

SUMMARY OF CHANGES
ITS SECTION 3.1

Change Description	Affected Pages
A self-identified change for ITS 3.1.1, 3.1.4, 3.1.5, and 3.1.6 Bases has been made. This change revises ITS 3.1.1, 3.1.4, 3.1.5, and 3.1.6 Bases to incorporate miscellaneous editorial changes, and is administrative.	Pages 25, 130, 132, 133, 157, 158, 183, and 184 of 357.
The change described in the response to Question 200405261418 for ITS 3.1.3 has been made. This change provides an additional Justification for Deviations (JFD) discussion on how the clarity is enhanced by moving the Surveillance Notes into the Frequency for the NUREG-1431, Revision 2 Improved Standard Technical Specifications (ISTS) SR 3.1.3.2.	Page 80 of 357.
The change described in the response to Question 200405261443 for ITS 3.1.4 Bases has been made. This change restores ITS 3.1.4 Bases for Required Actions A.1.1 and A.1.2 to the ISTS 3.1.4 Bases for Required Actions A.1.1 and A.1.2 wording.	Pages 114, 136, and 137 of 357.
A self-identified change for ITS 3.1.4 has been made. This change revises ITS 3.1.4 Required Action B.1.1 to state "Verify SDM is within limits" to be consistent with terminology used in this and other Specifications (e.g., ITS 3.1.5 Required Action A.1.1).	Page 122 of 357.
The change described in the response to Question 200405261512 for ITS 3.1.7 has been made. This change restores ITS 3.1.7 Condition C to the ISTS 3.1.7 Condition C wording.	Pages 197, 198, 202, 207, 210, and 220 of 357.
The change described in the response to Question 200405261545 for ITS 3.1.7 has been made. This change restores ITS SR 3.1.7.1 Frequency to the ISTS SR 3.1.7.1 Frequency.	Pages 199, 208, 209, and 223 of 357.
A self-identified change for ITS 3.1.7 Bases has been made. This change revises ITS 3.1.7 ACTIONS A.1 Bases to clarify that the position of control rod H-8 cannot be determined indirectly using the movable incore detectors, because the control rod is located in the center of the core.	Pages 218, 219 and 224 of 357.
A self-identified change for ITS 3.1.8 has been made. This change revises ITS 3.1.8 to add ITS LCO 3.3.1 Function 18.d to the listing of applicable ITS LCO 3.3.1 Functions as a result of the change described in the response to Question 200409291347 for ITS 3.3.1.	Pages 229, 230, 231, 236, 239, and 248 of 357.
A self-identified change for ITS 3.1.8 has been made. CTS Amendments 283 (Unit 1) and 267 (Unit 2) have been incorporated into the ITS submittal. This CTS change modified the Frequency of CTS 4.10.4.2. This change does not affect the ITS.	Pages 229, 230, and 234 of 357.

Change Description	Affected Pages
A self-identified change for CTS 3/4.1.2.3 has been made. CTS Amendments 281 (Unit 1) and 265 (Unit 2) have been incorporated into the ITS submittal. This CTS change adopted the allowances of TSTF-359 and affects CTS 3.1.2.3 Action e. This change does not affect the ITS.	Pages 287 and 289 of 357.

VOLUME 6

CNP UNITS 1 AND 2
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS SECTION 3.1
REACTIVITY CONTROL SYSTEMS

Revision 1

LIST OF ATTACHMENTS

1. ITS 3.1.1
2. ITS 3.1.2
3. ITS 3.1.3
4. ITS 3.1.4
5. ITS 3.1.5
6. ITS 3.1.6
7. ITS 3.1.7
8. ITS 3.1.8
9. Relocated/Deleted Current Technical Specifications (CTS)

ATTACHMENT 1

ITS 3.1.1, SDM

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.1.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F — A.2

LIMITING CONDITION FOR OPERATION

within the limits specified in the COLR — LA.1

LCO 3.1.1 — 3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4. with k_{eff} < 1.0 — A.3, A.4

ACTION:

not within limits — within 15 minutes — LA.1, L.1, L.2

ACTION A — With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

within limits — LA.1

SR 3.1.1.1 — 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). — See ITS 3.1.4; See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5. — See ITS 3.1.6

d. ~~Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.~~ — L.3

*See Special Test Exception 3.10.1. — A.4

ITS 3.1.1

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

MODE 2 with k_{eff} < 1.0 — M.1

SR 3.1.1.1 e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. ~~Reactor coolant system boron concentration,~~
2. ~~Control rod position,~~
3. ~~Reactor coolant system average temperature,~~
4. ~~Fuel burnup based on gross thermal energy generation,~~
5. ~~Xenon concentration,~~
6. ~~Samarium concentration, and~~
7. ~~Boron penalty (MODE 4 only).~~

LA.2

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within plus or minus 1% Delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading. — See ITS 3.1.2

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ITS 3.1.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - ~~TAVG LESS THAN OR EQUAL TO 200°F~~ A.2

LIMITING CONDITION FOR OPERATION

within the limits specified in the COLR LA.1

LCO 3.1.1 3.1.1.2 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.0% Delta k/k~~.

APPLICABILITY: MODE 5. LA.1

ACTION:

not within limits; within 15 minutes L.1; L.2

ACTION A With the SHUTDOWN MARGIN ~~less than 1.0% Delta k/k~~, ~~immediately~~ initiate ~~and continue~~ boration ~~at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

within limits LA.1

SR 3.1.1.1 4.1.1.2 The SHUTDOWN MARGIN shall be determined to be ~~greater than or equal to 1.0% Delta k/k~~:

~~a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ See ITS 3.1.4 ~~If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).~~ See ITS Chapter 1.0

SR 3.1.1.1 b. At least once per 24 hours ~~by consideration of the following factors:~~ LA.2

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration~~

~~6. Samarium concentration, and~~

~~7. Boron penalty.~~

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A.1

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A.1

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ITS 3.1.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN ~~- T_{AVG} GREATER THAN 200°F~~ A.2

LIMITING CONDITION FOR OPERATION

LCO 3.1.1 3.1.1.1 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.3% Delta k/k~~ within the limits specified in the COLR. LA.1

APPLICABILITY: MODES ~~1, 2*~~ with k_{eff} < 1.0, 3, and 4. A.3 A.4

ACTION: LA.1 L.1 L.2

ACTION A With the SHUTDOWN MARGIN ~~less than 1.3% Delta k/k~~ not within limits, ~~immediately~~ within 15 minutes initiate ~~and continue~~ boration ~~at greater than or equal to 24 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ~~greater than or equal to 1.3% Delta k/k~~ within limits: LA.1

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS 3.1.4 See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. See ITS 3.1.6

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.~~ L.3

~~*See Special Test Exception 3.10.1.~~ A.4

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ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

MODE 2 with k_{eff} < 1.0 — M.1

SR 3.1.1.1

e. When in ~~MODES 3 or 4~~, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and
7. Boron penalty (MODE 4 only).

LA.2

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within plus or minus 1% Delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

See ITS 3.1.2

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ITS 3.1.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN ~~- T_{avg} LESS THAN OR EQUAL TO 200°F~~ — A.2

LIMITING CONDITION FOR OPERATION

LCO 3.1.1 — 3.1.1.2 The SHUTDOWN MARGIN shall be ~~greater than or equal to 1.0% Delta k/k~~ within the limits specified in the COLR — LA.1

APPLICABILITY: MODE 5. — LA.1

ACTION:

ACTION A — With the SHUTDOWN MARGIN ~~less than 1.0% Delta k/k~~ not within limits, ~~immediately~~ within 15 minutes initiate ~~and continue~~ boration ~~at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent~~ until the required SHUTDOWN MARGIN is restored. — L.1, L.2

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 — 4.1.1.2 The SHUTDOWN MARGIN shall be determined to be ~~greater than or equal to 1.0% Delta k/k~~ within limits: — LA.1

~~a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ — See ITS 3.1.4 ~~If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).~~ — See ITS Chapter 1.0

SR 3.1.1.1 — b. At least once per 24 hours ~~by consideration of the following factors:~~ — LA.2

~~1. Reactor coolant system boron concentration,~~

~~2. Control rod position,~~

~~3. Reactor coolant system average temperature,~~

~~4. Fuel burnup based on gross thermal energy generation,~~

~~5. Xenon concentration,~~

~~6. Samarium concentration, and~~

~~7. Boron penalty.~~

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A.1

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A.1

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DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.1.1.1 provides SHUTDOWN MARGIN (SDM) requirements in MODES 1, 2, 3, and 4. CTS 3.1.1.2 provides SDM requirements in MODE 5. ITS 3.1.1 provides SDM requirements in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. This changes the CTS by combining the SDM requirements for MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. The change in Applicability for MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ are described in DOC A.3.

This change is acceptable because the requirements have not changed. Combining the Specifications is an editorial change. Any technical changes resulting from this combination are discussed in other DOCs. This change is designated as administrative because it does not result in a technical change to the CTS.

A.3 CTS 3.1.1.1 provides SDM requirements in MODES 1, 2, 3, and 4. CTS 4.1.1.1.1.b states that when in MODES 1 and 2 with $k_{eff} \geq 1.0$, verify that the control bank withdrawal is within the limits of Specification 3.1.3.5 (Unit 1) and Specification 3.1.3.6 (Unit 2), Control Rod Insertion Limits. ITS 3.1.1 is Applicable in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5. ITS 3.1.6 contains the control bank insertion requirements. This changes the CTS by dividing the SDM requirements and placing those applicable in MODE 2 with $k_{eff} < 1.0$ and MODES 3, 4, and 5 in ITS 3.1.1 and placing those applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ in the control bank Specifications.

The purpose of CTS 3.1.1.1 is to ensure that the SDM assumed in the accident analyses is available. When the reactor is critical, SDM is verified by ensuring that the control rods are within the control rod insertion limits. The Applicability Bases to ITS 3.1.1 states that in MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." This change is acceptable because the SDM requirements have not changed. Even though CTS 3.1.1.1 is applicable in MODES 1 and 2, the CTS Surveillances only requires the verification that control rod bank withdrawal is within the control rod insertion limits (i.e., CTS 3.1.3.5 (Unit 1) and CTS 3.1.3.6 (Unit 2)). The ITS also verifies SDM in MODES 1 and 2 by the rod insertion limits. Any changes to the rod insertion limit requirements are discussed in DOCs for those Specifications. This change is designated as administrative because it does not result in a technical change to the CTS.

A.4 The Applicability of CTS 3.1.1.1 is MODES 1, 2, 3, and 4 with a footnote for MODE 2 stating "See Special Test Exception 3.10.1." ITS 3.1.1 Applicability does not contain the footnote or a reference to the Special Test Exception.

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.1.1.1.1.e requires SDM to be determined to be within its limit every 24 hours when in MODES 3 and 4. ITS SR 3.1.1.1 requires SDM to be determined to be within its limit not only in MODES 3 and 4, but also in MODE 2 with $k_{eff} < 1.0$. This changes the CTS by expanding the applicability of the Surveillance to include MODE 2 with $k_{eff} < 1.0$.

The purpose of the CTS 4.1.1.1.1.e is to verify that sufficient SDM is available. CTS 4.1.1.1.1.b states that when the reactor is in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, SDM is verified by determining that the control rods are above the rod insertion limits. In MODE 2 with $k_{eff} < 1.0$, CTS 4.1.1.1.1.c verifies SDM by determining that the predicted critical position is within the rod insertion limits within 4 hours prior to achieving criticality. However, no CTS Surveillance requires a periodic verification of SDM when in MODE 2 with $k_{eff} < 1.0$. This change is acceptable because the ITS requires specific verification that the SDM is within the limit when in MODE 2 with $k_{eff} < 1.0$ on a periodic basis. This change is designated as more restrictive because it expands the conditions under which a Surveillance must be performed.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.1.1.1 and associated Action and CTS 4.1.1.1.1 require that the SDM be $\geq$ 1.3% Δk/k when in MODES 1, 2, 3, and 4. CTS 3.1.1.2 and associated Action and CTS 4.1.1.2 requires that the SDM be $\geq$ 1.0% Δk/k when in MODE 5. ITS 3.1.1 states that the SDM shall be within the limits of the COLR, ITS 3.1.1 ACTION A provides actions for when the SDM is not within the limits, and ITS SR 3.1.1.1 requires verification that the SDM is within limits. This changes the CTS by relocating the SDM limits, which must be confirmed on a cycle-specific basis, to the COLR.

The removal of these cycle-specific parameter limits from the Technical Specifications to the COLR is acceptable because the cycle-specific limits are developed or utilized under NRC-approved methodologies which will ensure that the Safety Limits are met. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the SDM requirement. The methodologies used to develop the parameters in the COLR have obtained prior approval by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.1.1.1.1.e and 4.1.1.2.b require determination that the SDM is within limits, and specifically require the consideration of the following factors: reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and boron penalty (MODES 4 and 5 only). ITS SR 3.1.1.1 requires determination that SDM is within limits, but does not describe the factors that must be considered in the calculation. This information is relocated to the Bases. This changes the CTS by removing details on how the SDM calculation is performed from the Specifications and placing the information in the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the SDM be within limits. The details of how SDM is calculated does not need to appear in the Specification in order for the requirement to apply. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

<u>LESS RESTRICTIVE CHANGES</u>

L.1 *(Category 3 – Relaxation of Completion Time)* CTS 3.1.1.1 and CTS 3.1.1.2 Actions state that when the SDM is less than the applicable limit, boration must be initiated immediately. ITS 3.1.1 ACTION A states that when SDM is not within limits, boration must be initiated within 15 minutes. This changes the CTS by relaxing the Completion Time from "immediately" to 15 minutes.

The purpose of CTS 3.1.1.1 and CTS 3.1.1.2 Actions is to restore the SDM to within its limit promptly. This change is acceptable because the Completion Time

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The ITS Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. In addition, the ITS Bases for the ACTION state that boration must be initiated promptly. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.1 and CTS 3.1.1.2 Actions state that when the SDM is not within the applicable limits, boration must be initiated and continued at ≥ 34 gpm of a solution containing ≥ 6,550 ppm boron or equivalent until the required SDM is restored. ITS 3.1.1 ACTION A states that with the SDM not within limits, initiate boration to restore SDM to within limits. This changes the CTS by eliminating the specific values of flow rate and boron concentration that must be used to restore compliance with the LCO.

The purpose of the CTS 3.1.1.1 and CTS 3.1.1.2 Actions is to restore the SDM to within its limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Removing the specific values of flow rate and boron concentration from the CTS Action provides flexibility in the restoration of the SDM and eliminates conflicts between the SDM value and the specific boration values in the CTS Action. As stated in the ITS Bases for ACTION A, "In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid tank or the refueling water storage tank. The operator should borate with the best source available for the unit conditions." Specifying a minimum flow rate and concentration in the ACTION may not accomplish the objective of raising the RCS boron concentration as soon as possible. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.1.1.1.d requires verification that SDM is within its limit, "Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5 (Unit 1) and Specification 3.1.3.6 (Unit 2)." The ITS does not contain a similar requirement.

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

The purpose of CTS 4.1.1.1.1.d is to verify the core design predictions by determining the SDM with the control rods at the insertion limits. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify the LCO is within limit. The core design predictions, such as rod worth, boron worth, and critical boron concentration, are verified during the startup physics test program. Thus, the SDM continues to be verified in a manner and at a Frequency necessary to give confidence that the parameter is within limit. The critical boron concentration is verified periodically by ITS 3.1.2. Therefore, the core design parameters upon which SDM relies are verified before exceeding 5% RATED THERMAL POWER after each refueling outage. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

SDM
3.1.1

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1.1, LCO 3.1.1.2

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

3.1.1.1 Action, 3.1.1.2 Action

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limits.	A.1	Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

4.1.1.1.1, 4.1.1.1.e, 4.1.1.2, 4.1.1.2.b

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within limits.	24 hours

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

None.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

SDM
B 3.1.1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

INSERT 1 making and two independent

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. ①

INSERT 2

INSERT 3

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

transients

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System together with the boration system provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions. ② ② ②

making and holding

INSERT 4

INSERT 5

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration. ③ ③

MODE 3, 4, 5, or 6

INSERT 6

2 **INSERT 1**

Plant Specific Design Criterion (PSDC) 27

2 **INSERT 2**

provided. According to PSDC 28 (Ref. 1), the reactivity controls must be

2 **INSERT 3**

from any hot standby or hot operating condition. According to PSDC 29 (Ref. 1), one of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast enough to prevent exceeding acceptable fuel damage limits. SDM should assure subcriticality with the most reactive rod cluster control assembly fully withdrawn. According to PSDC 30 (Ref. 1), the reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies, and shall be capable of limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

2 **INSERT 4**

along with the shutdown and control rods

3 **INSERT 5**

When the unit is in MODE 1 or MODE 2 with the reactor critical,

3 **INSERT 6**

When the unit is in MODE 2 with the reactor subcritical, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the estimated critical control bank position.

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anticipated operational transients

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis. 2 1

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events ; 5

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident) ; and 2 5

200

anticipated operational transients

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1. 3 2 2

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

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BASES

APPLICABLE SAFETY ANALYSES (continued)

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition; and

~~c. Startup of an inactive reactor coolant pump (RCP), and~~

c. ~~d.~~ Rod ejection.

(5)

Each of these events is discussed below.

INSERT 7

~~In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.~~ (2)

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level ~~trip or a high pressurizer pressure~~ trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits. (2)

(3)

INSERT 8

~~The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.~~ (2)

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). ~~Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.~~ (4)

2 **INSERT 7**

The boron dilution analysis covers operation during shutdown, refueling, startup, and power operation. The purpose of the analysis is to show that, from initiation of the event, sufficient time is available to allow the operator to determine the cause of the dilution and to take corrective action before the SDM is lost.

2 **INSERT 8**

, overtemperature ΔT, overpower ΔT, or pressurizer water level

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SDM
B 3.1.1

BASES

LCO	SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration. The MSLB (Ref. 1) and the boron dilution (Ref. 4) ~~accidents~~ analyses are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 5). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable. (2) (2) (2)
APPLICABILITY	In MODE 2 with k_{eff} < 1.0 and in MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with keff ≥ 1.0, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6. (3)
ACTIONS	A.1 If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the ~~borated~~ refueling water storage tank. The operator should borate with the best source available for the ~~plant~~ unit conditions. (2) In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of 34 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 44 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of 34 gpm and 6550 ppm represent (6) (6) (6)

BASES

ACTIONS (continued)

INSERT 8a ②

typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $K_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth. ⑦ ③

INSERT 9

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects: ⑤

a. RCS boron concentration ;

b. Control bank position ; ② ⑤

c. RCS average temperature ;

d. Fuel burnup based on gross thermal energy generation ; ⑤

e. Xenon concentration ;

f. Samarium concentration and ;

g. Isothermal temperature coefficient (ITC). INSERT 10 ②

INSERT 11

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. ②

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26. INSERT 12 ①

2. U FSAR, Chapter 15. (14) ② ⑥

2 **INSERT 8a**

the current licensed values

2 **INSERT 9**

MODE 2 with $k_{eff} < 1.0$, and

2 **INSERT 10**

h. Boron penalty (MODES 4 and 5 only).

2 **INSERT 11**

The boron penalty must be applied in MODES 4 and 5 since all reactor coolant pumps may be stopped in these MODES. This extra amount of boron ensures that minimum response times are met for the operator to diagnose and mitigate an inadvertent boron dilution event prior to loss of SDM.

1 **INSERT 12**

UFSAR, Section 1.4.5.

BASES

REFERENCES (continued) (u) Section 14.2.5

3. FSAR, ~~Chapter [15]~~. (2) (6)

INSERT 13 (2)

(5) ~~(4)~~ 10 CFR 100.

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2 **INSERT 13**

4. UFSAR, Section 14.1.5.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1 BASES, SHUTDOWN MARGIN (SDM)

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Changes are made to the Background section to be consistent with the discussion in the Applicability section.

4. The Applicable Safety Analyses discussion states that SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). It also says that even though SDM is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of the accident analysis assumptions. The additional sentence has been deleted. The NRC Final Policy Statement on Technical Improvements of July 22, 1993 (58 FR 39132) states that process variables captured by Criterion 2 are not limited to only those directly monitored and controlled from the control room. It also states that Criterion 2 includes other features or characteristics that are specifically assumed in Design Basis Accident and Transient analyses even if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors). Since the Final Policy Statement provides guidance on which types of parameters satisfy Criterion 2, there is no reason to duplicate these words in the CNP ITS.

5. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

6. The brackets have been removed and the proper plant specific information/value has been provided.

7. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

There are no specific NSHC discussions for this Specification.

ATTACHMENT 2

ITS 3.1.2, Core Reactivity

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL — Core Reactivity — A.2

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION — Add proposed LCO 3.1.2 — A.2

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k. — See ITS 3.1.1

APPLICABILITY: MODES 1, 2*, 3, and 4. — L.1

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored. — See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. — See ITS 3.1.4 If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). — See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5. — See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5. — See ITS 3.1.1

Add proposed ACTIONS A and B — L.2

*See Special Test Exception 3.10.1. — See ITS 3.1.1

ITS 3.1.2

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and
7. Boron penalty (MODE 4 only).

See ITS 3.1.1

M.1

Prior to entering MODE 1 after refueling and

L.3

LA.1

SR 3.1.2.1 — 4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within plus or minus 1% Delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

ITS 3.1.2

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

Core Reactivity — A.2

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.1.2 — A.2

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k. — See ITS 3.1.1

APPLICABILITY: MODES 1, 2*, 3, and 4. — L.1

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored. — See ITS 3.1.1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. — See ITS 3.1.4

If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). — See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. — See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6. — See ITS 3.1.1

Add proposed ACTIONS A and B — L.2

*See Special Test Exception 3.10.1. — See ITS 3.1.1

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ITS 3.1.2

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and
7. Boron penalty (MODE 4 only).

See ITS 3.1.1

M.1

Prior to entering MODE 1 after refueling and

L.3

SR 3.1.2.1 4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within plus or minus 1% Delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

LA.1

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DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 4.1.1.1.2 requires the overall core reactivity balance be compared to predicted values to demonstrate agreement within +/- 1% $\Delta k/k$. However, this Surveillance is currently part of the SHUTDOWN MARGIN Specification. A new LCO, ITS LCO 3.1.2, requires the measured core reactivity to be within +/- 1% $\Delta k/k$ of predicted values. This changes the CTS by having a separate Specification for the Core Reactivity requirement.

This change is acceptable because the requirements have not changed. Converting the requirement from a Surveillance in the SHUTDOWN MARGIN Specification to an LCO is consistent with the ITS format and content guidance. Any technical changes resulting from this change are discussed in other DOCs. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 ITS SR 3.1.2.1 requires the measured core reactivity to be determined to be within +/- 1% $\Delta k/k$ of the predicted value prior to entering MODE 1 after each refueling. The CTS does not contain a similar requirement. This changes the CTS by adding an additional performance requirement for the core reactivity balance SR.

This change is acceptable because it requires a test that demonstrates agreement between the core design and the core design predictions prior to raising core power above 5% after each refueling. This verification, which is currently performed as part of the startup physics testing program, gives additional confidence that the core design is acceptable for operation at full power. This change is designated as more restrictive because it adds a Surveillance Requirement that does not appear in the CTS.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.1.1.1.2 requires comparison of the actual and predicted core reactivity balance and specifically requires consideration of at least those factors stated in Specification 4.1.1.1.1.e. CTS 4.1.1.1.1.e requires determination of SDM and requires the consideration of the following factors: reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. ITS SR 3.1.2.1 requires comparison of the actual and predicted core reactivity, but does not describe the factors that must be considered in the calculation. This information is relocated to the Bases. This changes the CTS by removing details on how the core reactivity balance comparison calculation is performed from the CTS and placing the information in the Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the core reactivity balance comparison be within +/- 1% $\Delta k/k$. The details of how this comparison is calculated does not need to appear in the Specification in order for the requirement to apply. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.1.1.1 is applicable in MODES 1, 2, 3, and 4. ITS 3.1.2 is applicable in MODES 1 and 2. This changes the CTS by reducing the applicable MODES in which the core reactivity requirement must be met.

The purpose of CTS Surveillance 4.1.1.1.2 is to verify the core design by comparing the actual and predicted core reactivity. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The core reactivity balance can only be determined when the reactor is critical (MODES 1 and 2). Additionally, the Surveillance Frequency is once per 31 EFPD, which only continues to accrue when the reactor is critical. Therefore, reducing the applicable MODES from MODES 1, 2, 3, and 4 to MODES 1 and 2 does not result in a reduction of the verification of this important measure of core design accuracy. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.1.1.1 does not contain Actions to follow if the core reactivity balance Surveillance is not met. If the core reactivity balance Surveillance was not met, LCO 3.0.3 would be entered. LCO 3.0.3 requires the plant to be in MODE 3 within 7 hours, MODE 4 within 13 hours, and MODE 5 within 37 hours. ITS 3.1.2 contains ACTIONS to follow if the core reactivity balance LCO is not met. If the LCO is not met, 7 days is provided to re-evaluate the core design and safety analysis, to determine that the reactor core is acceptable for continued operation, and to establish appropriate operating restrictions and SRs. If these actions are not completed within the 7 days, the plant must be in MODE 3 within 6 hours. This changes the CTS by providing 7 days to evaluate and provide compensatory measures for not meeting the core reactivity balance requirement and then requiring entry into MODE 3 instead of requiring an immediate shutdown and entry into MODE 5.

The purpose of CTS 4.1.1.1.2 is to verify the accuracy of the core design by comparing the predicted and actual core reactivity throughout core life. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Should the core reactivity balance requirement not be met, time is required to determine the cause of the disagreement and what adjustments may be needed to the operating conditions of the core. The startup physics testing program is used to verify most of the critical core design parameters, such as control rod worth, boron worth, and moderator temperature coefficient. In addition, there is considerable conservatism in the application of these values in the accident analysis. Therefore, allowing a time to evaluate the difference and make any adjustments to the operational controls is acceptable. The 7 day Completion Time is reasonable considering the complexity of the evaluations and the time to meet administrative requirements, such as 10 CFR 50.59 safety evaluation preparation and approval. If it cannot be determined within 7 days that the core is acceptable for continued operation, the unit must be shutdown. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Surveillance 4.1.1.1.2 requires the overall core reactivity balance to be compared with the predicted value once per 31 EFPD. The CTS also requires the predicted reactivity values to be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. ITS SR 3.1.2.1 also allows the measured core reactivity to be compared to the predicted values every 31 EFPD, but the ITS SR is only required after 60 EFPD of core burnup. The ITS also requires the adjustment of the predicted values to the actual values prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. This changes the CTS by not requiring the periodic, at-power core reactivity comparison until core burnup reaches 60 EFPD.

DISCUSSION OF CHANGES
ITS 3.1.2, CORE REACTIVITY

The purpose of CTS 4.1.1.1.2 is to verify the agreement between the actual and predicted core reactivity. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The CTS and the ITS requires the predicted core reactivity values to be normalized to the actual values prior to exceeding 60 EFPD of core burnup. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after fuel loading, is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. In addition, a new Frequency has been added to ensure core reactivity is within limits prior to entering MODE 1 after each refueling (see DOC M.1). This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

DOC A.2

LCO 3.1.2 The measured core reactivity shall be within ± 1% $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

DOC L.1

ACTIONS

DOC L.2

DOC L.2

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Measured core reactivity not within limit.	A.1	Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7 days
	<u>AND</u>		
	A.2	Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	- NOTE - The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.	must
	Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Prior to entering MODE 1 after each refueling AND - NOTE - Only required after 60 EFPD 31 EFPD thereafter

4.1.1.1.2

①

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2, CORE REACTIVITY

1. ISTS SR 3.1.2.1 has been modified to be consistent with the current licensing basis. The predicted reactivity values must (not may) be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each refueling. This is necessary to ensure there is a benchmark for the design calculations. This change is also consistent with the ISTS Bases.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

INSERT 1

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore reactivity balance is used as a measure of the predicted versus measured core reactivity during ~~power~~ operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

and startup

① ②

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever

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1 INSERT 1

According to Plant Specific Design Criterion (PSDC) 28 (Ref. 1), the reactivity controls provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. According to PSDC 29 (Ref. 1), one of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast enough to prevent exceeding acceptable fuel damage limits. SDM should assure subcriticality with the most reactive rod cluster control assembly fully withdrawn. According to PSDC 30 (Ref. 1), the reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies, and shall be capable of limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

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BASES

BACKGROUND (continued)

neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

adjusted

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity ~~is~~ decreasing. As the fuel depletes, the RCS boron concentration is reduced to ~~decrease negative reactivity and~~ maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated. (2)

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses. unit (2)

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity. unit (2)

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown

BASES

APPLICABLE SAFETY ANALYSES (continued)

curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4,

BASES

APPLICABILITY (continued)

and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

BASES

ACTIONS (continued)

B.1

INSERT 2 (4)

~~If the core reactivity cannot be restored to within the 1% Δk/k limit~~, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. (2)

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including RCS boron concentration (2) control rod position, ~~moderator~~ RCS average (2) temperature, ~~fuel temperature~~, fuel ~~depletion~~ (2) INSERT 3, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.~~ UFSAR, Section 1.4.5 (1)
2. ~~F~~SAR, Chapter ~~15~~ 14 (2) (3)

4 **INSERT 2**

If any Required Action and associated Completion Time is not met

2 **INSERT 3**

burnup based on gross thermal energy generation

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2 BASES, CORE REACTIVITY

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Changes made to be consistent with the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.2, CORE REACTIVITY

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 3.1.3, Moderator Temperature Coefficient

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.1.3

A.1

ITS

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

LCO 3.1.3 3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than or equal to the limit shown in Figure 3.1-2. upper A.2

APPLICABILITY: BOL limit - MODES 1 and 2* only# A.3
EOL limit - MODES 1, 2 and 3 only# lower

ACTION: upper A.2

a. With the MTC more positive than the BOL limit specified in the COLR: MODE 2 with k_{eff} < 1.0 A.4

ACTION A 1. Establish and maintain ~~control rod withdrawal limits sufficient to restore~~ the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. ~~These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.5.~~ A.5
ACTION B

~~2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.~~ L.1

~~3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~ L.2

ACTION C b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours. lower A.2

Applicability ~~* With K_{eff} greater than or equal to 1.0~~
~~# See Special Test Exception 3.10.4~~ A.3

ITS

A.1

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

SR 3.1.3.1 a) The MTC shall be measured and compared to the EOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

upper

A.2

SR 3.1.3.2 b) The MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. The measured value shall be compared to the 300 ppm surveillance limit specified in the COLR. In the event this comparison indicates that the MTC will be more negative than the EOL limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit.

lower

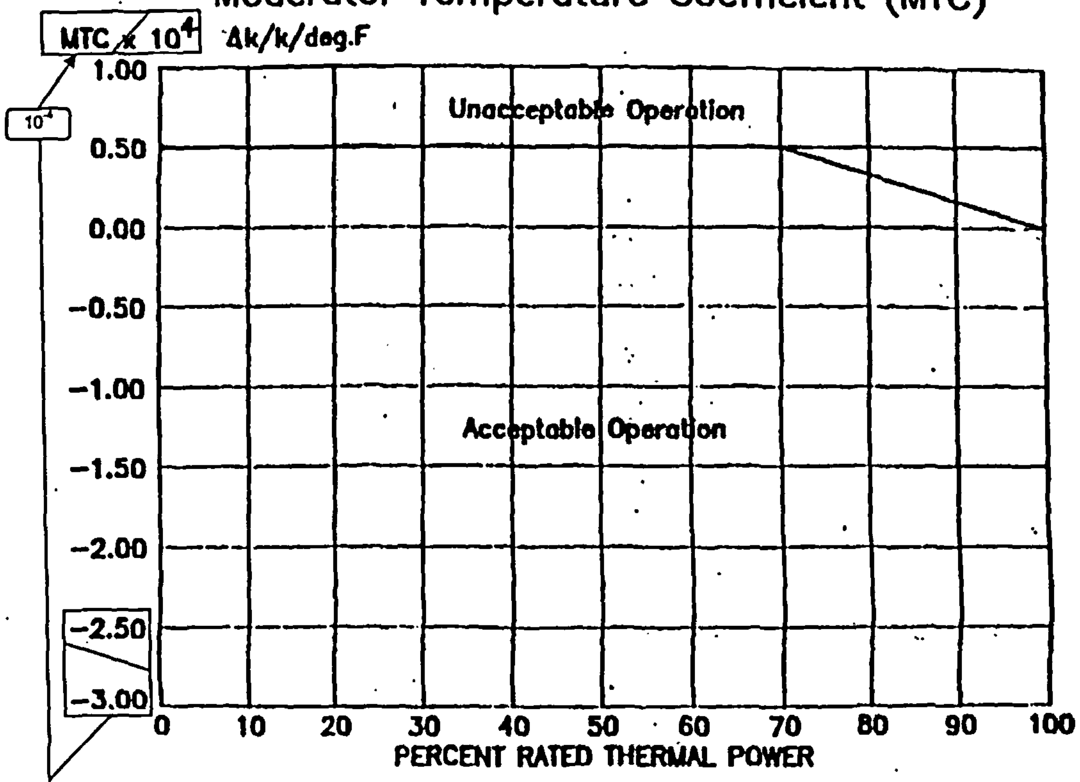
A.2

L.3

Figure 3.1.3-1

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



A.1

A.6

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ITS 3.1.3

ITS

A.1

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

LCO 3.1.3 — 3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than or equal to the limit shown in Figure 3.1-2. upper — A.2

APPLICABILITY: ~~BOL~~ Limit - MODES 1 and 2* only~~#~~ — A.3
~~EOL~~ Limit - MODES 1, 2 and 3 only~~#~~ lower — A.2

ACTION:

a. With the MTC more positive than the ~~BOL~~ upper limit specified in the COLR: MODE 2 with k_{eff} < 1.0 — A.4

ACTION A — 1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limit within 24 hours or

ACTION B — be in HOT STANDBY within the next 6 hours. ~~These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.~~ — A.5

2. ~~Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.~~ — L.1

3. ~~Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~ — L.2

ACTION C — b. With the MTC more negative than the ~~EOL~~ lower limit specified in the COLR, be in HOT SHUTDOWN within 12 hours. — A.2

Applicability — * With K_{eff} greater than or equal to 1.0
~~# See Special Test Exception 3.10.3~~ — A.3

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ITS 3.1.3

A.1

ITS

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

upper — A.2

SR 3.1.3.1 a) The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

SR 3.1.3.2 b) The MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. The measured value shall be compared to the 300 ppm surveillance limit specified in the COLR. In the event this comparison indicates that the MTC will be more negative than the EOL limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit.

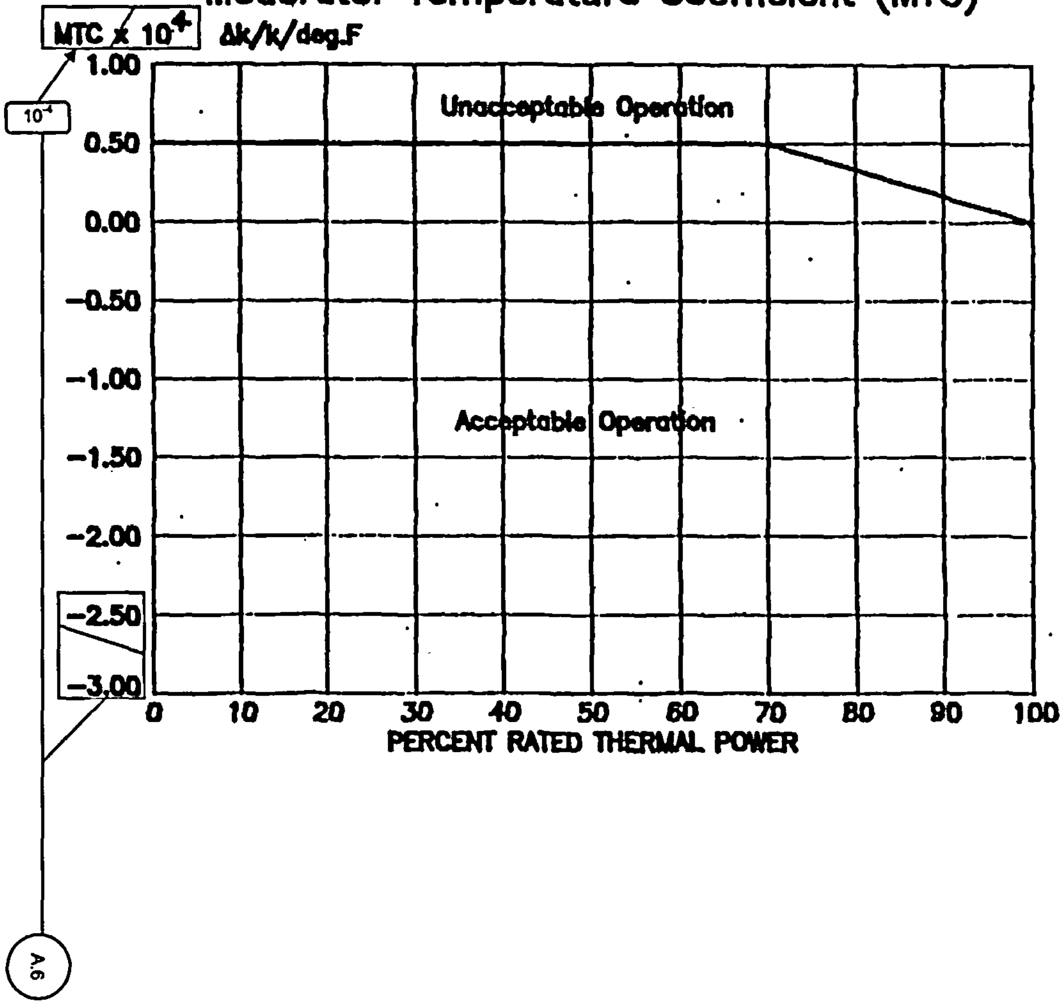
L.3

lower — A.2

Figure 3.1.3-1

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



A.1

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DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.1.1.4 refers to the BOL MTC limit and the EOL MTC limit. ITS 3.1.3 refers to these values as the upper MTC limit and lower MTC limit, respectively.

This change is acceptable because the requirements have not changed. The BOL MTC value is the most positive, upper limit and the EOL MTC value is the most negative, lower limit. The terminology used in the ITS is an editorial preference selected for consistency with that used in NUREG-1431. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A.3 The Applicability of CTS 3.1.1.4 is modified by footnote # stating "See Special Test Exception 3.10.4." ITS 3.1.3 Applicability does not contain the footnote or a reference to the Special Test Exception.

The purpose of the footnote reference is to alert the reader that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A.4 CTS 3.1.1.4 Action a.1 states that if the MTC is more positive than the BOL (i.e., upper) limit, control rod withdrawal limits must be imposed within 24 hours or the unit must be in HOT STANDBY within the next 6 hours. ITS 3.1.3 ACTION A states that with the MTC not within the upper limit, establish administrative control rod withdrawal limits within 24 hours or ACTION B requires the unit to be in MODE 2 with $k_{eff} < 1.0$ within the next 6 hours. This changes the CTS by requiring the plant to be in MODE 2 with $k_{eff} < 1.0$ instead of HOT SHUTDOWN (i.e., MODE 3).

This change is acceptable because the requirements have not changed. In accordance with CTS LCO 3.0.1, Actions are only required to be followed while in the MODE of applicability. The CTS upper MTC limit is applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. Therefore, under the CTS, the unit does not have to enter MODE 3 because the applicability of the Action ends when in MODE 2 with $k_{eff} < 1.0$. As a result, there is no difference between the CTS and ITS requirements. This change is designated as administrative because it does not result in a technical change to the CTS.

A.5 CTS 3.1.1.4 Action a.1 states that if the MTC is more positive than the BOL limit, then control rod withdrawal limits must be established. It also states that these

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DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.5 (Unit 1) and Specification 3.1.3.6 (Unit 2). The ITS does not include this sentence.

This change is acceptable because the requirements have not changed. The CTS reference to Specification 3.1.3.5 (Unit 1) and Specification 3.1.3.6 (Unit 2) is an "information only" statement that neither adds, eliminates, or modifies requirements. The ITS convention is to not include these types of statements. This change is designated as administrative because it does not result in a technical change to the CTS.

A.6 CTS Figure 3.1-2 provides the maximum upper limit for MTC from 0% to 100% RATED THERMAL POWER (RTP). The Figure indicates that the value for MTC can vary from -3.00 to 1.00 x 10^4 Δk/k/°F. ITS Figure 3.1.3-1 includes the same curve however the range has changed to -2.00 to 1.00 (x 10^{-4} Δk/k/°F). This changes the CTS by using the correct exponential (10^4 in the CTS to 10^{-4} in the ITS) and changing the range for MTC.

This change is acceptable because the requirements have not changed. The maximum upper limit for MTC when < 70% RTP is 0.50 10^{-4} Δk/k/°F and the maximum upper limit at 100% RTP is zero. This change is consistent with how similar values are presented in the ITS. Since this curve only provides the maximum upper limit there is no need to provide a wide range from -3.00 x 10^{-4} Δk/k/°F to 1.00 x 10^{-4} Δk/k/°F. The lower value of -2.00 x 10^{-4} Δk/k/°F is sufficient. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 5 – Deletion of Surveillance Requirement)* CTS 3.1.1.4 Action a.2 states that if the measured MTC is more positive than the BOL (i.e., upper) limit, then the control rod withdrawal limits established in Action a.1 must be maintained until subsequent measurement verifies that the MTC has been restored to within its limits for the all rods withdrawn condition. ITS 3.1.3 does not contain a requirement that the control rod withdrawal limits be maintained until MTC is confirmed to be within its limit by measurement. However, ITS LCO 3.0.2 states that the Required Actions shall be followed until the LCO is met or no longer applicable. The ITS Bases state that physics calculations may be used to determine the time in cycle life at which the calculated MTC will meet the LCO requirement, and at this point in core life the condition may be exited and the control rod withdrawal limits removed. This changes the CTS by eliminating the Surveillance Requirement verifying the MTC to be within its limit before removing the control rod withdrawal limits.

The purpose of CTS 3.1.1.4 Action a.2 is to ensure that the additional operational restrictions required to maintain the MTC within the assumptions in the safety analyses are maintained until the MTC value without the restrictions is within the LCO limits. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analyses. Thus, appropriate values continue to be tested in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. The measurement of the MTC, boron endpoint, and control rod worth prior to entering MODE 1 is sufficient to verify the nuclear design so that it can be accurately predicted when the all rods out, full power equilibrium MTC is within the LCO limit. Performing another measurement of beginning of cycle MTC to confirm this prediction is not necessary to give confidence that MTC is within its limit. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.2 *(Category 8 – Deletion of Reporting Requirements)* CTS 3.1.1.4 Action a.3 requires that a Special Report be prepared and submitted to the NRC within 10 days if the measured MTC is more positive than the BOL limit. The Special Report must describe the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. ITS 3.1.3 does not include this requirement.

The purpose of CTS 3.1.1.4 Action a.3 is to provide information describing the event to the NRC. This change is acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. A Licensee Event Report is required to be submitted by 10 CFR 50.73(a)(2)(i)(B) for any operation or condition outside of the plant's Technical Specifications. Therefore, a report to the NRC is still required. This change is designated as less restrictive because reports that would be submitted under the CTS will not be required under the ITS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.1.1.4.b) requires MTC to be determined to be within limits. MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. The measured value shall be compared to the 300 ppm Surveillance limit specified in the COLR. In the event this comparison indicates that the MTC will be more negative than the EOL (i.e., lower) limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit. ITS

DISCUSSION OF CHANGES
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

SR 3.1.3.2 requires the verification that MTC is within the lower limit. The first proposed Frequency is once each cycle within 7 effective full power days (EFPD) after reaching an equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. The second Frequency is 14 EFPD thereafter if MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR until the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. This changes the CTS by eliminating the requirement to verify that MTC is met at least once per 14 EFPD if the measured MTC at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR.

The purpose of CTS 4.1.1.4.b) is to periodically verify that the MTC EOL (i.e., lower) limit is within limit if the 300 ppm Surveillance limit in the COLR is not met. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of assurance that the MTC lower limit will not be exceeded. This will help ensure that the MTC EOL (lower) limit is not exceeded for the remainder of the cycle. The new 60 ppm Surveillance limit for RTP-ARO boron concentration of ≤ 60 ppm will be incorporated into the COLR. This new limit is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, then the MTC lower limit will not be exceeded because of the gradual manner in which MTC changes with core burnup. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.4

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ~~[~~ ≤ [ ] Δk/~~k~~/°F at hot ~~zero~~ power ~~]~~ that specified in Figure 3.1.3-1~~]~~. 1

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit,
MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
Action a.1	A. MTC not within upper limit.	A.1	Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
Action a.1	B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
Action b	C. MTC not within lower limit.	C.1	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.1.1.4.a)	SR 3.1.3.1	Verify MTC is within upper limit.	Prior to entering MODE 1 after each refueling

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.3.2	~~- NOTES -~~ ~~1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.~~ ~~2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle.~~ ~~3. SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR.~~ Verify MTC is within lower limit.	~~Once each cycle~~

4.1.1.4.b)

2

INSERT 1

2 INSERT 1

Once each cycle within 7 effective full power days (EFPD) after reaching an equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm

AND

14 EFPD thereafter if MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR until the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR

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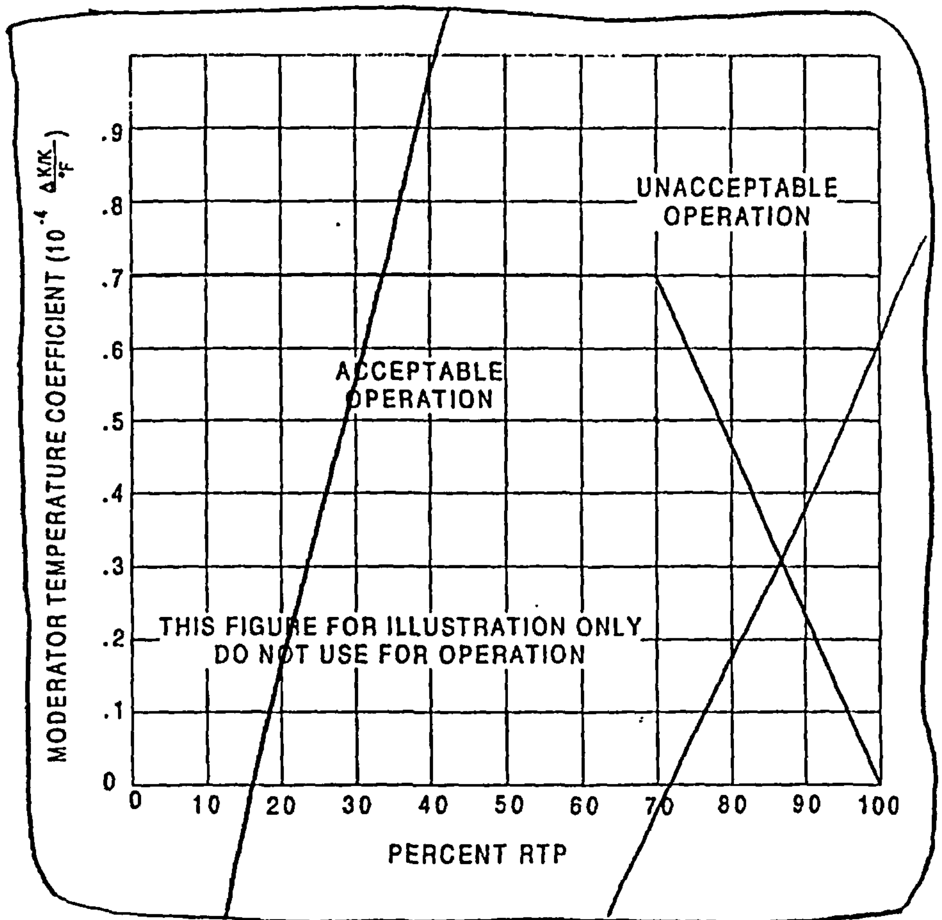
MTC
3.1.3

CTS

Figure 3.1-2



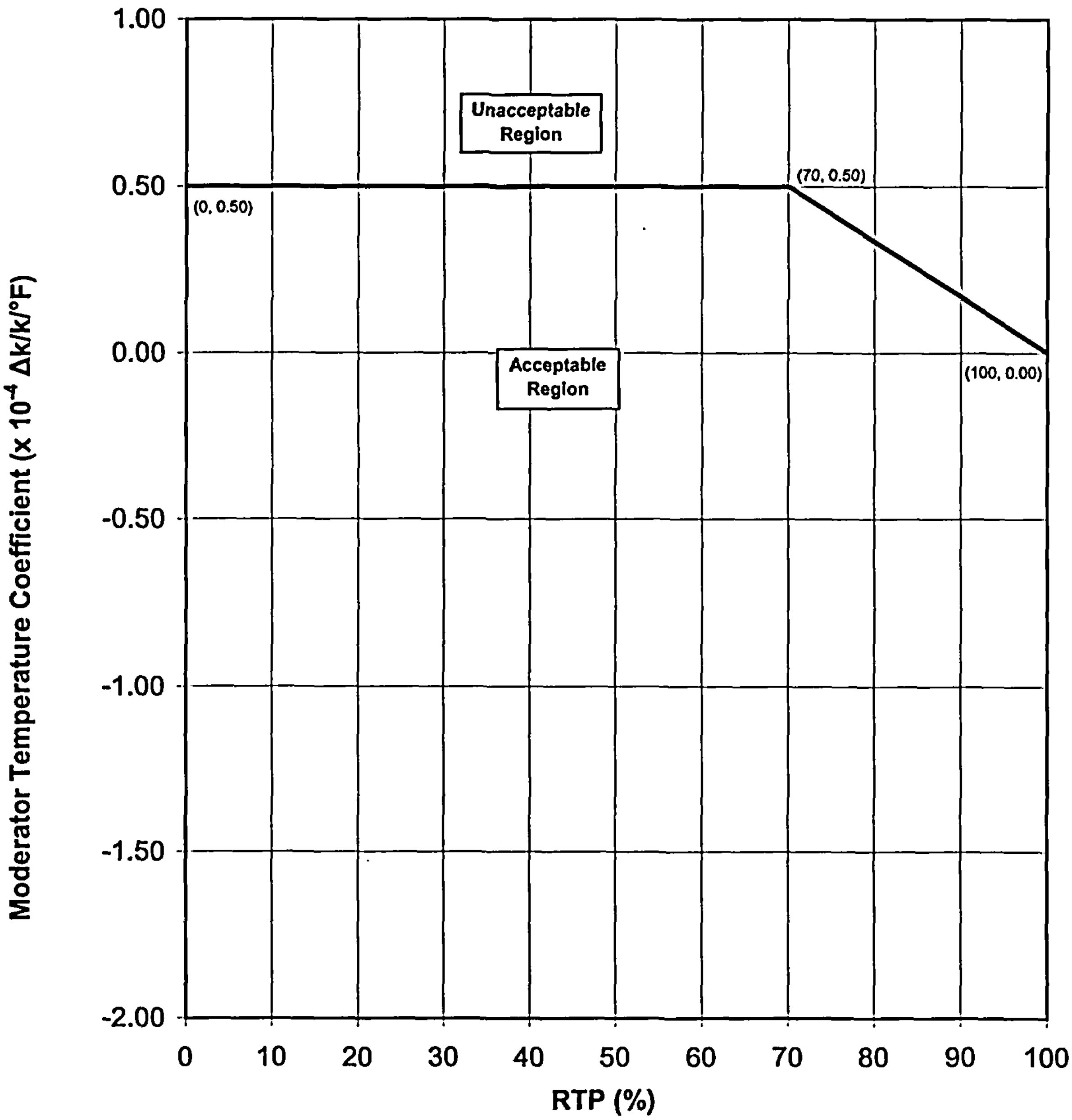
INSERT 2 (3)

Figure 3.1.3 - 1 (page 1 of 1)
Moderator Temperature Coefficient ~~vs~~ versus ~~Rated Thermal Power~~ Percent (4)

(4)

3

INSERT 2



1.00
0.50
0.00
-0.50
-1.00
-1.50
-2.00
Unacceptable Region
Acceptable Region
(0, 0.50)
(70, 0.50)
(100, 0.00)
0
10
20
30
40
50
60
70
80
90
100
RTP (%)
Moderator Temperature Coefficient (x 10^-4 Δk/k/°F)

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The ISTS SR 3.1.3.2 Surveillance Notes have been combined into the Surveillance Frequency for clarity and consistency. ISTS SR 3.1.3.2 Note 1 modifies the Frequency of "Once each cycle," and it would be clearer to avoid the use of a specific Note in the Surveillance column and just include the words in the Frequency. Thus, the Frequency in ITS SR 3.1.3.2 is modified to state "Once each cycle within 7 effective full power days (EFPD) after reaching an equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm." ISTS SR 3.1.3.2 Note 2 provides an additional Surveillance Frequency that must be met if certain conditions exist following performance of the initial Surveillance (i.e., the Surveillance performed to meet the ISTS SR 3.1.3.2 Frequency as modified by ISTS SR 3.1.3.2 Note 1). ISTS SR 3.1.3.2 Note 3 further modifies ISTS SR 3.1.3.2 Note 2, stating that the additional Surveillance Frequency required by Note 2 does not have to be met under certain conditions. Instead of a Note providing an additional Frequency, with a further Note modifying the additional Frequency, a new Frequency has been added stating "14 EFPD thereafter if MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR until the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR." This Frequency is connected to the first Frequency by use of the logical connector "AND." This manner of writing the second "conditional" type of Frequency, which only starts upon completion of a previous Surveillance Frequency, is consistent with the manner in which these type of Frequencies are formatted in other Specifications. For example, this specific "conditional" Frequency is shown in ITS 1.4, "Frequency," Example 1.4-2. Some specific instances in the ISTS where this "conditional" type of Frequency is used is shown in ISTS SR 3.1.2.1, ISTS SR 3.2.1.1, ISTS SR 3.2.1.2, ISTS SR 3.2.2.1, ISTS SR 3.2.3.2, and ISTS SR 3.2.3.3. In all of these Surveillances, the second Frequency does not start until after the first Frequency is met. Therefore, this modification to include the Surveillance Notes in the Frequency results in a clearer understanding of the required Frequencies and is consistent with the manner in which other similar type of Surveillance Frequencies are written.

3. The appropriate MTC vs. THERMAL POWER CURVE has been included consistent with the current licensing basis.

4. Typographical/grammatical error corrected.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

Reference (1)

BACKGROUND

According to ~~GDC 11 (Ref.~~ 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

The

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. ~~Both initial and~~ reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the ~~plant~~ unit accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles ~~that are designed to achieve high burnups or that have changes to other characteristics~~ are evaluated to ensure that the MTC does not exceed the EOC limit. (1) (1) (1)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the U FSAR accident and transient analyses. (1)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

BASES

BACKGROUND (continued)

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup. INSERT 1 (1)

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2 and 3) (1) (7)

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events. (1)

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3). (1)

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature. (1)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. *The most conservative combination appropriate to the accident is then used for the analysis (Ref. 1).* (1)

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. *The measured value may be extrapolated to project the EOC value,* in order to confirm reload design predictions.

1 INSERT 1

but also to a significant extent from the effects of buildup of plutonium and fission products

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BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration. 6

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded. Near 1

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met. (lower) 2

(upper) The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule. (Figure 3.1.3-1) 4

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is 3

BASES

APPLICABILITY (continued)

not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

upper

If the BOC MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits. (2)

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems. (1)

unit

C.1

lower

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours. (2) (2) (1)

lower unit

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems. (1)

unit

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC upper MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks. (2)

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC lower LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met. (2)

SR 3.1.3.2 is modified by three Notes that includes the following requirements:

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

b. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOC limit will not be exceeded

INSERT 2 (4)

4

INSERT 2

Performing the Surveillance once each cycle within 7 effective full power days (EFPD) after reaching an equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm is soon enough after the performance of SR 3.1.3.1 to ensure the lower limit will not be exceeded since the MTC changes after initial performance are gradual with core depletion and boron concentration reduction.

The Frequency of 14 EFPD thereafter, if MTC is more negative than 300 ppm Surveillance limit (not LCO limit) specified in the COLR or until the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR, is adequate for monitoring the change in MTC with core bumup since changes to MTC are relatively slow. The Surveillance limit for MTC at a RTP-ARO boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the lower limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

because of the gradual manner in which MTC changes with core burnup.

INSERT 3 1

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 11.~~ INSERT 4

3 → 2. 4 FSAR, Chapter [15]. 14 2 1 5

4 → 3. WCAP 9272-NP-A, "Westinghouse Reload Safety Evaluation Methodology," ~~July 1985~~. March 1978 1

~~4. FSAR, Chapter [15].~~ 1

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1

INSERT 3

UFSAR, Section 3.3.1 (Unit 1) and 3.3.1.2 (Unit 2).

1

INSERT 4

2. UFSAR, Section 1.4.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3 BASES, MODERATOR TEMPERATURE COEFFICIENT (MTC)

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The ISTS Bases variously refer to the "upper MTC limit," the "BOC MTC limit," the "lower MTC limit," and the "EOC MTC limit." References to the BOC and EOC MTC limit are eliminated and "upper" and "lower" are substituted to eliminate confusion and to be consistent with the Specification.

3. Typographical/grammatical error corrected.

4. Changes are made to be consistent with changes made to the Specification.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. The Applicable Safety Analyses discussion states that MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). It also says that even though MTC is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration. The additional sentence has been deleted. The NRC Final Policy Statement on Technical Improvements of July 22, 1993 (58 FR 39132) states that process variables captured by Criterion 2 are not limited to only those directly monitored and controlled from the control room. It also states that Criterion 2 includes other features or characteristics that are specifically assumed in Design Basis Accident and Transient analyses even if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors). Since the Final Policy Statement provides guidance on which types of parameters satisfy Criterion 2, there is no reason to duplicate these words in the CNP ITS.

7. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.3, MODERATOR TEMPERATURE COEFFICIENT (MTC)

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 4

ITS 3.1.4, Rod Group Alignment Limits

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A.1

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

LCO 3.1.4 3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE with all individual indicated rod positions within the allowed rod misalignment of their group step counter demand position as follows:

- for THERMAL POWER less than or equal to 85% of RATED THERMAL POWER, the allowed rod misalignment is ±18 steps, and
- for THERMAL POWER greater than 85% of RATED THERMAL POWER, the allowed rod misalignment is ±12 steps or as determined from Figure 3.1-4. Figure 3.1-4 permits an allowed rod misalignment from ±13 steps (for APL equal to 101%) to ±18 steps (for APL greater or equal to 106%) provided the value of R (defined in Figure 3.1-4) is greater than or equal to 1.04. A.2

APPLICABILITY: MODES 1~~*~~ and 2~~*~~ A.3

ACTION:

ACTION A a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour, and be in HOT STANDBY within 6 hours. LA.1 Add proposed Required Action A.1.2 L.1

ACTION D b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than the allowed rod misalignment, be in HOT STANDBY within 6 hours. L.2 Add proposed Required Actions D.1.1 and D.1.2 M.1

ACTION B c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the allowed rod misalignment, POWER OPERATION may continue provided that within one hour either: L.2

1. The affected rod is restored to OPERABLE status within the above alignment requirements, or THERMAL POWER level is reduced to less than or equal to 85% of RATED THERMAL POWER for rod misalignments less than or equal to ±18 steps, or A.4

2. ~~The affected rod is declared inoperable and~~ the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that: L.2 Add proposed Required Action B.1.2 L.1

   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions, and L.3

~~*See Special Test Exceptions 3.10.2 and 3.10.4~~ A.3

ITS 3.1.4

A.1

ITS

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION B b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

two L.4

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within ~~one~~ hour ~~and within the next 4 hours the High neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or~~ L.5

~~e) The remainder of the rods in the group with the inoperable rod are aligned to within the allowed rod misalignment of the inoperable rod within one hour while maintaining the rod sequence and insertion limits as specified in the COLR; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.~~ A.5

Add proposed ACTION C M.2

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours ~~except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.~~ L.6

SR 3.1.4.2 4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 92 days.

~~4.1.3.1.3 The allowed rod misalignment for THERMAL POWER greater than 85% of RATED THERMAL POWER shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2.~~ A.6

ITS

A.1

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE
FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

L.3

ITS 3.1.4

ITS

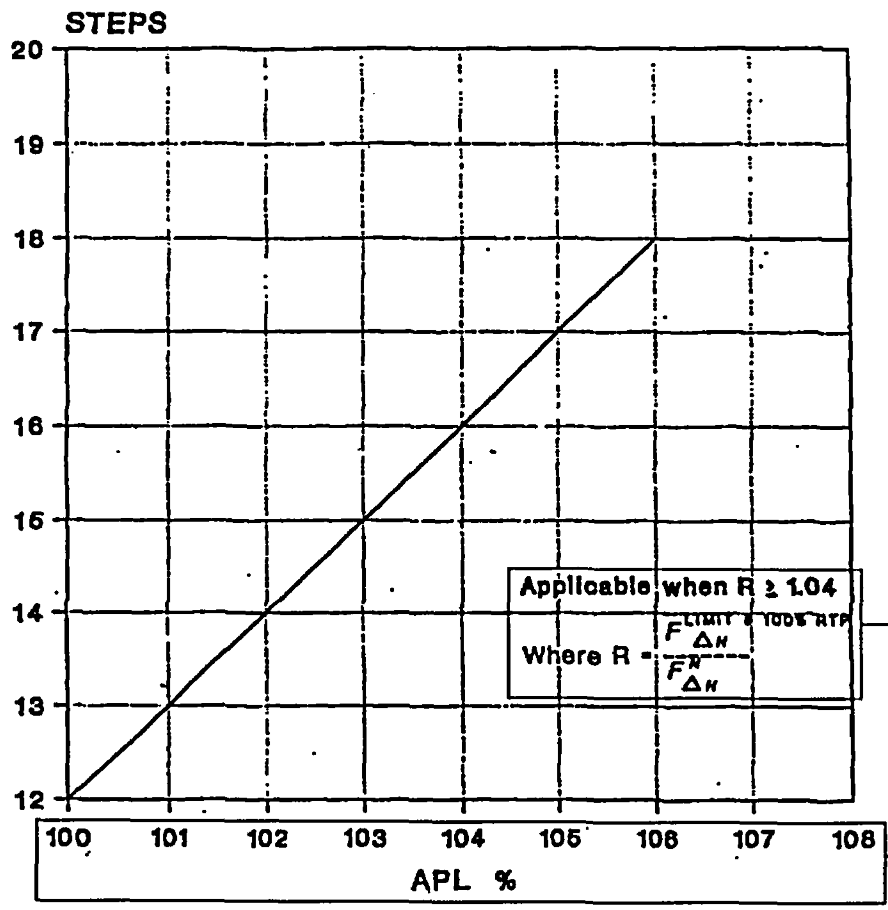
A.1

ALLOWED ROD MISALIGNMENT ABOVE 85% RTP

Figure 3.1.4-1

FIGURE 3.1-4

LCO 3.1.4 Note



A.2

A.6

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

SR 3.1.4.3 3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position ~~(specified in the COLR)~~ shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

~~With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ A.7

Add proposed ACTION A M.3

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to ~~entering MODE 2~~: criticality L.7

a. For all rods following each removal of the reactor vessel head,

~~b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and~~ L.8

~~c. At least once per 18 months.~~ L.9

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ITS 3.1.4

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k. — See ITS 3.1.1

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

Required Action A.1.1 — 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k: — L.10

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). — See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5. — See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5. — See ITS 3.1.1

*See Special Test Exception 3.10.1. — See ITS 3.1.1

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ITS 3.1.4

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - TAVG LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% Delta k/k.

See ITS 3.1.1

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 ~~The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% Delta k/k:~~

L.10

a. ~~Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable.~~ If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

See ITS Chapter 1.0

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration
6. Samarium concentration, and
7. Boron penalty.

See ITS 3.1.1

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ITS

A.1

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

LCO 3.1.4

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE with all individual indicated rod positions within the allowed rod misalignment of their group step counter demand position as follows:

- for THERMAL POWER less than or equal to 85% of RATED THERMAL POWER, the allowed rod misalignment is ±18 steps, and
- for THERMAL POWER greater than 85% of RATED THERMAL POWER, the allowed rod misalignment is ±12 steps or as determined from Figure 3.1-4. Figure 3.1-4 permits an allowed rod misalignment from ±13 steps (for AFL equal to 101%) to ±18 steps (for AFL greater or equal to 106%) provided the value of R (defined in Figure 3.1-4) is greater than or equal to 1.04. A.2

APPLICABILITY: MODES 1* and 2* A.3

ACTION:

ACTION A

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour, and be in HOT STANDBY within 6 hours. LA.1

Add proposed Required Action A.1.2 L.1

ACTION D

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than the allowed rod misalignment, be in HOT STANDBY within 6 hours. L.2

Add proposed Required Actions D.1.1 and D.1.2 M.1

ACTION B

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the allowed rod misalignment, POWER OPERATION may continue provided that within one hour either: L.2

1. The affected rod is restored to OPERABLE status within the above alignment requirements, or THERMAL POWER level is reduced to less than or equal to 85% of RATED THERMAL POWER for rod misalignments less than or equal to ±18 steps, or A.4

2. The affected rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that: L.2

Add proposed Required Action B.1.2 L.1

a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions, and L.3

*See Special Test Exceptions 3.10.2 and 3.10.3 A.3

ITS 3.1.4

A.1

ITS

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION B

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

two — L.4

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within ~~one~~ hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or — L.5

~~e) The remainder of the rods in the group with the inoperable rod are aligned to within the allowed rod misalignment of the inoperable rod within one hour while maintaining the rod sequence and insertion limits as specified in the COLR; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.~~ — A.5

Add proposed ACTION C — M.2

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. — L.6

SR 3.1.4.2 4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 92 days.

~~4.1.3.1.3 The allowed rod misalignment for THERMAL POWER greater than 85% of RATED THERMAL POWER shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2.~~ — A.6

ITS 3.1.4

A.1

ITS

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

L.3

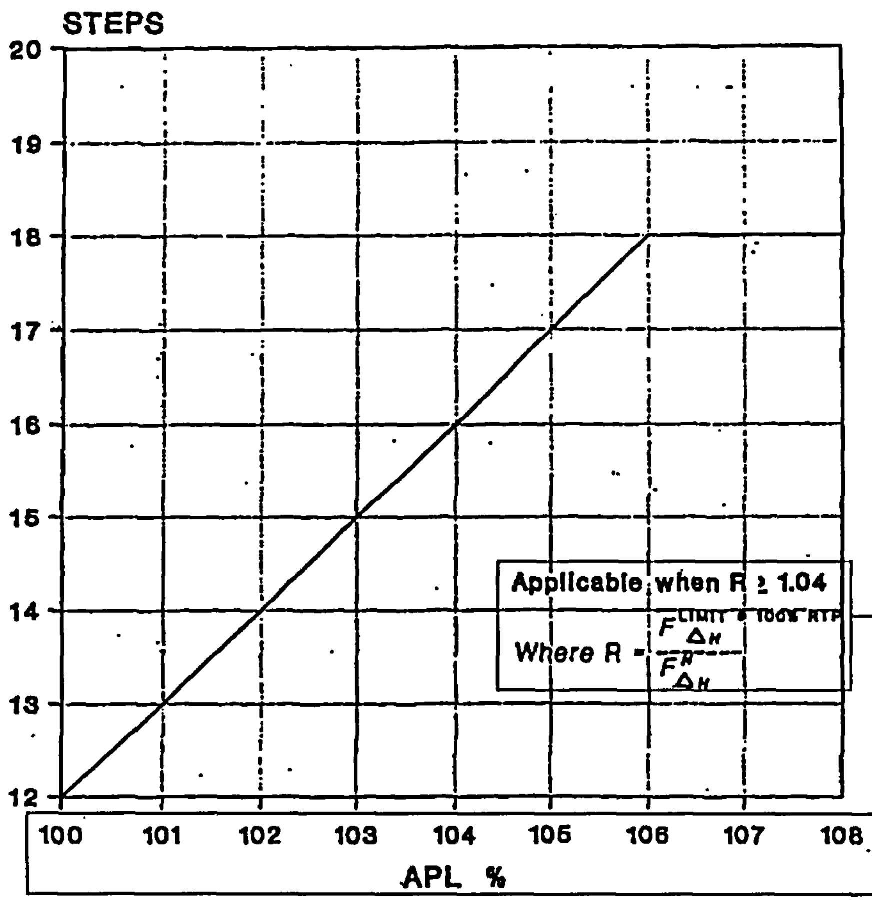
ITS

A.1

ALLOWED ROD MISALIGNMENT ABOVE 85% RTP

Figure 3.1.4-1

FIGURE 3.1-4



A.2

A.6

LCO 3.1.4 Note

ITS

A.1

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

SR 3.1.4.3

3.1.3.4 The individual full length ~~(shutdown and control)~~ rod drop time from the fully withdrawn position ~~(specified in the COLR)~~ shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

500 — L.11

a. T_{avg} greater than or equal to ~~541~~°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 AND 2

ACTION:

~~With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ — A.7

Add proposed ACTION A — M.3

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to ~~entering MODE 2~~:

criticality — L.7

a. For all rods following each removal of the reactor vessel head;

~~b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and~~ — L.8

~~c. At least once per 18 months.~~ — L.9

ITS 3.1.4

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k. — See ITS 3.1.1

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

Required Action A.1.1 — 4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k: — L.10

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). — See ITS Chapter 1.0

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6. — See ITS 3.1.6

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6. — See ITS 3.1.1

*See Special Test Exception 3.10.1. — See ITS 3.1.1

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ITS 3.1.4

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% Delta k/k. See ITS 3.1.1

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% Delta k/k: L.10

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS Chapter 1.0

b. At least once per 24 hours by consideration of the following factors: See ITS 3.1.1

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and
7. Boron penalty.

DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.1.3.1 specifies the rod misalignment limits for full length (shutdown and control) rods at a THERMAL POWER > 85% RATED THERMAL POWER (RTP) and at THERMAL POWER ≤ 85% RTP. At a THERMAL POWER > 85% RTP the allowed rod misalignment is +/- 12 steps or as determined from Figure 3.1-4. In addition, CTS 3.1.3.1 states that Figure 3.1-4 permits an allowed rod misalignment from +/- 13 steps (for ALLOWABLE POWER LEVEL (APL) equal to 101%) to +/- 18 steps (for APL greater or equal to 106%) provided the value of R (defined in Figure 3.1-4) is ≥ 1.04. The R limit and definition are maintained in the ITS 3.1.4 Note and the range of rod misalignment allowed is maintained in ITS Figure 3.1.4-1. ITS LCO 3.1.4 states that with THERMAL POWER > 85% RTP, the individual rod positions shall be within 12 steps of their group step counter demand position or as determined from Figure 3.1.4-1, and the Note to ITS LCO 3.1.4 states the R limit and provides the definition. ITS LCO 3.1.4 does not contain the allowed misalignment range and ITS Figure 3.1.4-1 does not include the R limit or definition.

The purpose of the details of CTS 3.1.3.1 is to clarify the details provided in the CTS Figure. However, the information provided in the two locations is duplicative. This change is acceptable because the technical requirements have not changed. The R limit and definition are maintained in the ITS 3.1.4 Note and the range of rod misalignment allowed is maintained in ITS Figure 3.1.4-1. Since the details are duplicative there is no reason to maintain the information in both locations. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 The Applicability of CTS 3.1.3.1 is modified by footnote * that states "See Special Test Exceptions 3.10.2 and 3.10.4" (Unit 1) and "See Special Test Exceptions 3.10.2 and 3.10.3" (Unit 2). ITS 3.1.4 Applicability does not contain the footnote or a reference to the Special Test Exceptions.

The purpose of the footnote reference is to alert the user that a Special Test Exception exists that may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 3.1.3.1 Action c.1 states that with one full length rod misaligned from the group step counter demand position by more than the rod misalignment requirements, POWER OPERATION may continue provided that within one hour, the affected rod is restored to OPERABLE status within the above alignment

DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

requirements, the THERMAL POWER level is reduced to less than or equal to 85% RTP for rod misalignments less than or equal to ± 18 steps, or other compensatory measures described in the Action are taken. ITS 3.1.4 does not contain a Required Action stating that the rod must be restored to OPERABLE status within the alignment limits.

This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available Required Action and it is the convention in the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS 3.1.3.1 Action c.2.e) states that with one full length rod misaligned from the group step counter demand position by more than the rod misalignment requirements, POWER OPERATION may continue provided that the remainder of the rods in the same group as the inoperable rod are aligned to within the allowed rod misalignment of the inoperable rod within one hour while maintaining the rod sequence and insertion limits as specified in the COLR; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 (Unit 1) and Specification 3.1.3.6 (Unit 2) during subsequent operation. ITS 3.1.4 does not contain a Required Action stating that the remainder of the rods in the group must be aligned with the misaligned rod.

This change is acceptable because the technical requirements have not changed. Moving the remainder of the rods in a group to within the LCO limit of the misaligned rod while maintaining compliance with all other rod position requirements is simply restoring compliance with the LCO. Restoration of compliance with the LCO is always an available Required Action and it is the convention in the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as administrative because it does not result in technical changes to the CTS.

A.6 CTS Figure 3.1-4, Allowed Rod Misalignment above 85% RTP, is based upon the current Allowable Power Level (APL) as determined in CTS 3.2.6. In addition, CTS 4.1.3.1.3 requires the allowed rod misalignment for THERMAL POWER > 85% RTP to be determined in conjunction with the measurement of APL as defined in CTS 4.2.6.2. The term APL has been changed to $F^W_Q(Z)$, as described in the DOCs for ITS 3.2.1. Therefore, in the ITS, the allowed rod misalignment is being based upon $F^W_Q(Z)$. In order to maintain a similar value in the ITS Figure as is in the CTS Figure, the term in ITS Figure 3.1.4-1 is (CFQ x K(Z))/$F^W_Q(Z)$. In addition, the ITS does not include a specific SR in ITS 3.1.4 to calculate the new allowed rod misalignment every time an $F^W_Q(Z)$ determination is made. This changes the CTS by using the term $F^W_Q(Z)$ in lieu of the term APL, and not including a specific SR to calculate the allowed rod misalignment every time $F^W_Q(Z)$ is determined.

This change is acceptable since, as described in the DOCs for ITS 3.2.1, the term $F^W_Q(Z)$ is analogous to APL. Also, the specific SR is not needed because each time the $F^W_Q(Z)$ Surveillance is performed in ITS 3.2.1, the allowed rod alignment limit (if using ITS Figure 3.1.4-1) must be established based on the

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DISCUSSION OF CHANGES
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

most recently calculated actual value of $F^W_Q(Z)$. Thus, the technical requirements have not changed; the verification that the individual rod positions are within alignment limits must always be performed and compared to the existing limit. This change is designated as administrative because it does not result in a technical change to the CTS.

A.7 The CTS 3.1.3.3 (Unit 1) and CTS 3.1.3.4 (Unit 2) Action requires that with the drop time of any full length rod determined to exceed the limits of the LCO, to restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2. The ITS does not have a similar requirement.

CTS 4.0.4 and ITS SR 3.0.4 require verification that Surveillances are met prior to entering the MODE in which they apply. CTS 4.0.4 and ITS SR 3.0.4 also prohibit entering a MODE or condition with the Surveillance not met and while relying on Actions. Therefore, since the Applicability of CTS 3.1.3.3 (Unit 1) and CTS 3.1.3.4 (Unit 2) is MODES 1 and 2, the Action prohibiting entry into MODES 1 and 2 with the rod drop time requirements not met is redundant to CTS 4.0.4 and ITS SR 3.0.4. This change is acceptable because the technical requirements have not changed. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.1.3.1 Action b states that with more than one full length rod inoperable or misaligned from the group step counter demand position by more than the allowed rod misalignment, be in HOT STANDBY within 6 hours. ITS 3.1.4 ACTION D states that with more than one rod not within alignment limit, verify SDM is within limits or initiate boration to restore required SDM to within limit within one hour, and be in MODE 3 in 6 hours. This changes the CTS by adding new requirements to verify SDM limits or to initiate boration to restore SDM limits.

The purpose of CTS 3.1.3.1 Action b is to place the unit in a condition in which the equipment is not required. More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored. This change is acceptable because it is consistent with the requirements of the assumptions of the safety analyses to be within the SDM limit. The change has been designated as more restrictive because it adds explicit actions to verify SDM or to restore SDM within limits.

M.2 CTS 3.1.3.1 Action c states that with one full length rod misaligned, POWER OPERATION may continue provided that certain actions are completed within one hour. If those actions are not complete, CTS 3.0.3 would be entered

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requiring entry into Hot Standby (MODE 3) within 7 hours, for a total time from condition discovery to entry into MODE 3 of 8 hours. ITS 3.1.4 ACTION C states that if any Required Action and associated Completion Time of Condition B (one rod not within alignment limits) is not met, the unit must be in MODE 3 within 6 hours. The shortest Completion Time in ITS ACTION B is one hour. Therefore, under the ITS, the shortest possible time from discovery of the condition to entry into MODE 3 is 7 hours. This changes the CTS by providing one less hour for entry into MODE 3 following discovery of a misaligned rod if Required Actions are not met.

The purpose of requiring a shutdown when a rod misalignment cannot be corrected is to bring the unit to a subcritical condition prior to the build up of an undesirable reactor core power distribution. This change is acceptable because it provides an adequate period of time to correct the condition or be in a MODE in which the requirement does not apply. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power in an orderly manner and without challenging unit systems.

M.3 The CTS 3.1.3.3 (Unit 1) and CTS 3.1.3.4 (Unit 2) Action requires that with the drop time of any full length rod determined to exceed the limits of the LCO, to restore the rod drop time to within the limit prior to proceeding to MODE 1 or 2. However, no specific actions are stated in CTS 3.1.3.3 (Unit 1) and CTS 3.1.3.4 (Unit 2) if the unit is in MODE 1 or 2 when the rod drop time is discovered to not be within limits. Therefore, a CTS 3.0.3 entry would be required. CTS 3.0.3 allows one hour to prepare for a shutdown and requires the unit to be in MODE 3 within 7 hours. ITS 3.1.4 ACTION A applies with one or more rod(s) inoperable. It requires the verification of SDM to be within limits or to initiate boration to restore SDM to within limit within 1 hour, and requires the unit to be in MODE 3 in 6 hours. This changes the CTS by adding new requirements associated with SDM and changing the requirement to be outside of the MODE of Applicability from 7 hours to 6 hours.

The purpose of requiring a shutdown when a drop time of any full length rod is not met is to bring the unit to a subcritical condition. With one or more slow control rod(s) there is a potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution in the reactor core, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored. In addition, the new time to reach MODE 3 is consistent with the time provided in other Specifications. This change is acceptable because it is consistent with the requirements of the assumptions of the safety analyses to be within the SDM limit. The change has been designated as more restrictive because it adds explicit actions to verify SDM or to restore SDM within limits and reduces the time required to be in MODE 3.

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RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.3.1 Action a applies when one or more full length rods are inoperable "due to being immovable as a result of excessive friction or mechanical interferences or known to be untrippable." ITS 3.1.4 Condition A applies when one or more rod(s) are inoperable. ITS 3.1.4 Condition A does not list the ways in which the rods can be inoperable (i.e., "due to being immovable as a result of excessive friction or mechanical interferences or known to be untrippable"). This changes the CTS by moving the details of the reason the rod is considered inoperable to the Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. While the ITS Bases will not include the specific words being removed from the CTS, the words used in the ITS Bases, "(i.e., untrippable)" is synonymous to the removed CTS words, and provides clarity. The ITS still retains the requirement for the shutdown and control rods to be OPERABLE and provides a Condition for when the rod is inoperable. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program described in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 Actions a and c.2 require satisfying the SHUTDOWN MARGIN requirement in accordance with Specification 3.1.1.1. In the same conditions, ITS 3.1.4 requires verification that the SHUTDOWN MARGIN is within limits or initiating boration to restore SDM to within limits. This changes the CTS by providing the option to initiate action to establish compliance with the SDM requirement within 1 hour instead of declaring the Required Action not met and following ITS LCO 3.0.3.

The purpose of CTS 3.1.3.1 Actions a and c.2 is to ensure that adequate SHUTDOWN MARGIN exists. Following misalignment of a rod, boration may be required to reestablish compliance with the SHUTDOWN MARGIN requirements. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe

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operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Providing a short period of time to reestablish the SHUTDOWN MARGIN requirement instead of entering ITS LCO 3.0.3 is justified because of the existing conservatisms in the SHUTDOWN MARGIN calculations and the fact that the rod is still trippable. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 Action a specifies requirements for one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable. CTS 3.1.3.1 Action b specifies requirements for more than one full length rod inoperable or misaligned from the group step counter demand position by more than the allowed rod misalignment. CTS 3.1.3.1 Action c specifies requirements for one full length rod inoperable due to causes other than those addressed by Action a, above, or misaligned from its group step counter demand position by more than the allowed rod misalignment. CTS 3.1.3.1 Action c.2 requires the affected rod to also be declared inoperable. ITS 3.1.4 ACTION A specifies requirements for one or more rod(s) inoperable. ITS 3.1.4 ACTION B specifies requirements for one rod not within alignment limits. ITS 3.1.4 ACTION D specifies requirements for more than one rod not within alignment limits. This changes the CTS by considering shutdown and control rods that are trippable but misaligned to be OPERABLE and excludes other types of control rod inoperabilies not addressed in CTS 3/4.1.3.1 (e.g., insertion times). The requirement to declare a misaligned rod inoperable in CTS 3.1.3.1, Action c.2, is deleted. The requirements for control rod drop times are addressed in DOC M.3.

The purpose of ITS 3.1.4 is to ensure that the shutdown and control rods are capable of performing their safety function of inserting into the core when required. A secondary function of the control rods is to maintain alignment so that the reactor core power distribution is consistent with the safety analyses. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. In the ITS, rod OPERABILITY is related only to trippability, and a misaligned rod is not considered inoperable if it can be tripped. Misalignment is addressed by the ITS 3.1.4 LCO, but is separate from OPERABILITY. In both cases, trippability and misalignment, the ITS continues to provide appropriate compensatory measures. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 Action c.2.a) states that when a rod is misaligned, POWER OPERATION may continue if a reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days. This re-evaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions. ITS 3.1.4 Required Action B.6 states that when one rod is misaligned, re-evaluate the safety analyses and confirm results remain valid for the duration of operation under these conditions. This changes the CTS by eliminating

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Table 3.1-1, which lists the specific events to be re-evaluated and the Action to evaluate those specific events.

The purpose of CTS 3.1.3.1 Action c.2.a) is to ensure that the accident analyses performed for the reload core continue to be acceptable during operation with a misaligned rod. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. The elimination of a specific set of events to be re-evaluated does not change the requirement to verify continued operation is acceptable and places the responsibility on the licensee to re-evaluate all accident analyses which may be affected by a misaligned rod. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.4 *(Category 3 – Relaxation of Completion Time)* CTS 3.1.3.1 Action c.2.d) states that with one rod misaligned, reduce the THERMAL POWER level to ≤ 75% of RATED THERMAL POWER within one hour. ITS 3.1.4 Required Action B.2 requires THERMAL POWER to be reduced to ≤ 75% RTP within 2 hours. This changes the CTS by changing the Completion Time from one hour to two hours.

The purpose of CTS 3.1.3.1 Action c.2.d) is to reduce reactor core power to ensure that the increases in linear heat generation rate due to misalignment of a rod does not result in exceeding the design limits. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Trip System. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.5 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.1 Action c.2.d) states that with one rod misaligned, reduce the THERMAL POWER level to ≤ 75% of RATED THERMAL POWER and reduce the high neutron flux trip setpoint to ≤ 85% of RTP within the next 4 hours. ITS 3.1.4 Required Action B.2 requires THERMAL POWER to be reduced to ≤ 75% RTP, but does not require the high neutron flux trip setpoint to be reduced. This changes the CTS by eliminating the Required Action to reduce the high neutron flux trip setpoint.

The purpose of CTS 3.1.3.1 Action c.2.d) is to reduce reactor core power to ensure that the increases in linear heat generation rate due to misalignment of a rod does not result in exceeding the design limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk

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ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Lowering the high neutron flux trip setpoint increases the chance for an inadvertent reactor trip due to the changes being made to the Reactor Trip System without providing a commensurate amount of added safety. Administrative methods of maintaining reactor power below that allowed by the Required Action are sufficient to protect the core. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.6 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.1.3.1.1 requires the position of each full length rod to be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. ITS SR 3.1.4.1 requires verification that the individual rod positions are within the alignment limits every 12 hours. This changes the CTS by eliminating the requirement to verify the individual rod positions to be within alignment limits every 4 hours when the Rod Position Deviation Monitor is inoperable.

The purpose of CTS 4.1.3.1.1 is to periodically verify that the rods are within the alignment limits specified in the LCO. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of rod position verification when the Rod Position Deviation Monitor is inoperable is unnecessary, since an inoperability of the alarm does not increase the probability that the rods are misaligned. The Rod Position Deviation Monitor alarm is for indication only. Its use is not credited in any safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.7 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.1.3.3 (Unit 1) and CTS 4.1.3.4 (Unit 2) require the rod drop time test to be performed prior to entering MODE 2 following each removal of the reactor vessel head. ITS SR 3.1.4.3 requires this test to be performed prior to criticality after each removal of the reactor head. This changes the CTS by allowing the rod drop test to be delayed from before entering MODE 2 to prior to criticality.

The purpose of the CTS and ITS is to confirm rod drop times as soon as practicable after the reactor vessel head is re-installed. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. MODE 2 begins at $k_{eff} \geq 0.99$. Criticality occurs when $k_{eff} = 1.0$. Therefore, this change only slightly extends the period when the test must be completed. The test must still be completed before any significant THERMAL POWER level is achieved. This change is designated as less restrictive because Surveillances will be

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ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

completed at a later time after the reactor vessel head is re-installed and the plant is in MODE 2.

L.8 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.3.b (Unit 1) and CTS 4.1.3.4.b (Unit 2) require the rod drop time of full length rods to be demonstrated through measurement prior to entering MODE 2 for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods. The ITS does not include this testing requirement.

The purpose of CTS 4.1.3.3.b (Unit 1) and CTS 3.1.3.4.b (Unit 2) is to verify OPERABILITY of the control rods following maintenance that could alter their operation. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, modification, or replacement of a component, post-maintenance testing is required to demonstrate the OPERABILITY of the system or component. This is described in the Bases for ITS SR 3.0.1 and required under ITS SR 3.0.1. The OPERABILITY requirements for the rod control system are described in the Bases for ITS 3.1.4. In addition, the requirements of 10 CFR 50, Appendix B, Section XI (Test Control) provide adequate controls for test programs to ensure that testing incorporates applicable acceptance criteria. Compliance with 10 CFR 50, Appendix B, is required under the unit operating license. As a result, post-maintenance testing will continue to be performed and an explicit requirement in the Technical Specifications is not necessary. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.9 *(Category 10 – 18 to 24 Month Surveillance Frequency Change, Non-Channel Calibration Type)* CTS 4.1.3.3.c (Unit 1) and CTS 4.1.3.4.c (Unit 2) require the rod drop time of full length rods to be demonstrated through measurement prior to entering MODE 2 following each removal of the reactor vessel head and at least once per 18 months. ITS SR 3.1.4.3 requires the test to be performed prior to criticality after each removal of the reactor head. The requirements in the CTS to perform the test following each removal of the reactor vessel head and at least once per 18 months normally coincide with one another. The head is removed once each cycle (approximately once every 18 months) unless there is a need to remove the head prior to the end of the cycle. This changes the CTS by extending the Frequency of the Surveillance from 18 months (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in CTS 4.0.2 and ITS SR 3.0.2) to prior to criticality after each removal of the reactor head. This new Surveillance could occur up to once every 24 months (i.e., a maximum of 30 months or greater accounting for the allowable grace period specified in CTS 4.0.2 and ITS SR 3.0.2) depending on when the head is removed.

The purpose of CTS 4.1.3.3.c (Unit 1) and CTS 4.1.3.4.c (Unit 2) is to ensure the rods insert within the rod drop criteria. This change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in

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ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical surveillance data and maintenance data sufficient to determine failure modes have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. Extending the Surveillance test interval for the rod drop test SR is acceptable because the rods are tested during the cycle to ensure the rods are positioned within the rod alignment criteria and to ensure rod freedom of movement (trippability). This testing, which exercises the rods, helps to ensure the rods are able to drop into the core during the cycle and detect significant failures of the rods. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed Surveillance Frequency of prior to criticality after each removal of the reactor head even if performed at or greater than the maximum interval allowed by ITS SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.10 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.1.1.1.a and CTS 4.1.1.2.a require verification of SHUTDOWN MARGIN within one hour after detection of inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) are inoperable. These requirements are applicable in MODES 1, 2, 3, 4, and 5. ITS 3.1.4 Required Action A.1.1 requires the verification of SDM to be within limits within 1 hour. These verifications are required in MODES 1 and 2 with one or more control rod(s) inoperable. This changes the CTS by not requiring any explicit SDM verifications for inoperable control rod(s) in MODES 3, 4, and 5 other than the normal verifications specified in ITS SR 3.1.1.1 (once every 24 hours). For MODE 1 and 2 operations, this changes the CTS by not requiring the verification of SDM on a once per 12 hour basis for one or more inoperable rod(s).

The purpose of CTS 4.1.1.1.1.a and CTS 4.1.1.2.a are to provide the appropriate compensatory measures to determine SDM when control rod(s) are inoperable during operations in MODES 1, 2, 3, 4, and 5. The purpose of the ITS 3.1.4 ACTIONS are to provide the appropriate compensatory actions for inoperable control rods in MODES 1 and 2. The purpose of ITS SR 3.1.1.1 is to provide the normal Frequency for verification of SDM regardless of the status of the control rod(s). When the plant is operating in MODES 1 and 2, with one or more rod(s) inoperable the unit must be in MODE 3 within 6 hours. After reaching MODE 3, ITS 3.1.4 no longer applies therefore it is inappropriate to specify additional actions after the unit is outside the Applicability of the Specification. Nevertheless, SDM must still be verified in accordance with ITS SR 3.1.1.1 every 24 hours. This SDM verification must also compensate for the reactivity worth of the control rod that is not fully inserted since it is required by the definition of SDM. Therefore, ITS 3.1.4 ACTIONS provide the appropriate compensatory measures. In MODES 3, 4, and 5, SDM will be monitored in accordance with ITS SR 3.1.1.1 every 24 hours. This change is acceptable since SDM will still be

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required to be monitored every 24 hours, and based on the definition of SDM the reactivity worth of any rod not capable of being fully inserted must be accounted for in the determination of SDM. Thus, SDM continues to be monitored in a manner and at a Frequency necessary to give confidence that the assumptions in the safety analyses are protected. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.11 *(Category 1 - Relaxation of LCO Requirements)* CTS 3.1.3.4 (Unit 2) contains the specific requirements for rod drop time testing. The CTS specifies that the rod drop time be verified at an RCS T_{avg} of ≥ 541°F. ITS SR 3.1.4.3 specifies the rod drop time be verified at a RCS T_{avg} of ≥ 500°F. This changes the CTS by lowering the required temperature at which rod drop time must be verified.

The purpose of CTS 3.1.3.4 (Unit 2) is to ensure the rods insert within the rod drop time criteria. The performance of rod drop time tests ensure that the required negative reactivity insertion (amount and rate) from a reactor trip is within the values assumed in the safety analyses. This change will allow rod drop time testing to begin earlier during a startup following a refueling outage. The proposed change is acceptable because the specified rod drop time remains unchanged and the proposed 500°F test temperature is conservative compared to the CTS requirement of 541°F. Since the moderator becomes denser as the RCS temperature is decreased, a lower RCS temperature results in slower rod drops due to the density change of the water. However, the limiting rod drop time requirement of the CTS (≤ 2.7 seconds (Unit 2)) is maintained in the ITS and must still be met. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.3.1

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

as follows:

Individual indicated rod positions shall be ~~within 12 steps of their group step counter demand position~~ INSERT 1

(1)

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable. (3.1.3.1 ACTION a)	A.1.1 Verify SDM ~~to be~~ within ~~the~~ limits ~~specified in the COLR~~. (15) OR A.1.2 Initiate boration to restore SDM to within limit (5) AND A.2 Be in MODE 3	1 hour (2) (3) 1 hour (2) 6 hours
B. One rod not within alignment limits. (3.1.3.1 ACTION c.2)	~~B.1 Restore rod to within alignment limits.~~ ~~OR~~ B.2.1.1 Verify SDM ~~to be~~ within ~~the~~ limits ~~specified in the COLR~~. (15) OR	~~1 hour~~ 1 hour (3) (4)

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1

INSERT 1

a. With THERMAL POWER $\leq$ 85% RTP, within 18 steps of their group step counter demand position; and

b. With THERMAL POWER > 85% RTP, within 12 steps of their group step counter demand position or as determined from Figure 3.1.4-1.

-NOTE-

The limits of Figure 3.1.4-1 are only applicable when $R \geq 1.04$, where $R = \dfrac{F_{\Delta H}^{\text{Limit @ 100\%RTP}}}{F_{\Delta H}^{N}}$.

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.1.3.1 Action C.2		B.2.1.2 Initiate boration to restore SDM to within limit. (5) AND	1 hour	(2)
		B.2.2 Reduce THERMAL POWER to ≤ 75% RTP. AND	2 hours	
		B.2.3 Verify SDM is within ~~the~~ limits ~~specified in the COLR~~. AND	Once per 12 hours	(3)
		B.2.4 Perform SR 3.2.1.1 and SR 3.2.1.2. AND	72 hours	(4) (2)
		B.2.5 Perform SR 3.2.2.1. AND	72 hours	
		B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days	
DOC M.2	C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours	
3.1.3.1 ACTION b.	D. More than one rod not within alignment limit. (5)	D.1.1 Verify SDM is within ~~the~~ limits ~~specified in the COLR~~. OR	1 hour	(2) (3)

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CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.1.3.1 Action b		D.1.2 Initiate boration to restore required SDM to within limit. 5 AND D.2 Be in MODE 3.	1 hour 6 hours	2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.3.1.1	SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours 5	2
4.1.3.1.2	SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ ~~10~~ steps in either direction. 8	92 days	5
LCO 3.1.3.3 (Unit 1), LCO 3.1.3.4 (Unit 2), 4.1.3.3 (Unit 1), 4.1.3.4 (Unit 2)	SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ ~~[2.2]~~ 2.4 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F and 3 b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head (Unit 1) and 2.7 seconds (Unit 2)	6 7

INSERT 2 1

1 **INSERT 2**

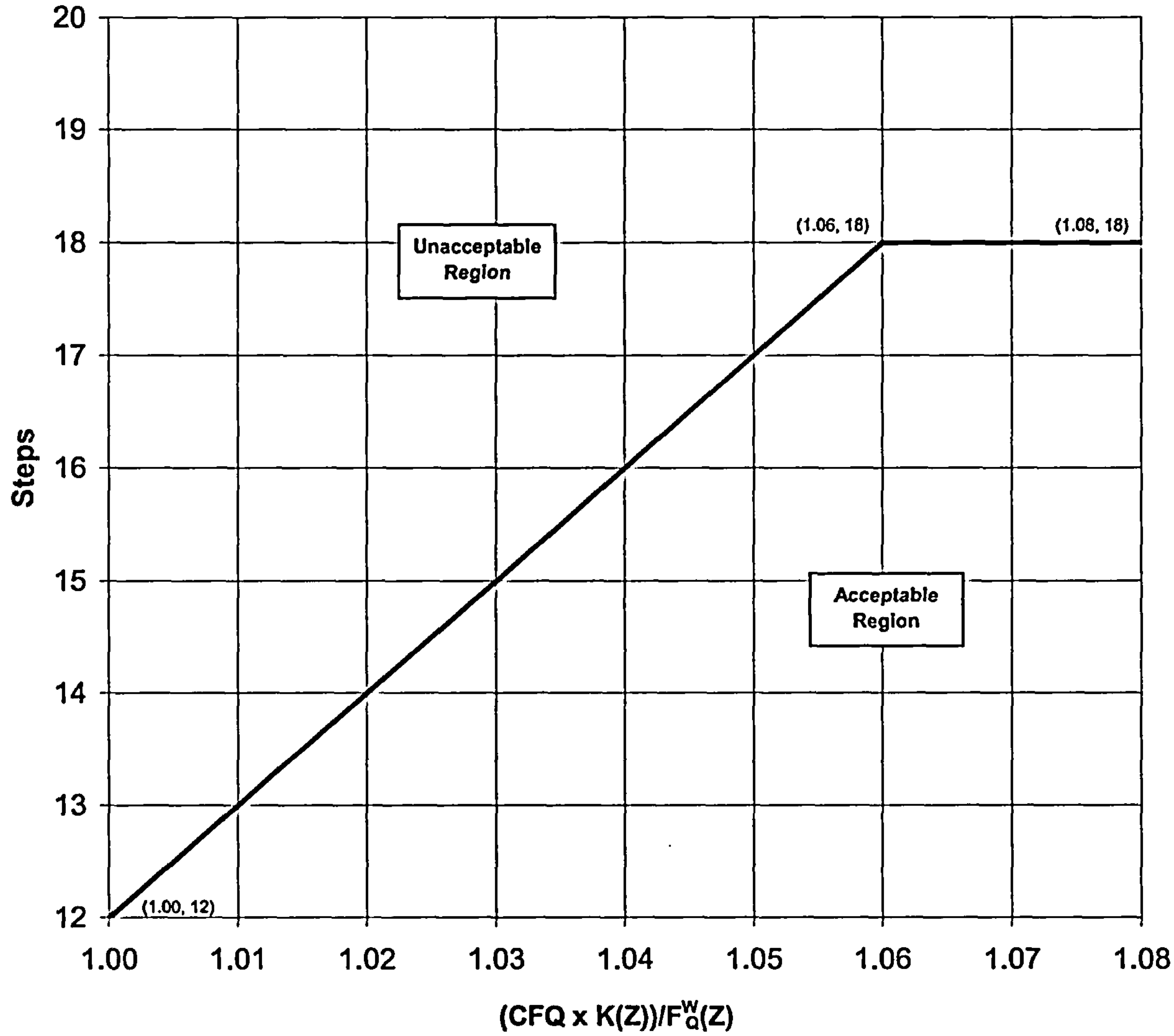

Figure 3.1.4-1
Allowed Rod Misalignment Above 85% RTP

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

1. The LCO has been modified to incorporate a CNP specific allowance. The change allows the alignment criteria to vary as a function of $F^W_Q(Z)$. This change to the LCO has been made consistent with the allowances in License Amendments 193 (Unit 1) and 179 (Unit 2) dated March 15, 1995 (as modified in the ITS 3.1.4 DOCs).

2. Typographical/grammatical error corrected.

3. Changes are made to be consistent with the format of the ITS. The location of where a parameter's limits reside, whether in the COLR or an actual LCO statement, is not normally specified in the Required Action. The Required Action normally states that the parameter shall be "within limits."

4. ISTS 3.1.4 Required Action B.1 requires restoration of a rod not within alignment limits within 1 hour or performance of a number of other actions, such as verification of SHUTDOWN MARGIN, reduction in reactor power, measurement of hot channel factors, and re-evaluation of the safety analyses. The Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 4.1.6.g, states "A Required Action which requires restoration, such that the Condition is no longer met, is considered superfluous. It is only included if it would be the only Required Action for the Condition or it is needed for presentation clarity." Neither exception applies in this case. In fact, the inclusion of Required Action B.1 requires an additional level of indenting and numbering for the remaining Required Actions in Condition B, which reduces its clarity. Therefore, Required Action B.1 is deleted and the subsequent Required Actions renumbered.

5. SR 3.1.4.2 has been modified to incorporate a CNP specific allowance, consistent with the CNP licensing basis. The amount of insertion to verify rod trippability has been changed from 10 steps to 8 steps.

6. The brackets have been removed and the proper plant specific information/value has been provided.

7. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. 8

The applicable criteria for these reactivity and power distribution design requirements are ~~10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1),~~ INSERT 1 and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. ~~2~~ 3). 1 1

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. If a bank of RCCAs consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. ~~All units have~~ There are four control banks and ~~at least two~~ four shutdown banks. 2

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern,

1

INSERT 1

Plant Specific Design Criterion (PSDC) 6, "Reactor Core Design" (Ref. 1), PSDC 28, "Reactivity Hot Shutdown Capability" (Ref. 2), PSDC 29, "Reactivity Shutdown Capability" (Ref. 2), PSDC 30, "Reactivity Holddown Capability" (Ref. 2)

BASES

BACKGROUND (continued)

using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System. (2)

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod. (2)

The DRPI System provides a highly accurate indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy. The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy. (2) (1) (2)

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (4) (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that: (2)

a. There be no violations of:

1. Specified acceptable fuel design limits, or (5)

BASES

APPLICABLE SAFETY ANALYSES (continued)

; ③

2. Reactor Coolant System (RCS) pressure boundary integrity and

b. The core remains subcritical after accident transients.

INSERT 2 ②

~~Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking.~~ A different ~~The second~~ type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

INSERT 3 ②

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank ~~inserted to its insertion limit~~. Satisfying limits on ~~departure from nucleate boiling ratio~~ DNBR in both of these cases bounds the situation when a rod is misaligned from its group ~~by 12 steps~~. ①

INSERT 4 ②

INSERT 5 ②

②

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

~~The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.~~ ⑦

~~Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.~~ ⑦

2 **INSERT 2**

There are three RCCA misalignment accidents which are analyzed which include one or more dropped RCCAs, a dropped RCCA bank, and a statically misaligned RCCA (Ref. 4).

2 **INSERT 3**

For the dropped RCCA(s) or dropped RCCA bank misalignment accidents a negative reactivity insertion will result. Power may be reestablished either by reactivity feedback or control bank withdrawal. Following plant stabilization, normal rod retrieval or shutdown procedures are followed. For dropped RCCA events in the automatic rod control mode, the Rod Control System detects the drop in power and initiates control bank withdrawal. In all cases, the minimum departure from nucleate boiling ratio (DNBR) remains above the limit.

2 **INSERT 4**

and the remainder of the bank inserted

2 **INSERT 5**

within the limits specified in the LCO.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. INSERT 6 (4)

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. ~~The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.~~ INSERT 7 (2)

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the ~~plant~~ unit. (2) In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

4

INSERT 6

or as determined from Figure 3.1.4-1 with THERMAL POWER ≥ 85% or within 18 steps of their group step counter demand position when THERMAL POWER is < 85% RTP

2

INSERT 7

The safety analysis assumes a misalignment of one or more RCCA(s) or an entire RCCA bank. A misalignment of 30 steps will not cause power distribution worse than the design limits. Power distribution evaluations for steady state and load following conditions with rod misalignment of 30 steps showed that the increase in peaking factors could be accommodated at or below 85% RTP. Evaluations also showed that above 85% RTP, a misalignment of 30 steps could be accommodated if the margin in (CFQ x K(Z))/F^W_Q(Z) is at least 1.06 and margin in $F^N_{\Delta H}$ is at least 4%. For lower (CFQ x K(Z))/F^W_Q(Z) values the allowable misalignment is reduced.

BASES

ACTIONS

A.1.1 and A.1.2

③

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM. ②

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

A.2

INSERT 9 ⑤

unit

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. ②

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. ②

unit

~~B.1~~

~~When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.~~ ④

~~An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.~~

B.②1.1 and B.②1.2 ④

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

INSERT 8

Not Used

5 INSERT 9

When one or more rods are inoperable

BASES

ACTIONS (continued)

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps. (2)

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration. OPERABLE (5) (4)

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (4)

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. power (5)

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 7). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. (2) (2)

Trip

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$, and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in (U) FSAR Chapter 15 (Ref. 6) that may be adversely affected will be (2)

BASES

ACTIONS (continued)

evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

any associated 5

When any Required Action ~~s~~ cannot be completed within the ~~e~~ associated Completion Time, the unit must be brought to a MODE ~~or Condition~~ in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE ~~2 with K_{eff} < 1.0~~ 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the ~~plant~~ unit systems. 5 5 2

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases ~~of~~ of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored. 8

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE ~~or Condition~~ in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE ~~2 with K_{eff} < 1.0~~ 3 within 6 hours. 5 5

BASES

ACTIONS (continued)

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{eff} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems. 3 unit 5 2

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2 with $K_{eff} \geq 1.0$, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken. 5 8

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average

BASES

SURVEILLANCE REQUIREMENTS (continued)

moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

unit

This Surveillance is performed during a ~~plant~~ outage, due to the plant conditions needed to perform the SR and the potential for an unplanned ~~plant~~ transient if the Surveillance were performed with the reactor at power. (2) (2)

unit

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10 and GDC 26.~~ UFSAR, Section 1.4.2 (1)

(3) ~~2.~~ 10 CFR 50.46.

2. UFSAR, Section 1.4.5 (1)

(4) ~~3.~~ (U)FSAR, ~~Chapter [15]~~. Section 14.1.3 (2)

(5) ~~4.~~ (U)FSAR, ~~Chapter [15]~~. Section 14.2.6 (2)

~~5. FSAR, Chapter [15].~~ (6)

6. (U)FSAR, Chapter ~~[15]~~ (14). (2)

~~7. FSAR, Chapter [15].~~ (6) (6)

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4 BASES, ROD GROUP ALIGNMENT LIMITS

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

4. The Bases are changed to reflect changes made to the Specification.

5. Changes made to be consistent with the Specification.

6. The brackets have been removed and the proper plant specific information/value has been provided.

7. The discussion of the Required Actions when the LCO is not met has been deleted since it is not appropriate in the Applicable Safety Analyses Section. This information is adequately discussed in the Bases for ACTIONS B.2, B.3, B.4, B.5, and B.6. This is also consistent with the format of the ISTS.

8. Typographical/grammatical error corrected.

Attachment 1, Volume 6, Rev. 1, Page 142 of 357

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.4, ROD GROUP ALIGNMENT LIMITS

There are no specific NSHC discussions for this Specification.

Attachment 1, Volume 6, Rev. 1, Page 144 of 357

ATTACHMENT 5

ITS 3.1.5, Shutdown Bank Insertion Limits

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.1.5

A.1

ITS

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

LCO 3.1.5 3.1.3.4 All shutdown rods shall be limited in physical insertion as specified in the COLR. A.2

APPLICABILITY: MODES 1* and 2*# M.1

ACTION:

one or more shutdown banks

ACTION A With ~~a maximum of one shutdown rod~~ inserted beyond the insertion limit specified in the COLR, ~~except for surveillance testing pursuant to~~ Applicability Note ~~Specification 4.1.3.1.2,~~ within ~~one~~ two hour either: L.1

a. Restore the rod to within the insertion limit specified in the COLR, or

Add proposed Required Actions A.1.1 and A.1.2

~~b. Declare the rod to be inoperable and apply Specification 3.1.3.1.~~ L.1

Add proposed ACTION B

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 4.1.3.4 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

~~a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and~~ L.2

b. At least once per 12 hours thereafter.

~~* See Special Test Exceptions 3.10.2 and 3.10.4.~~ A.2

~~# With K_{eff} greater than or equal to 1.0~~ M.1

A.1

ITS

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

LCO 3.1.5 — 3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the COLR. A.2

APPLICABILITY: MODES 1~~*~~ and 2~~*#~~ M.1

ACTION: one or more shutdown banks

ACTION A — With ~~a maximum of one shutdown rod~~ inserted beyond the insertion limit specified in the COLR, ~~except for surveillance testing pursuant to Specification 4.1.3.1.2,~~ within ~~one~~ two hour either: L.1

Applicability Note

a. Restore the rod to within the insertion limit specified in the COLR, or

Add proposed Required Action A.1.1 and A.1.2

~~b. Declare the rod to be inoperable and apply Specification 3.1.3.1.~~ L.1

Add proposed ACTION B

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 — 4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

~~a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and~~ L.2

b. At least once per 12 hours thereafter.

~~* See Special Test Exceptions 3.10.2 and 3.10.3.~~ A.2

~~# With K_{eff} greater than or equal to 1.0~~ M.1

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 The Applicability of CTS 3.1.3.4 (Unit 1) and CTS 3.1.3.5 (Unit 2) is modified by footnote * that states "See Special Test Exceptions 3.10.2 and 3.10.4" (Unit 1) and "See Special Test Exceptions 3.10.2 and 3.10.3" (Unit 2). ITS 3.1.5 Applicability does not contain the footnote or a reference to the Special Test Exceptions.

The purpose of the footnote reference is to alert the user that Special Test Exceptions exist that may modify the Applicability of the Specification. This change is acceptable because it is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.1.3.4 (Unit 1) and CTS 3.1.3.5 (Unit 2) are applicable in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. ITS 3.1.5 is applicable in MODES 1 and 2. This changes the CTS by expanding the Applicability from MODE 2 with the reactor critical to all of MODE 2.

The purpose of CTS 3.1.3.4 (Unit 1) and CTS 3.1.3.5 (Unit 2) is to ensure that the shutdown banks are fully withdrawn prior to withdrawing the control banks in order to ensure that there is sufficient shutdown margin available to quickly shutdown the reactor. This change is acceptable because applying that requirement prior to removing the control banks and bringing the reactor critical ensures that the shutdown margin is available and is consistent with plant operation, in that the shutdown banks are completely withdrawn before beginning to withdraw the control banks and approaching criticality. This change is designated as more restrictive because it increases the conditions under which Technical Specification controls will be applied.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.4 Action (Unit 1) and CTS 3.1.3.5 Action (Unit 2) provide compensatory actions for a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR. The actions require that within one hour, either restore the rod to within the insertion limit specified in the COLR or declare the rod inoperable and apply Specification 3.1.3.1. For more than one shutdown rod beyond the insertion limit the CTS would result in an CTS 3.0.3 entry. ITS 3.1.5 ACTION A provides Required Actions for one or more shutdown banks not within limits. ITS 3.1.5 Required Action A.1.1 requires the verification that SDM is within limits in one hour and ITS 3.1.5 Required Action A.1.2 requires the initiation of boration to restore SDM to within limits in one hour (only one of these Required Actions must be performed). In addition, ITS 3.1.5 Required Action A.2 requires the restoration of shutdown banks to within limits in 2 hours. With any Required Action and associated Completion Time (of Condition A) not met the unit must be in MODE 3 in the following 6 hours. This changes the CTS by allowing more than one shutdown rod to be outside the insertion limits specified in the COLR, provides an additional hour to restore the shutdown bank or control rods to within limits, eliminates the allowance to declare the rod inoperable and take the ACTIONS of Specification 3.1.3.1, and adds the requirement to verify SDM or to initiate boration within one hour. It also eliminates the requirement to enter LCO 3.0.3 if more than one shutdown rod is inserted beyond the insertion limits.

The purpose of CTS 3.1.3.4 Action (Unit 1) and CTS 3.1.3.5 Action (Unit 2) is to ensure that the shutdown banks are fully withdrawn in order to ensure that there is sufficient shutdown margin available to quickly shutdown the reactor. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering that only a small amount of time is provided to reestablish the required features and the low probability of a DBA occurring during the repair period. Allowing an additional hour to restore one or more shutdown banks (or more than one shutdown rod) inserted below the insertion limit is appropriate as it avoids a shutdown, a unit transient, while the rod control system is not in fully working order. The ITS requires verification that the shutdown margin requirement is met or actions to restore the shutdown margin to within its limit within 1 hour, so all safety analysis assumptions are being met. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.1.3.4.a (Unit 1) and CTS 4.1.3.5.a (Unit 2) require verification that each shutdown rod is within the insertion limit specified in the COLR within 15 minutes prior to withdrawal of any control rods in control rod banks A, B, C, and D during an approach to reactor criticality. ITS 3.1.5 does not require verification that the shutdown rods are above the insertion limits within 15 minutes prior to control bank withdrawal. This changes the CTS by eliminating the requirement that the shutdown banks be verified to be above the insertion limit within 15 minutes prior to withdrawing control banks A, B, C, and D.

DISCUSSION OF CHANGES
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

The purpose of CTS 4.1.3.4.a (Unit 1) and CTS 4.1.3.5.a (Unit 2) is to verify that the shutdown banks are withdrawn above the insertion limit prior to withdrawing the control banks. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. Under the ITS Applicability of MODE 2 and the requirement of ITS LCO 3.0.4, the shutdown banks must be above the insertion limit prior to entering the ITS Applicability of MODE 2. However, it is not required to verify compliance within a specified time prior to initial control bank withdrawal. Specifying a time is not necessary to ensure that the shutdown banks are above the insertion limit prior to initial control bank withdrawal as long as the shutdown banks are withdrawn before withdrawing the control banks. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.3.4(Unit 1), LCO 3.1.3.5(Unit 2)

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

3.1.3.4 Action (Unit 1), 3.1.3.5 Action (Unit 2)

- NOTE -
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is within ~~the~~ limits ~~specified in the COLR~~.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

3.1.3.4 Action (Unit 1), 3.1.3.5 Action (Unit 2)

DOC L1

1

2

5

CTS

SURVEILLANCE REQUIREMENTS

4.1.3.4(unit 1), 4.1.3.5(unit 2)

	SURVEILLANCE	FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the insertion limits specified in the COLR.	12 hours

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

1. Changes are made to be consistent with the format of the ITS. The location of where a parameter's limits reside, whether in the COLR or an actual LCO statement, is not normally specified in the Required Action. The Required Action normally states that the parameter shall be "within limits."

2. Typographical/grammatical error corrected.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are ~~10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1)~~ INSERT 1 ① and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. ④

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have~~ There are ② four control banks and ~~at least two~~ four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown

1

INSERT 1

Plant Specific Design Criterion (PSDC) 6, "Reactor Core Design" (Ref. 1), PSDC 27, "Redundancy of Reactivity Control" (Ref. 2), PSDC 28, "Reactivity Hot Shutdown Capability" (Ref. 2), PSDC 29, "Reactivity Shutdown Capability" (Ref. 2), PSDC 30, "Reactivity Holddown Capability" (Ref. 2), PSDC 33, "Reactor Coolant Pressure Boundary Capability" (Ref. 3)

BASES

BACKGROUND (continued)

banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod. 2 2

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

1. Specified acceptable fuel design limits, or 1 3

2. RCS pressure boundary integrity, and 1

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3). 1 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1. (4)

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the ~~plant~~ unit to remain in an unacceptable condition for an extended period of time. (2)

BASES

ACTIONS (continued)

B.1

INSERT 2 5

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. 2

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28. 1

INSERT 3

4 2. 10 CFR 50.46. 1

5 3. U FSAR, Chapter 15 14 2 6

5

INSERT 2

any Required Action and associated Completion Time is not met

1

INSERT 3

1. UFSAR, Section 1.4.2.
2. UFSAR, Section 1.4.5.
3. UFSAR, Section 1.4.6.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5 BASES, SHUTDOWN BANK INSERTION LIMITS

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

4. Typographical/grammatical error corrected.

5. Change made to be consistent with the Specification.

6. The brackets have been removed and the proper plant specific information/value has been provided.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.5, SHUTDOWN BANK INSERTION LIMITS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 6

ITS 3.1.6, Control Bank Insertion Limits

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A.1

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

, sequence, and overlap limits — M.1

LCO 3.1.6 — 3.1.3.5 The control banks shall be limited in physical insertion as specified in the COLR.

A.2

APPLICABILITY: MODES ~~1#~~ and ~~2#*~~.

ACTION:

ACTION A / Applicability Note — With the control banks inserted beyond the insertion limits, ~~except for surveillance testing pursuant to Specification 4.1.3.1.2;~~ either:

Add proposed Required Actions A.1.1 and A.1.2 — M.2

ACTION A — a. Restore the control banks to within the limits within two hours, ~~or~~

~~b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR,~~ or — A.3

Add proposed ACTION B — M.1

ACTION C — c. Be in ~~HOT STANDBY~~ within 6 hours.

MODE 2 with k_{eff} < 1.0 — A.4

SURVEILLANCE REQUIREMENTS

SR 3.1.6.2 — 4.1.3.5. The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours ~~except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.~~ — L.1

Add proposed SR 3.1.6.3 — M.1

~~* See Special Test Exceptions 3.10.2 and 3.10.4~~ — A.2

Applicability — # With K_{eff} greater than or equal to 1.0.

A.1

Figure 3.1-1 intentionally deleted.

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A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k. — See ITS 3.1.1

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. — See ITS 3.1.4
If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). — See ITS Chapter 1.0

SR 3.1.6.2 b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

SR 3.1.6.1 c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5. — See ITS 3.1.1

*See Special Test Exception 3.10.1. — See ITS 3.1.1

ITS

A.1

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

, sequence, and overlap limits — M.1

LCO 3.1.6 — 3.1.3.6 The control banks shall be limited in physical insertion as specified in the COLR.

A.2

APPLICABILITY: MODES 1* and 2*#.

ACTION:

ACTION A / Applicability Note / ACTION A — With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

Add proposed Required Actions A.1.1 and A.1.2 — M.2

a. Restore the control banks to within the limits ~~within two hours~~, or

b. ~~Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR,~~ or — A.3

Add proposed ACTION B — M.1

ACTION C — c. Be in at least ~~HOT STANDBY~~ within 6 hours.

MODE 2 with $k_{eff} < 1.0$ — A.4

SURVEILLANCE REQUIREMENTS

SR 3.1.6.2 — 4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours ~~except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.~~ — L.1

Add proposed SR 3.1.6.3 — M.1

~~* See Special Test Exceptions 3.10.2 and 3.10.3~~ — A.2

Applicability — # With K_{eff} greater than or equal to 1.0.

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A.1

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A.1

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

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ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{AVG} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Delta k/k. See ITS 3.1.1

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Delta k/k, immediately initiate and continue boration at greater than or equal to 34 gpm of a solution containing greater than or equal to 6,550 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. See ITS 3.1.4 If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s). See ITS Chapter 1.0

SR 3.1.6.2 b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

SR 3.1.6.1 c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6. See ITS 3.1.1

*See Special Test Exception 3.10.1. See ITS 3.1.1

DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 The Applicability of CTS 3.1.3.5 (Unit 1) and CTS 3.1.3.6 (Unit 2) is modified by footnote * that states "See Special Test Exceptions 3.10.2 and 3.10.4" (Unit 1) and "See Special Test Exceptions 3.10.2 and 3.10.3" (Unit 2). ITS 3.1.6 Applicability does not contain the footnote or a reference to the Special Test Exceptions.

The purpose of the footnote reference is to alert the user that Special Test Exceptions exist that may modify the Applicability of the Specification. This change is acceptable because it is an ITS convention to not include these types of footnotes or cross-references. This change is designated as administrative as it incorporates an ITS convention with no technical change to the CTS.

A.3 CTS 3.1.3.5 Actions a and b (Unit 1) and CTS 3.1.3.6 Actions a and b (Unit 2) state that with the control banks inserted beyond the insertion limits, restore the control banks to within the insertion limits within two hours or reduce the THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR. ITS 3.1.6 Required Action A.2 requires the control bank to be restored to within limits within 2 hours. This changes the CTS by eliminating the explicit statement that compliance with the LCO can be restored in order to exit the Action.

This change is acceptable because the requirements have not changed. Reducing THERMAL POWER so that the insertion limits, which are a function of power, are lowered and the control bank inserted below the insertion limits comes within the limit is the same as the CTS Action a option to "restore the control banks to within the insertion limit." This change is considered administrative because the technical requirements have not changed.

A.4 CTS 3.1.3.5 Action c (Unit 1) and CTS 3.1.3.6 Action c (Unit 2) require the unit to be in HOT STANDBY within 6 hours if Actions a or b are not met. The CTS Applicability is MODE 1 and 2 with $k_{eff} \geq 1.0$. ITS 3.1.6 ACTION C requires the unit to be in MODE 2 with $k_{eff} < 1.0$ within 6 hours. This changes the CTS by requiring the plant to be in MODE 2 with $k_{eff} < 1.0$ instead of HOT SHUTDOWN (i.e., MODE 3).

This change is acceptable because the requirements have not changed. In accordance with CTS LCO 3.0.1, Actions are only required to be followed while in the Mode of Applicability. The CTS control bank physical insertion limits are applicable in MODES 1 and 2 with $k_{eff} \geq 1.0$. Therefore, under the CTS, the unit does not have to enter MODE 3 because the Applicability of the CTS LCO has been exited when in MODE 2 with $k_{eff} < 1.0$. As a result, there is no difference

DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

between the CTS and ITS requirements. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.1.3.5 (Unit 1) and CTS 3.1.3.6 (Unit 2) require the control banks to be limited in physical insertion as specified in the COLR. ITS 3.1.6 requires the control banks to be within the insertion, sequence, and overlap limits specified in the COLR. ITS 3.1.6 ACTION B provides requirements when not meeting the overlap and sequence limits, and ITS SR 3.1.6.3 requires verification of the overlap and sequence every 12 hours. This changes the CTS by adding requirements on the control bank overlap and sequence limits to the Technical Specifications.

This change is acceptable because the control bank sequence and overlap are important assumptions in the core power distribution analyses. The addition of these requirements, ACTIONS, and Surveillance Requirement provides assurance that the core power distribution is maintained within the design predictions. This change is designated as more restrictive because new requirements are added to the CTS.

M.2 The CTS 3.1.3.5 Action (Unit 1) and the CTS 3.1.3.6 Action (Unit 2) require control banks inserted beyond the insertion limits to be restored within 2 hours. ITS 3.1.6 ACTION A contains the same requirement and adds the requirement to verify the SDM is within limits or initiate boration to restore SDM to within limits within 1 hour. This changes the CTS by adding the requirement to verify SDM or to initiate boration to restore the required SDM within one hour when control banks are below the insertion limits.

This change is acceptable because it verifies that the initial conditions of the accident analyses are maintained. In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, SDM is normally ensured by adhering to the control and shutdown bank insertion limits. If the control banks are not within their insertion limits, then SDM must be verified to be within limits or actions must be initiated to restore SDM to within limits. This change is designated as more restrictive because requirements are added to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

DISCUSSION OF CHANGES
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

LESS RESTRICTIVE CHANGES

L.1 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.1.3.5 (Unit 1) and CTS 4.1.3.6 (Unit 2) require the position of each control bank to be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours. ITS SR 3.1.6.2 requires verification that each control bank insertion is within the insertion limits specified in the COLR every 12 hours. This changes the CTS by eliminating the requirement to verify the control bank insertion to be within limits every 4 hours when the Rod Insertion Limit Monitor is inoperable.

The purpose of CTS 4.1.3.5 (Unit 1) and CTS 4.1.3.6 (Unit 2) is to periodically verify that the rods are within the alignment limit specified in the LCO. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of rod position verification when the Rod Insertion Limit Monitor is inoperable is unnecessary because inoperability of the alarm does not increase the probability that the control banks are inserted below the limits. The Rod Insertion Limit Monitor alarm is for indication only; its use is not credited in any of the safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.3.5 (Unit 1), LCO 3.1.3.6 (Unit 2)

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

3.1.3.5 Action (Unit 1), 3.1.3.6 Action (Unit 2)

- NOTE -
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

3.1.3.5 Action a (Unit 1), 3.1.3.6 Action a (Unit 2)

DOC M.1

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Control bank insertion limits not met.	A.1.1 Verify SDM is within the limits ~~specified in the COLR~~.	1 hour	1
	OR		
	A.1.2 Initiate boration to restore SDM to within limit. 5	1 hour	3
	AND		
	A.2 Restore control bank(s) to within limits.	2 hours	
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits ~~specified in the COLR~~.	1 hour	1
	OR		
	B.1.2 Initiate boration to restore SDM to within limit. 5	1 hour	3
	AND		

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
DOC M.1		B.2	Restore control bank sequence and overlap to within limits.	2 hours
3.1.3.5 Action c (Unit 1), 3.1.3.6 Action c (Unit 2)	C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2 with k_{eff} < 1.0.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.1.1.1.1.c	SR 3.1.6.1	Verify estimated critical control bank position is within the insertion limits specified in the COLR.	Within 4 hours prior to achieving criticality	2
4.1.3.5 (Unit 1), 4.1.3.6 (Unit 2), 4.1.1.1.1.b	SR 3.1.6.2	Verify each control bank insertion is within the insertion limits specified in the COLR.	12 hours	
DOC M.1	SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours	

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

1. Changes are made to be consistent with changes made to the Specifications

2. SR 3.1.6.1 is clarified to state that the estimated critical control bank position must be verified to be within the "insertion limits," instead of just "limits," specified in the COLR. Many limits are specified in the COLR and the clarification is needed to avoid confusion. This is also consistent with the ISTS Bases, which clarifies that the limits to be met are the insertion limits.

3. Typographical/grammatical error corrected.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are ~~10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1)~~ and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. INSERT 1 ① ④

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have~~ There are four control banks and ~~at least two~~ four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements. ②

The control bank insertion limits are specified in the COLR. ~~An example is provided for information only in Figure B 3.1.6-1.~~ The control banks are required to be at or above the insertion limit lines. ③

INSERT 2 ③ ④

~~Figure B 3.1.6-1 also indicates how~~ the control banks are moved in an ~~overlap pattern.~~ Overlap is the distance travelled together by two control banks. ~~The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, will be at 118 steps for a fully withdrawn position of 231 steps. The fully withdrawn position is defined in the COLR.~~ INSERT 2A ② ③

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically

1 **INSERT 1**

Plant Specific Design Criterion (PSDC) 6, "Reactor Core Design" (Ref. 1), PSDC 27, "Redundancy of Reactivity Control" (Ref. 2), PSDC 28, "Reactivity Hot Shutdown Capability" (Ref. 2), PSDC 29, "Reactivity Shutdown Capability" (Ref. 2), PSDC 30, "Reactivity Holddown Capability" (Ref. 2), PSDC 33, "Reactor Coolant Pressure Boundary Capability" (Ref. 3),

4 **INSERT 2**

The control bank sequence and overlap limits are specified in the COLR. Sequencing is the order in which the banks are moved.

2 **INSERT 2A**

as described in the Background section for Bases 3.1.4.

BASES

BACKGROUND (continued)

by the Rod Control System, but can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating or diluting). (2)

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

   1. Specified acceptable fuel design limits ~~or~~ ; (5)

   2. Reactor Coolant System pressure boundary integrity, and (5)

b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. ~~3~~ 6). (2)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 4). 2 2

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5).

control bank The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis. 2

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with keff < 1.0, and ~~MODES 3, 4, and 5 is not required~~, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. 6

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.6.2. 4 This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO. 6

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position, or ; (5)

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

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

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap~~s~~ limits provides an acceptable time for evaluating and repairing minor problems without allowing the ~~plant~~ unit to remain in an unacceptable condition for an extended period of time. (2) (6)

C.1

INSERT 3 (6)

~~If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times~~, the ~~plant~~ unit must be brought to MODE 2 with $k_{eff} < 1.0$, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. (2) (2)

6 **INSERT 3**

any Required Action and associated Completion Time is not met

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28. INSERT 4 1
4 2. 10 CFR 50.46.
5 3. 11 FSAR, Chapter 15 14. 2 7
4. FSAR, Chapter [15]. 2 7
5. FSAR, Chapter [15]. 2 7

1

INSERT 4

1. UFSAR, Section 1.4.2.
2. UFSAR, Section 1.4.5.
3. UFSAR, Section 1.4.6.

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3

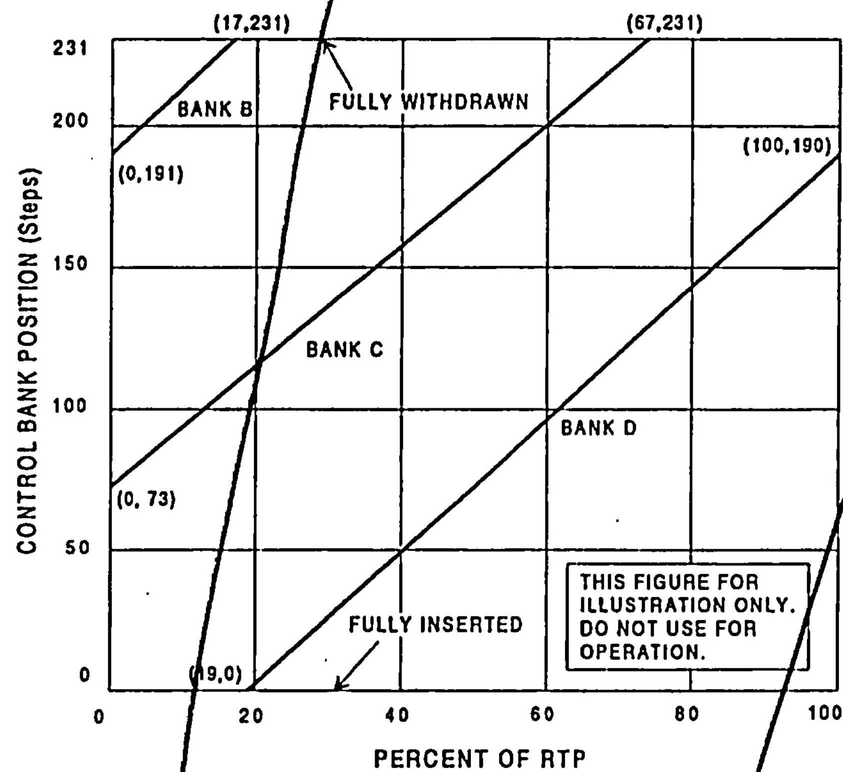

Figure B 3.1.6-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6 BASES, CONTROL BANK INSERTION LIMITS

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Since the ITS states the actual control bank insertion limits are specified in the COLR, the example is not needed in the Bases and has been deleted.

4. LCO 3.1.6 governs control bank insertion, sequence, and overlap limits. The Background section of the ITS 3.1.6 Bases discusses insertion and overlap, but does not discuss sequence. A discussion of control bank sequence is added for completeness.

5. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

6. The Bases are changed to be consistent with the ITS.

7. The brackets have been removed and the proper plant specific information/value has been provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.6, CONTROL BANK INSERTION LIMITS

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 7

ITS 3.1.7, Rod Position Indication

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS

A.1

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

LCO 3.1.7

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE ~~and capable of determining the control rod positions within the allowed rod misalignment specified in Specification 3.1.3.1.~~ LA.1

APPLICABILITY: MODES 1 and 2.

ACTION: Add proposed ACTIONS Note L.1

ACTION A

a. With a maximum of one rod position indicator channel per group inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours 4 hours L.2 and ~~immediately~~ after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

2. Reduce THERMAL POWER to less than or equal to 50% of RATED THERMAL POWER within 8 hours. L.3

ACTION C

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of the allowed rod misalignment of each other, at least once per 8 hours, or

2. Reduce THERMAL POWER to less than or equal to 50% of RATED THERMAL POWER within 8 hours. L.3

Add proposed ACTION B L.5

Add proposed ACTION D M.1

SURVEILLANCE REQUIREMENTS

~~4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within the allowed rod misalignment at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.~~ M.2

Add proposed SR 3.1.7.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS-OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.1.7 3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE ~~and capable of determining the control rod positions within the allowed rod misalignment specified in Specification 3.1.3.1.~~ LA.1

APPLICABILITY: MODES 1 and 2.

ACTION: Add proposed ACTIONS Note L.1

ACTION A

a. With a maximum of one rod position indicator channel per group inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and ~~immediately~~ 4 hours after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or L.2

2. Reduce THERMAL POWER to less than or equal to 50% of RATED THERMAL POWER within 8 hours. L.3

ACTION C

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of the allowed rod misalignment of each other, at least once per 8 hours, or

2. Reduce THERMAL POWER to less than or equal to 50% of RATED THERMAL POWER within 8 hours. L.3

Add proposed ACTION B L.5

Add proposed ACTION D M.1

SURVEILLANCE REQUIREMENTS

~~4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within the allowed rod misalignment at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.~~ M.2

Add proposed SR 3.1.7.1

DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.1.3.2 does not contain an Action to follow if the provided Actions cannot be met. Therefore, CTS 3.0.3 would be entered, which would allow 1 hour to initiate a shutdown and to be in HOT STANDBY within 7 hours. ITS 3.1.7 contains ACTION D, which states that the plant must be in MODE 3 within 6 hours if any Required Action and associated Completion Time is not met. This changes the CTS by eliminating the one hour to initiate a shutdown and, consequently, allowing one hour less for the unit to be in MODE 3.

This change is acceptable because it provides an appropriate compensatory measure for the described conditions. If any Required Action and associated Completion Time cannot be met, the unit must be placed in a MODE in which the LCO does not apply. The LCO is applicable in MODES 1 and 2. Requiring a shutdown to MODE 3 is appropriate in this condition. The one hour allowed by CTS 3.0.3 to prepare for a shutdown is not needed because the operators have had time to prepare for the shutdown while attempting to follow the Required Actions and associated Completion Times. This change is designated as more restrictive because it allows less time to shutdown than does the CTS.

M.2 CTS 4.1.3.2 requires that each rod position indicator channel be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within the allowed rod misalignment at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours. ITS 3.1.7 does not contain this requirement because it is duplicative of the requirement in CTS 4.1.3.1.1 (ITS SR 3.1.4.1). A new Surveillance has been added (ITS SR 3.1.7.1) to perform a CHANNEL CALIBRATION of each rod position channel once prior to criticality after each removal of the reactor head. This changes the CTS by adding the ITS requirement of SR 3.1.7.1.

The purpose of ITS SR 3.1.7.1 is to provide additional assurance that the rod position indicator channels are calibrated. This change is acceptable because it provides additional assurance that the rod position indicator channels are OPERABLE. This change is designated as more restrictive, because it adds a new Surveillance Requirement to the CTS.

DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.1.3.2 requires all shutdown and control rod position indicator channels and the demand position indication system to be OPERABLE and capable of determining the control rod positions within the allowed rod misalignment specified in Specification 3.1.3.1. ITS LCO 3.1.7 requires both the Rod Position Indication System and the Demand Position Indication System to be OPERABLE, but the details of what constitutes an OPERABLE system are moved to the Bases. This changes the CTS by removing details of what constitutes an OPERABLE system to the Bases.

The removal of these details, which are related to the system design capabilities, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the Rod Position Indication System and the Demand Position Indication System be OPERABLE. The details on the capability requirements of the systems do not need to appear in the specification in order for the requirement to apply. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.2 Action a covers the inoperabilities for a maximum of one rod position indicator channel per group. CTS 3.1.3.2 Action b covers the inoperabilities for a maximum of one demand position indicator per bank. ITS 3.1.7 ACTIONS are modified by a Note that states "Separate Condition entry is allowed for each rod position indicator and each demand position indicator." ITS ACTION A covers inoperabilities for one rod position indication (RPI) per group for one or more groups and ITS ACTION B covers inoperabilities for more than one RPI per group. ITS ACTION C covers the inoperabilities for one or more demand position indicators. This changes the CTS by allowing separate Condition entry for each inoperable rod position indicator and each inoperable demand position indicator instead of for a maximum of one rod position indicator channel per group and a maximum of one demand position indicator per bank. Other modifications associated with CTS 3.1.3.2 Action b (ITS 3.1.7 ACTION C) are discussed in DOC L.4, while the addition of ITS ACTION B is discussed in DOC L.5.

DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

The purpose of CTS 3.1.3.2 Action a is to provide compensatory actions for a maximum of one inoperable rod position indicator channel per group while the purpose of CTS 3.1.3.2 Action b is to provide compensatory actions for a maximum of one inoperable demand position indicator per bank. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. This change will allow separate Condition entry for each inoperable rod position indicator and each inoperable demand position indicator while the CTS do not. CTS 3.1.3.2 Action a only allows the unit to operate in this Action for only one inoperable rod position indication per group, while CTS 3.1.3.2 Action b only allows the unit to operate in this Action for a maximum of one demand position indicator per bank. The ITS will allow each inoperable rod position indication or each inoperable demand position indicator inoperability to be tracked separately. This change is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 3 – Relaxation of Completion Time)* CTS 3.1.3.2 Action a.1 states that with a maximum of one individual rod position indicator channel per group inoperable, determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and "immediately" after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position. ITS 3.1.7 Required Action A.1 states to verify the position of the rod with an inoperable position indicator by using the movable incore detectors once per 8 hours and "once within 4 hours" after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position. This changes the CTS by allowing 4 hours to verify the rod position instead of requiring the verification immediately.

The purpose of CTS 3.1.3.2 Action a.1 is to verify rod position using the movable incore detector system after the rods have been moved significantly. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. Using the movable incore detector system to determine the position of a rod cannot be performed immediately. Four hours is a reasonable time to use the movable incore detector system to measure the core flux around the control rod and analyze the data to determine the control rod position. This short period of time to determine the position will not result in significant perturbation of the core power distribution if the rod is misaligned, and since the probability of a DBA or transient that would be affected by the potentially misaligned rod is very low for

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DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

the short period of time allowed to determine the rod position. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS 3.1.2.1 Action a.2 and Action b.2 require the unit to reduce THERMAL POWER to less than 50% of RATED THERMAL POWER. ITS 3.1.7 Required Actions A.2 and C.2 require the unit to be at a THERMAL POWER of less than or equal to 50% RATED THERMAL POWER under the same conditions. This changes the CTS by allowing a unit to be at 50% RATED THERMAL POWER instead of less than 50% RATED THERMAL POWER.

The purpose of CTS 3.1.2.1 Action a.2 and Action b.2 is to place the unit into a condition where rod position or rod position demand is not significantly affecting core peaking factors. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. This change is acceptable since with THERMAL POWER at 50% RATED THERMAL POWER, rod position and rod position demand do not significantly affect core peaking factors. The specified THERMAL POWER is consistent with safe operation under the specified Condition. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.4 Not Used.

L.5 *(Category 4 – Relaxation of Required Action)* CTS 3.1.3.2 does not have an action for when more than one rod position indicator channel is inoperable per group. CTS 3.0.3 would be entered in this condition. CTS 3.0.3 requires a shutdown to HOT STANDBY within 7 hours. ITS 3.1.7 ACTION B applies when more than one RPI per group is inoperable and requires the rods to be placed under manual control immediately, monitoring and recording of RCS T_{avg} once per hour, and restoration of all but one RPI to OPERABLE status within 24 hours. This changes the CTS by allowing operation for an additional 24 hours with more than one RPI per group inoperable.

The purpose of ITS 3.1.7, ACTION B is to provide time to repair inoperable RPIs before requiring a plant shutdown. This change is acceptable because the ITS Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant indications. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. Providing time to repair multiple inoperable

DISCUSSION OF CHANGES
ITS 3.1.7, ROD POSITION INDICATION

RPIs before requiring a shutdown is reasonable as the safest course of action with inoperable RPIs is to not move the control rods. The compensatory measures ensure that the rods are not moved unintentionally and monitor rod position using other indications. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.3.2

LCO 3.1.7 The ~~Digital~~ Rod Position Indication (~~D~~RPI) System and the Demand Position Indication System shall be OPERABLE. 1 2

APPLICABILITY: MODES 1 and 2.

ACTIONS

DOC L.1

- NOTE -
Separate Condition entry is allowed for each ~~inoperable~~ rod position indicator and each demand position indicator. 8

Action G

DOC L.5

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ~~D~~RPI per group inoperable for one or more groups.	A.1	Verify the position of the rod~~s~~ with inoperable an position indicator~~s~~ indirectly by using movable incore detectors.	Once per 8 hours INSERT 1
	OR		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. More than one ~~D~~RPI per group inoperable.	B.1	Place the control rods under manual control.	Immediately
	AND		
	B.2	Monitor and ~~R~~record ~~RCS~~ Reactor Coolant System T_{avg}.	Once per 1 hour
	AND		

2 4 3 2 5

INSERT 1

AND

Once within 4 hours after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position

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CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
	B.3 Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors. AND	Once per 8 hours	3
	B.4 3 Restore inoperable position indicators to OPERABLE status such that a maximum of one [DRPI] per group is inoperable. a	24 hours	3 5 2
C. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors. OR C.2 Reduce THERMAL POWER to ≤ 50% RTP.	[4] hours 8 hours	3
Action 6 C D. One demand position indicator per bank inoperable for one or more banks.	D.1.1 Verify by administrative means all [DRPI]s for the affected bank(s) are OPERABLE. c AND D.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart. OR	Once per 8 hours Once per 8 hours within the required rod misalignment limits	2 3 4 4 6

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C	~~B~~.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours	③
DOC M.1 ~~C~~ D. Required Action and associated Completion Time not met.	~~C~~ D.1 Be in MODE 3.	6 hours	③

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
DOC M.2	SR 3.1.7.1 ~~Verify each [D]RPI agrees within [12] steps of the group demand position for the [full indicated range] of rod travel.~~ INSERT 2 7	Once prior to criticality after each removal of the reactor head

7 **INSERT 2**

-NOTE-
The sensor may be excluded.

Perform a CHANNEL CALIBRATION of each RPI.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7, ROD POSITION INDICATION

1. CNP utilizes an analog rod position indication system. Therefore, reference to a digital rod position indication system have been removed.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. ISTS 3.1.7 ACTION C has two Required Actions that are connected with an OR. However, the stated Completion Times for these two Required Actions are different (4 hours and 8 hours, respectively). Due to the convention in the ISTS as described in Section 1.3, the two Completion Times associated with the two Required Actions OR logical connector must be the same, since either Required Action can be chosen. Therefore, to be consistent with the format of the ISTS, ISTS 3.1.7 ACTION C has been deleted and a new, conditional Completion Time has been added to Required Action A.1. This ensures that the intent of the ISTS is maintained, in that a verification of the position of the rod with an inoperable position indicator is still being performed once within 4 hours after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position. In addition, since the unit is in both Conditions A and B when more than one rod position indicators per group are inoperable, and Required Action A.1 requires the identical position check required by Required Action B.3, there is no reason to include the position check as Required Action B.3. This is also consistent with the format of the ISTS. Appropriate renumbering changes have also been made due to these deletions.

4. The words in ISTS Required Action A.1, ISTS Required Action D.1.1 (ITS Required Action C.1.1), and ISTS Required Action D.1.2 (ITS Required Action C.1.2) have been modified to be singular, versus plural, when referring to a rod or a bank. This has been done since the ACTIONS Note allows separate Condition entry for each rod position indicator and each demand position indicator; thus the Required Action only applies to the individual rod or bank whose indicator in inoperable.

5. Typographical/grammatical error corrected.

6. The ISTS Required Action D.1.2 alignment criteria has been revised to be consistent with the current licensing basis requirements. The CTS allows the alignment criteria to vary as a function of Allowable Power Level (changed to vary as a function of $F_Q^W(Z)$ as described in the ITS 3.1.4 DOCs) at THERMAL POWER levels $\geq$ 85% as indicated in CTS Figure 3.1.4-1. This change to the Required Action has been made consistent with the allowances in License Amendments 193 (Unit 1) and 179 (Unit 2) based on a Letter from the NRC dated March 15, 1995 (as modified in the ITS 3.1.4 DOCs). The alignment criteria is specified in ITS 3.1.4.

7. The ISTS requirement to verify each RPI agrees within 12 steps of the group demand position for the full indicated range of rod travel prior to criticality after each removal of the reactor vessel head is replaced with the requirement to perform a CHANNEL CALIBRATION of each RPI, except for the sensor. Because of the thermal drift characteristics of the CNP RPIs, performing a full range comparison of RPI and demand position before criticality is not useful, as the RPI response will change with RPI temperature. The ITS requires a CHANNEL CALIBRATION of each RPI, which involves calibrating the electronics to known input voltages. Actual RPI position is adjusted for thermal drift.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7, ROD POSITION INDICATION

8. Change made to be consistent with similar Notes in other places in the ISTS.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Rod Position Indication

INSERT 1

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. (1) (2)

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System. (3)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for

1 **INSERT 1**

Plant Specific Design Criterion (PSDC) 12 (Ref.1), instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor operating variables.

BASES

BACKGROUND (continued)

each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The [D]RPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube ~~with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the [D]RPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the [D]RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches).~~ With an indicated deviation of ~~12~~ steps between the group step counter and [D]RPI, the maximum deviation between actual rod position and the demand position could be ~~24~~ steps, ~~or 15 inches~~. (3) (4) (4)(3)

INSERT 2

up to 18

30

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

ion ies

~~The control~~ rod position indicat~~or channels~~ satis~~fy~~ Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident. (2)

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4

INSERT 2

The RPI System is capable of monitoring rod position within at least $\pm$ 12 steps.

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BASES

LCO

LCO 3.1.7 specifies that ~~one~~ the ~~[D]~~RPI System and ~~one~~ the Bank Demand Position Indication System be OPERABLE ~~for each control rod~~. For the ~~control~~ rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following: 2 2

~~a. The [D]RPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits,"~~ 8

a ~~b~~. For the ~~[D]~~RPI System there are no failed coils ~~;~~ and 8 3 6

b ~~c~~. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the ~~[D]~~RPI System. 8 5

~~The 12 step agreement limit between the Bank Demand Position Indication System and the [D]RPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.~~ 8

~~A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).~~

These requirements ensure that ~~control~~ rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. 2

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the ~~[D]~~RPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the ~~plant~~ unit. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. 2 4

BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

INA

When one [D]RPI channel per group fails, the position of the rod may still be determined indirectly by use of the movable incore detectors. ~~The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved.~~ Based on experience, normal power operation does not require excessive movement of banks. ~~If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required.~~ Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. ③ ② ⑦ INSERT 3A ⑨ ②

INSERT 3 ②

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2). ② ④

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant ~~systems and allowing for rod position determination by Required Action A.1 above.~~ unit ④

B.1, B.2, B.3, and B.4 ③ ⑤

INA ⑤

When more than one [D]RPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. Placing the Rod Control System in manual assures unplanned rod motion ~~will~~ not occur. Together with the indirect position determination available via movable incore detectors will minimize the potential for rod misalignment. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition. ③ ② will ⑥ ⑥

2

INSERT 3

If a rod has been significantly moved (in excess of 24 steps in one direction, since the position was last determined), Required Action A.1 is still appropriate but must be initiated promptly to begin verifying that the rod is still properly positioned, relative to their group positions. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod position with inoperable position indicator indirectly by using movable incore detectors. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. One RPI per group inoperable for one or more groups; and

b. A rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position.

If at any time during the existence of Condition A (one RPI per group inoperable for one or more groups) a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position, this Completion Time begins to be tracked.

9

INSERT 3A

The only exception is if the RPI for rod H-8 is inoperable, since rod H-8 is directly in the center of the core and its position cannot be determined indirectly by use of the movable incore detectors. In this condition, Required Action of A.2 below is required.

BASES

ACTIONS (continued)

Monitoring and recording reactor coolant T_{avg} help assure that significant changes in power distribution and SDM are avoided. The once per hour Completion Time is acceptable because only minor fluctuations in RCS temperature are expected at steady state plant operating conditions.

(Required Action A.1)

The position of the rods may be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Verification of control rod position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the [D]RPI System to operation while avoiding the plant challenges associated with the shutdown without full rod position indication. (2) INSERT 4 unit (3) (4) (6)

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 and C.2 below is required. (2)

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.1, as applicable are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within [4] hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of [4] hours provides an acceptable period of time to verify the rod positions. (2)

(C) D.1.1 and D.1.2 (2)

With one demand position indicator per bank inoperable, the rod positions can be determined by the [D]RPI System. Since normal power operation does not require excessive movement of rods, verification by (3)

2 **INSERT 4**

and once within 4 hours after a rod with an inoperable position indicator has been moved in excess of 24 steps in one direction since the last determination of the rod's position

BASES

ACTIONS (continued) INSERT 5

administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ~~≤ 12 steps apart~~ within the allowed Completion Time of once every 8 hours is adequate. (5)

C ~~D~~.2 (2)

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. ~~3~~ 2). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP. (4)

D ~~E~~.1 (2)

If ~~the~~ only Required Action ~~cannot be completed within the~~ and associated Completion Time is not met, the ~~plant~~ unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. (2) (4) (4)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

~~Verification that the [D]RPI agrees with the demand position within [12] steps ensures that the [D]RPI is operating correctly. Since the [D]RPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~ (3)

~~This surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.~~ INSERT 6 (5)

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 13.~~ INSERT 7 (1)
2. U FSAR, Chapter ~~[15]~~ 14. (4) (3)
3. ~~FSAR, Chapter [15].~~ (4)

5 **INSERT 5**

within the required rod misalignment limits

5 **INSERT 6**

SR 3.1.7.1 is the performance of a CHANNEL CALIBRATION for each RPI channel.

The calibration verifies the accuracy of each RPI channel. The Frequency of once prior to criticality after each removal of the reactor head is based on operating experience and considers channel reliability.

The SR is modified by a Note stating that the sensors are excluded from the CHANNEL CALIBRATION. This is acceptable since the RPIs are adjusted as necessary to compensate for thermal effects.

1 **INSERT 7**

UFSAR, Section 1.4.3.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7 BASES, ROD POSITION INDICATION

1. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

2. The Bases are changed to reflect the Specification.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

5. The Bases are changed to reflect changes made to the Specification.

6. Typographical/grammatical error corrected.

7. The description in the Bases of ACTIONS A.1 of the alternate manner to perform Required Action A.1 (by verifying LCO 3.2.1, LCO 3.2.2, and LCO 3.1.1 are met every 8 hours) has been deleted. This option will not be used at CNP.

8. The requirement that the RPI indicates within the agreement limit of the group step counter demand position has been deleted since the requirement is already covered by ITS LCO 3.1.4. If the agreement limit is not met, then the ACTIONS of LCO 3.1.4, "Rod Group Alignment Limits," should be entered. As written in these Bases, both the ACTIONS of ITS LCO 3.1.4 and ITS LCO 3.1.7 would have to be entered if not within the agreement limit. The appropriate ACTIONS are those of ITS LCO 3.1.4. ITS LCO 3.1.7 should only cover the actual RPI System, not the agreement limits.

9. A modification has been added to Required Action A.1 that excepts the use of movable incore detectors for determining the position for rod H-8 therefore requiring entry into Required Action A.2 (reduce THERMAL POWER to $\leq$ 50% RTP in 8 hours). Rod H-8 is located directly in the center of the core and therefore has no symmetric rod to compare to for determining relative position.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.7, ROD POSITION INDICATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 8

ITS 3.1.8, PHYSICS TESTS Exceptions - MODE 2

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.1.8

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.10 SPECIAL TEST EXCEPTIONS

and the number of required channels for LCO 3.3.1 Functions 2, 3, 6, and 18.d may be reduced to 3 — A.2

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

LCO 3.1.8 3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4 and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

Add LCO 3.4.2 — L.1

Add proposed LCO 3.1.8.a

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

~~b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.~~ — A.3

Add proposed LCO 3.1.8.b — M.1

APPLICABILITY: MODE 2.

During PHYSICS TEST initiated in — A.4

ACTION:

ACTION B With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

Add proposed ACTION A — M.1

Add proposed ACTIONS C and D — L.1

SURVEILLANCE REQUIREMENTS

SR 3.1.8.2 4.10.4.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per ~~hour~~ during PHYSICS TESTS.

30 minutes — M.2

~~4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating PHYSICS TESTS.~~ — L.2

Add proposed SR 3.1.8.1 — L.1

Add proposed SR 3.1.8.3 — M.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.10 SPECIAL TEST EXCEPTIONS

PHYSICS TEST

and the number of required channels for LCO 3.3.1 Functions 2, 3, 6, and 18.d may be reduced to 3 — A.2

LIMITING CONDITION FOR OPERATION

LCO 3.1.8 3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

Add LCO 3.4.2 — L.1

Add proposed LCO 3.1.8.a — L.1

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and.

~~b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.~~ — A.3

Add proposed LCO 3.1.8.b — M.1

APPLICABILITY: MODE 2.

During PHYSICS TEST initiated in — A.4

ACTION:

ACTION B With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

Add proposed ACTION A — M.1

Add proposed ACTIONS C and D — L.1

SURVEILLANCE REQUIREMENTS

SR 3.1.8.2 4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per ~~hour~~ during PHYSICS TESTS.

30 minutes — M.2

~~4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating PHYSICS TESTS.~~ — L.2

Add proposed SR 3.1.8.1 — L.1

Add proposed SR 3.1.8.3 — M.1

DISCUSSION OF CHANGES
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) state that the limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS. ITS LCO 3.1.8 includes an allowance to reduce the required number of channels for ITS LCO 3.3.1, "RTS Instrumentation," Function 2 (Power Range Neutron Flux), Function 3 (Power Range Neutron Flux Rate), Function 6 (Overtemperature ΔT), and Function 18.d (Power Range Neutron Flux, P-10), from "4" to "3." This changes CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) by adding an allowance to reduce the number of required RTS channels from "4" to " 3" for the specified Functions.

The purpose of CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) is to allow some flexibility during the performance of PHYSICS TESTS while ensuring appropriate limitations are in place to help ensure safe operation. This change is acceptable because the minimum channels required for OPERABILITY for these RTS Functions in CTS Table 3.3-1 is currently "3." This allowance is needed since the "Required Channels" in ITS 3.3.1, Reactor Trip System Instrumentation, is "4." This change from the CTS is discussed in the Discussion of Changes for ITS 3.3.1. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) state that the limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS provided the Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set at ≤ 25% of RATED THERMAL POWER. ITS 3.1.8 states that the requirement of certain Specifications may be suspended but contains no requirements on the Intermediate and Power Range Channels. The ITS contains the same requirements on the Intermediate and Power Range Channels in ITS LCO 3.3.1. This changes the CTS by eliminating the requirement that the Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set at ≤ 25% of RATED THERMAL POWER from the test exception.

This change is acceptable because the Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are contained in ITS LCO 3.3.1, "RTS Instrumentation." Repeating that requirement in the test exception LCO is unnecessary. This change is designated administrative as it eliminates a repeated requirement from the CTS, resulting in no technical change to the CTS.

DISCUSSION OF CHANGES
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

A.4 CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) are applicable in MODE 2. ITS 3.1.8 is applicable "During PHYSICS TESTS initiated in MODE 2." This changes the CTS such that the Specification is applicable in MODE 2 only when a PHYSICS TEST is initiated.

The purpose of the ITS 3.1.8 Applicability is to ensure that the Actions contained in the Specification are followed. The wording of the CTS appears to be contradictory because, if THERMAL POWER exceeds 5% RTP, then the test exception Specification Applicability is exited and the Actions no longer apply. However, it is clear that the CTS Action should be applied if THERMAL POWER exceeds 5% RTP and PHYSICS TESTS are in progress. The ITS Applicability eliminates this apparent contradiction and allows the test exception Conditions and Required Actions to be applied when the LCO is not met. This is consistent with the wording of the CTS Action. This change is designated as administrative because it clarifies the current wording of the Specification with no change in intent.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) state that limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS and provides restrictions that must be followed when utilizing the CTS exception. ITS 3.1.8 adds a requirement that SHUTDOWN MARGIN must be within the limits provided in ITS LCO 3.1.1 for MODE 2 with $k_{eff} < 1.0$. A Surveillance (SR 3.1.8.3) to verify the SHUTDOWN MARGIN every 24 hours and an ACTION (ACTION A) to follow if the SHUTDOWN MARGIN limit is not met are also added. This changes the CTS by imposing an additional requirement on the application of the test exception LCO.

This change is acceptable because it imposes reasonable restrictions on the performance of PHYSICS TESTS when the control rod and RCS minimum temperature Specifications are allowed to be violated. The Bases for ITS 3.1.1, "SHUTDOWN MARGIN," state that in MODE 2 with $k_{eff} > 1.0$, the SHUTDOWN MARGIN is ensured by compliance with the rod insertion limit Specifications. Under the test exception, those control rod insertion limits are allowed to be violated. Therefore, additional actions must be taken to ensure that sufficient SHUTDOWN MARGIN is available to shutdown the reactor and keep it subcritical if needed when in MODE 2 with $k_{eff} > 1.0$. This change is designated as more restrictive because it imposes additional restrictions not found in the CTS.

M.2 CTS 4.10.4.1 (Unit 1) and CTS 4.10.3.1 (Unit 2) require THERMAL POWER to be verified to be < 5% RTP once per hour. ITS SR 3.1.8.2 requires the same verification be performed every 30 minutes. This changes the CTS by increasing the Frequency of the THERMAL POWER verification.

This change is acceptable because the increased Frequency is consistent with similar verifications performed in the Specification. ITS SR 3.1.8.1, which verifies that the RCS lowest loop average temperature is ≥ 531°F, is also performed every 30 minutes. THERMAL POWER is a parameter readily available in the control room, so imposition of this more stringent requirement will have no effect on safety. This change is designated as more restrictive because a Surveillance will be performed more frequently in the ITS than in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) state that limitations of certain Specifications may be suspended during the performance of PHYSICS TESTS. ITS 3.1.8 provides an additional exception to LCO 3.4.2, "RCS Minimum Temperature for Criticality," provided the RCS lowest loop average temperature is ≥ 531°F. A Surveillance to verify the RCS lowest loop average temperature is ≥ 531°F every 30 minutes (proposed SR 3.1.8.1) has been added. In addition, ACTION C has been added to cover the situation when RCS lowest loop average temperature is not within limit. The Required Action is to restore RCS lowest loop average temperature to within limit within 15 minutes. If this is not met, then ACTION D requires the unit to be in MODE 3 within 15 minutes. This changes the CTS by allowing the suspension of LCO 3.4.2, "RCS Minimum Temperature for Criticality." However, it places a limitation on the RCS lowest loop average temperature that is allowed.

The purpose of CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) is to allow some flexibility during the performance of PHYSICS TESTS, while ensuring appropriate limitations are in place to help maintain safe operation. This change is acceptable because the LCO requirements continue to ensure that the process variables are maintained consistent with the safety analyses and licensing basis. This changes the CTS by allowing the suspension of LCO 3.4.2, "RCS Minimum Temperature for Criticality." However, it places a limitation on the RCS lowest loop average temperature that is allowed. CTS 3.1.1.5 (ITS 3.4.2, "RCS Minimum Temperature for Criticality") requires the RCS lowest operating loop temperature to be ≥ 541°F. Therefore, this change reduces the temperature for criticality by 10°F during the performance of PHYSICS TESTS. This is necessary to help facilitate the performance of certain tests, such as the determination of the Isothermal Temperature Coefficient. The lower limit on RCS average temperature is provided in the test exception LCO to ensure that the RCS temperature stays within the analyzed range. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

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DISCUSSION OF CHANGES
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

L.2 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.10.4.2 (Unit 1) and CTS 4.10.3.2 (Unit 2) require that CHANNEL FUNCTIONAL TESTS be performed on each Intermediate and Power Range channel prior to initiating PHYSICS TESTS. ITS SR 3.3.1.8 for the Power Range channels and ITS SR 3.3.1.10 for the Intermediate Range channels require the tests to be performed every 92 days and every 184 days, respectively. Since ITS 3.3.1 requires these channels to be OPERABLE in MODE 2 and in MODE 2 above the P-6 Interlock, respectively, this effectively ensures the tests are performed within their required Frequency prior to entering MODE 2 (i.e., prior to performing the PHYSICS TESTS). This changes the CTS by eliminating the time period prior to initiation of PHYSICS TESTS within which the testing must be performed.

The purpose of CTS 3.10.4 (Unit 1) and CTS 3.10.3 (Unit 2) is to allow the performance of PHYSICS TESTS on the reactor. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The performance of the normally scheduled CHANNEL OPERATIONAL TEST is sufficient to ensure the equipment is OPERABLE. LCO 3.3.1 requires a CHANNEL OPERATIONAL TEST on the Power Range channels (SR 3.3.1.8) every 92 days and on the Intermediate Range channels (SR 3.3.1.10) every 184 days. These Frequencies have been determined to be sufficient for verification that the equipment is working properly. The initiation of PHYSICS TESTS does not affect the ability of the equipment to perform its function, does not affect the trip setpoints or the RTS trip capability, and does not invalidate the previous Surveillances. Therefore, requiring this testing to be performed at a fixed time before the initiation of PHYSICS TESTS has no benefit. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

LCO 3.10.4 (unit 1), LCO 3.10.3 (unit 2)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3, "Moderator Temperature Coefficient,"
LCO 3.1.4, "Rod Group Alignment Limits,"
LCO 3.1.5, "Shutdown Bank Insertion Limits,"
LCO 3.1.6, "Control Bank Insertion Limits," and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

1

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2,3,6, and 18.e, may be reduced to 3, provided that:

4

a. RCS lowest loop average temperature is ≥ 531°F, 2

b. SDM is within the limits specified in the COLR, and 1

c. THERMAL POWER is ≤ 5% RTP. 5

INSERT 1 3

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

DOC M.1

Action

DOC L.1

3 **INSERT 1**

for MODE 2 with k_{eff} < 1.0 specified in LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

Doc L.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.1.8.1~~	~~Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per [SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1].~~	~~Prior to initiation of PHYSICS TESTS~~ 4
SR 3.1.8.2 1	Verify the RCS lowest loop average temperature is ≥ [531]°F.	30 minutes 4 2
SR 3.1.8.3 2	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes 4 5
SR 3.1.8.4 3	Verify SDM is within the limits specified in the COLR. for MODE 2 with $k_{eff} < 1.0$ LCO 3.1.1	24 hours 4 3

DOC L.1

4.10.4.1 (Unit 1), 4.10.3.1 (Unit 2)

DOC M.1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes are made to accurately reflect the requirement that must met, since the COLR lists more than one SDM limit.

4. ISTS SR 3.1.8.1 requires a CHANNEL OPERATIONAL TEST be performed on the Intermediate and Power Range channels "prior to initiation of PHYSICS TESTS." However, no finite time as to how soon prior to the PHYSICS TESTS is stated. The ITS Applicability for the Intermediate and Power Range channels includes MODE 2 above the P-6 Interlock and MODE 2, respectively, thus the normal, periodic Frequencies for SR 3.3.1.10 and SR 3.3.1.8 must be met prior to entering or soon after entering MODE 2. Therefore, the normal periodic Frequencies already ensure the "prior to initiation of PHYSICS TESTS" is met, and ISTS SR 3.1.8.1 is not necessary and has been deleted. Due to this deletion, the remaining SRs have been renumbered. In addition, ISTS LCO 3.1.8 references LCO 3.3.1 Function 18.e. In ITS Table 3.3.1-1, this Function has been renumbered as Function 18.d.

5. ISTS LCO 3.1.8.c and ISTS SR 3.1.8.3 have been revised to require THERMAL POWER $\leq$ 5% RTP. TSTF-14, Rev. 4, approved this change on May 2, 1997, but it was not properly adopted in NUREG-1431, Rev.2.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the ~~plant~~ unit. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2). ①

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed,
b. Validate the analytical models used in the design and analysis,
c. Verify the assumptions used to predict unit response,
d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design, and
e. Verify that the operating and emergency procedures are adequate.

③ ②

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

BASES

BACKGROUND (continued)

Some of ~~T~~he PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below: 8

a. Critical Boron Concentration - Control Rods Withdrawn; 3

~~b. Critical Boron Concentration - Control Rods Inserted;~~ 7

b ~~c.~~ Control Rod Worth; 3 2

c ~~d.~~ Isothermal Temperature Coefficient (ITC); and

d ~~e.~~ Neutron Flux Symmetry. 1

The first three ~~four~~ tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. 1

a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs. 4

~~b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% Δk/k when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate~~ 4

BASES

BACKGROUND (continued)

LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," or LCO 3.1.6, "Control Bank Insertion Limits."

b a. The Control Rod Worth Test (i.e., Rod Bank Worth Test) is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. INSERT 1 This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6. (1) (4)

c d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality." (1)

1 4 **INSERT 1**

the Dynamic Rod Worth Measurement Method (Ref. 5), moves the selected control bank over its entire length of travel. The worth of the bank is inferred from the change in the flux level upon insertion of the bank.

BASES

BACKGROUND (continued)

(d) e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP. (4)

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The ~~methods for development~~ (purpose) of the LCOs that are excepted by this LCO are described in the ~~Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5)~~. (1) (1)

Bases for the individual LCOs

The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations. (5)

The (4) FSAR defines requirements for initial testing of the facility, (5) including PHYSICS TESTS. Tables 14.1-1 and 14.1-2 (13.3-1) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (1997) (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 531°F, and SDM is within the limits provided in ~~the COLR~~ LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" for MODE 2 with $k_{eff} < 1.0$. (1) (3) (1) (6)

INSERT 1A

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are

1

INSERT 1A

and WCAP-13360-P-A, Revision 1 (Ref. 5)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases. (1)

Reference 3 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown ~~rods to be positioned outside of~~ banks their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met. (1)

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended INSERT 2 during the performance of PHYSICS TESTS provided: (7)

a. RCS lowest loop average temperature is ≥ 531°F; (4) (2)

b. SDM is within the limits provided in ~~the COLR~~ LCO 3.1.1; and (6) (2)

for MODE 2 with keff < 1.0 c. THERMAL POWER is ≤ 5% RTP. (9)

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The Applicability is stated as "During PHYSICS TESTS initiated in MODE 2" to ensure that the 5% ~~RTP~~ RTP maximum power level is not exceeded. Should the THERMAL POWER EXCEED 5% ~~RTP~~ RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions. (8)

7

INSERT 2

and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 18.d may be reduced to 3

BASES

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant unit (1) conditions. Boration will be continued until SDM is within limit. (7)

INSERT 3

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1 (8)

When THERMAL POWER is > ≥ 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant unit to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses. (1)

D.1

If the Required Action of Condition C is not met and ~~cannot be completed within the~~ associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant unit must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant unit systems. (7) (1) (1)

SURVEILLANCE REQUIREMENTS

~~SR 3.1.8.1~~ (6)

~~The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is~~

7

INSERT 3

In addition, the PHYSICS TEST exception must be suspended within 1 hour.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

~~performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.~~ 6

SR 3.1.8.~~2~~ 1 6

Verification that the RCS lowest loop T_{avg} is ≥ 531°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.~~3~~ 2 6

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the ~~plant~~ unit is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated. 9 1

SR 3.1.8.~~4~~ 3 6

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;
b. Control Bank position;
c. RCS average temperature;
d. Fuel burnup based on gross thermal energy generation;
e. Xenon concentration;
f. Samarium concentration;
g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);
h. Moderator Defect, when above the POAH; and Moderator Temperature

1 2 8

BASES

SURVEILLANCE REQUIREMENTS (continued)

i. Doppler Defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August, 1978.
4. ANSI/ANS-19.6.1-~~1985~~ 1997, ~~December 13, 1985~~ August 22, 1997. (1)
5. ~~WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.~~ (1)
~~6~~ 5. WCAP-~~11618, Including Addendum 1, April 1989~~. (1)

INSERT 4

1

INSERT 4

13360-P-A, "Westinghouse Dynamic Rod Worth Measurement Technique," Revision 1, October 1998.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8 BASES, PHYSICS TESTS EXCEPTIONS – MODE 2

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

3. The description of PHYSICS TESTS required for reload fuel cycles is revised to be consistent with the current guidelines, ANSI/ANS 19.6.1-1997, and the CNP startup physics testing program.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. The Applicable Safety Analyses description about "other tests" has been deleted since ITS 3.1.8 allows the suspension of the LCOs only for PHYSICS TESTS.

6. The Bases are changed to reflect changes made to the Specifications.

7. The Bases are revised to be consistent with the Specification.

8. Editorial/grammatical error corrected.

9. The LCO and SR 3.1.8.3 Bases Sections have been revised to require THERMAL POWER $\leq$ 5% RTP. TSTF-14, Rev. 4, approved this change on May 2, 1997, but it was not properly adopted in NUREG-1431, Rev. 2.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.8, PHYSICS TESTS EXCEPTIONS – MODE 2

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 9

Relocated/Deleted Current Technical Specifications (CTS)

CTS 3/4.1.1.3, Boron Dilution

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

CTS 3/4.1.1.3

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be greater than or equal to 2000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.*

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system less than 2000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

L.1

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be greater than or equal to 2000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying greater than or equal to 2000 gpm through the reactor coolant system.

*For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

D. C. COOK - UNIT 1 3/4 1-4 AMENDMENT NO. 120

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be greater than or equal to 2000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.*

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system less than 2000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be greater than or equal to 2000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying greater than or equal to 2000 gpm through the reactor coolant system.

L.1

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

D. C. COOK - UNIT 2 3/4 1-4 AMENDMENT NO. 82, 107

DISCUSSION OF CHANGES
CTS 3/4.1.1.3, BORON DILUTION

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.1.1.3 requires the flow rate of reactor coolant through the Reactor Coolant System (RCS) to be greater than or equal to 2000 gpm whenever a reduction in RCS boron concentration is being made. With the flow rate not within limit, immediate suspension of all operations involving a reduction in boron concentration is required. CTS 4.1.1.3 requires the RCS flow rate to be monitored prior to the start of a reduction in the RCS born concentration. The ITS does not include this Specification. This changes the CTS by eliminating this Specification.

The purpose of CTS 3.1.1.3 is to ensure there is enough flow to support adequate mixing, prevent stratification, and prevent and ensure that reactivity changes will be gradual during boron concentration reductions in the RCS. This flow rate will circulate the RCS volume in approximately 30 minutes. Therefore, the reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

This change is acceptable since the ITS contains several Specifications, each applicable during different MODES of operations, that require a certain number of RCS and/or residual heat removal (RHR) loops to be OPERABLE and in operation regardless of whether or not a reduction in RCS boron concentration is being made. These ITS Specifications also include the appropriate Surveillance to ensure the loops are OPERABLE and in operation. The flow limit is not included in most of the ITS Specifications because the capacity of the RCS pumps is significantly greater than 2000 gpm and because operation of the RHR System is controlled by plant operating procedures to ensure adequate flow. The reactor coolant flow rate of 2000 gpm is retained for MODE 6 operations as indicated in ITS SR 3.9.4.1 and SR 3.9.5.1.

DISCUSSION OF CHANGES
CTS 3/4.1.1.3, BORON DILUTION

In MODES 1 and 2, if any RCS loop is not OPERABLE and in operation, ITS LCO 3.4.4 ACTION A requires the unit to be in MODE 3 within 6 hours. If the unit is operating in MODES 3, 4, and 5 (with the RCS loops filled) and the required loops are not in operation, the associated ITS LCOs provide limitations that prohibit operations that would cause introduction of coolant with boron concentration less than required to meet SDM of ITS LCO 3.1.1. If the required loop is not in operation in MODE 5 (with the RCS loops not filled), ITS LCO 3.4.8 prohibits operations that can cause introduction of coolant with boron concentration less than required to meet ITS LCO 3.1.1 and prohibits draining operations that could further reduce the RCS water volume. If the unit is operating in MODE 6 with high reactor water level and the required loop is not in operation, ITS LCO 3.9.4 prohibits operations that would cause introduction of coolant with boron concentration less than required to meet ITS LCO 3.9.1. If the unit is operating in MODE 6 with low reactor water level and the required loops are not in operation, ITS LCO 3.9.5 prohibits operation that would cause introduction of coolant with boron concentration less than required to meet ITS LCO 3.9.1 and prohibits draining operations which can further reduce the RCS water volume. Since the requirements have been included in various Specifications, the change is appropriate. This change is designated as less restrictive because less stringent LCO requirements (explicit flow rates) are being applied in the ITS than were applied in the CTS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.1.3, BORON DILUTION

There are no specific NSHC discussions for this Specification.

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CTS 3/4.1.2.1, Flow Paths - Shutdown

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2. R.1

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperatures of the areas containing the flow path components from the boric acid tank to the blending tee are greater than or equal to 63°F when a flow path from the boric acid tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2. R.1

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperatures of the areas containing the flow path components from the boric acid tank to the blending tee are greater than or equal to 63°F when a flow path from the boric acid tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

DISCUSSION OF CHANGES
CTS 3/4.1.2.1, FLOW PATHS - SHUTDOWN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.1 provides requirements on the boration systems flow paths during shutdown. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.1 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Flow Paths - Shutdown Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Flow Paths - Shutdown Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Flow Paths - Shutdown Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-

DISCUSSION OF CHANGES
CTS 3/4.1.2.1, FLOW PATHS - SHUTDOWN

significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Flow Paths - Shutdown Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Flow Paths - Shutdown LCO and Surveillances may be relocated out of the Technical Specifications. The Flow Paths - Shutdown Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.1, FLOW PATHS - SHUTDOWN

There are no specific NSHC discussions for this Specification.

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CTS 3/4.1.2.2, Flow Paths - Operating

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. R.1

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperatures of the areas containing the flow path components from the boric acid tank to the blending tee are greater than or equal to 63°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on an RWST sequencing signal.

d. At least once per 18 months during shutdown by verifying that the flow path required by specification 3.1.2.2.a delivers at least 34 gpm to the Reactor Coolant System.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. R.1

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperatures of the areas containing the flow path components from the boric acid tank to the blending tee are greater than or equal to 63°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a RWST sequencing signal.

d. At least once per 18 months during shutdown by verifying that the flow path required by specification 3.1.2.2.a delivers at least 34 gpm to the Reactor Coolant System.

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DISCUSSION OF CHANGES
CTS 3/4.1.2.2, FLOW PATHS - OPERATING

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.2 provides requirements on the boration systems flow paths during operation. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.2 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Flow Paths - Operating Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Flow Paths - Operating Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Flow Paths - Operating Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-

DISCUSSION OF CHANGES
CTS 3/4.1.2.2, FLOW PATHS - OPERATING

significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Flow Paths - Operating Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Flow Paths - Operating LCO and Surveillances may be relocated out of the Technical Specifications. The Flow Paths - Operating Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.2, FLOW PATHS - OPERATING

There are no specific NSHC discussions for this Specification.

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CTS 3/4.1.2.3, Charging Pump - Shutdown

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flowpath associated with support of Unit 2 shutdown functions shall be available.*

R.1

See ITS 3.4.12

L.1

APPLICABILITY: Specification 3.1.2.3.a - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

R.1

L.1

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

R.1

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

See ITS 3.4.12

c. The provisions of Specification 3.0.3 are not applicable.

R.1

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return the required flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

L.1

SURVEILLANCE REQUIREMENTS

4.1.2.3.1.1 The above required charging pump shall be demonstrated OPERABLE by verifying that the pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

R.1

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

See ITS 3.4.12

CTS 3/4.1.2.3

~~3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS~~
~~3/4.1 REACTIVITY CONTROL SYSTEMS~~

~~CHARGING PUMP - SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

R.1

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 152°F.

See ITS 3.4.12

~~4.1.2.3.3 Charging line cross-tie valves to Unit 2 will be cycled full travel at least once per 18 months. Following cycling, the valves will be verified to be in their closed positions.~~

L.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flow path associated with support of Unit 1 shutdown functions shall be available.

R.1 L.1 See ITS 3.4.12

APPLICABILITY: Specification 3.1.2.3.a - MODES 5 and 6
Specification 3.1.2.3.b - At all times when Unit 1 is in MODES 1, 2, 3, or 4.

R.1 L.1

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

R.1

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

See ITS 3.4.12

c. The provisions of Specification 3.0.3 are not applicable.

R.1

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 1 and return the required flow path to available status within the next 60 days, or have Unit 1 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

L.1

SURVEILLANCE REQUIREMENTS

4.1.2.3.1.1 The above required charging pump shall be demonstrated OPERABLE by verifying that the pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5

R.1

* A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

See ITS 3.4.12

CTS 3/4.1.2.3

~~3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS~~
~~3/4.1 REACTIVITY CONTROL SYSTEMS~~

~~SURVEILLANCE REQUIREMENTS~~

R.1

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above-required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 152°F.

See ITS 3.4.12

4.1.2.3.3 ~~Charging line cross-tie valves to Unit 1 will be cycled full travel at least once per 18 months. Following cycling, the valves will be verified to be in their closed positions.~~

L.1

DISCUSSION OF CHANGES
CTS 3/4.1.2.3, CHARGING PUMP - SHUTDOWN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.3 provides requirements on the charging pumps during shutdown when used as part of the boration system. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual. It should be Noted that this Specification also has requirements concerning the maximum number of charging and safety injection pumps that can be OPERABLE. This Discussion of Change does not address these requirements; they are covered in ITS 3.4.12. It should also be Noted that this Specification has requirements associated with the safe shutdown requirements of 10 CFR 50 Appendix R. These requirements are discussed in DOC L.1.

This change is acceptable because CTS 3/4.1.2.3 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Charging Pumps - Shutdown Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Charging Pumps - Shutdown Specification does not satisfy criterion 2.

DISCUSSION OF CHANGES
CTS 3/4.1.2.3, CHARGING PUMP - SHUTDOWN

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Charging Pumps - Shutdown Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Charging Pumps - Shutdown Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Charging Pumps - Shutdown LCO and Surveillances may be relocated out of the Technical Specifications. The Charging Pumps - Shutdown Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.1.2.3.b states that one charging flow path associated with support of Unit 2 (Unit 1) and Unit 1 (Unit 2) shutdown functions shall be available. The ITS does not include these requirements. This changes the CTS by deleting these requirements from the CTS.

The purpose of CTS 3.1.2.3.b is to satisfy the safe shutdown requirements of 10 CFR 50 Appendix R. This change is acceptable because the LCO requirements in the Technical Requirements Manual continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. This change deletes the safe shutdown requirements of 10 CFR 50 Appendix R from the CTS. The opposite unit charging flow path requirements are not needed to satisfy the requirements of the unit safety analyses. CNP is still committed to the safe shutdown requirements of 10 CFR 50 Appendix R. In addition to this change, the Applicability and Action associated with CTS 3.1.2.3.b have been deleted, as well as CTS 4.1.2.3.3, which tests the capability of the unit cross tie valves to cycle. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.3, CHARGING PUMP – SHUTDOWN

There are no specific NSHC discussions for this Specification.

CTS 3/4.1.2.4, Charging Pumps - Operating

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that the pumps' developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

R.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that the pumps'developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

R.1

DISCUSSION OF CHANGES
CTS 3/4.1.2.4, CHARGING PUMPS - OPERATING

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.4 provides requirements on the charging pumps during operation when used as part of the boration system. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.4 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Charging Pumps - Operating Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Charging Pumps - Operating Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Charging Pumps - Operating Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-

DISCUSSION OF CHANGES
CTS 3/4.1.2.4, CHARGING PUMPS - OPERATING

significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Charging Pumps - Operating Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Charging Pumps - Operating LCO and Surveillances may be relocated out of the Technical Specifications. The Charging Pumps - Operating Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.4, CHARGING PUMPS - OPERATING

There are no specific NSHC discussions for this Specification.

CTS 3/4.1.2.5, Boric Acid Transfer Pumps - Shutdown

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

R.1

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional surveillance requirements other than those required by Specification 4.0.5.

CTS 3/4.1.2.5

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

R.1

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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DISCUSSION OF CHANGES
CTS 3/4.1.2.5, BORIC ACID TRANSFER PUMPS - SHUTDOWN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.5 provides requirements on the boric acid transfer pumps during shutdown. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.5 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Boric Acid Transfer Pumps - Shutdown Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Boric Acid Transfer Pumps - Shutdown Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Boric Acid Transfer Pumps - Shutdown Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-

DISCUSSION OF CHANGES
CTS 3/4.1.2.5, BORIC ACID TRANSFER PUMPS - SHUTDOWN

significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Boric Acid Transfer Pumps - Shutdown Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Boric Acid Transfer Pumps - Shutdown LCO and Surveillances may be relocated out of the Technical Specifications. The Boric Acid Transfer Pumps - Shutdown Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.5, BORIC ACID TRANSFER PUMPS - SHUTDOWN

There are no specific NSHC discussions for this Specification.

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CTS 3/4.1.2.6, Boric Acid Transfer Pumps - Operating

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

R.1

SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional surveillance requirements other than those required by Specification 4.0.5.

REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

R.1

SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional Surveillance Requirements other than those required by Specification 4.0.5.

DISCUSSION OF CHANGES
CTS 3/4.1.2.6, BORIC ACID TRANSFER PUMPS - OPERATING

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.6 provides requirements on the boric acid transfer pumps during operation. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.6 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Boric Acid Transfer Pumps - Operating Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Boric Acid Transfer Pumps - Operating Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Boric Acid Transfer Pumps - Operating Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-

DISCUSSION OF CHANGES
CTS 3/4.1.2.6, BORIC ACID TRANSFER PUMPS - OPERATING

significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Boric Acid Transfer Pumps - Operating Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Boric Acid Transfer Pumps - Operating LCO and Surveillances may be relocated out of the Technical Specifications. The Boric Acid Transfer Pumps - Operating Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.6, BORIC ACID TRANSFER PUMPS - OPERATING

There are no specific NSHC discussions for this Specification.

CTS 3/4.1.2.7, Borated Water Sources - Shutdown

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum usable borated water volume of 5000 gallons,
2. Between 6,550 and 6,990 ppm of boron, and
3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION: R.1

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the water level volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum usable borated water volume of 5,000 gallons,
2. Between 6,550 and 6,990 ppm of boron, and
3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum usable borated water volume of 90,000 gallons,
2. A minimum boron concentration of 2400 ppm, and
3. A minimum solution temperature of 70°F.

R.1

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes except: 1) heatup or cooldown of the reactor coolant volume provided that SHUTDOWN MARGIN sufficient to accommodate the change in temperature is maintained in accordance with Specification 3.1.1.2 in MODE 5 or Specification 3.9.1 in MODE 6, and the heatup or cooldown rate is restricted to 50°F or less in any one-hour period in MODE 5, or 2) addition of water from the RWST, provided the boron concentration in the RWST is greater than or equal to the minimum required by Specification 3.1.2.7.b.2.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

DISCUSSION OF CHANGES
CTS 3/4.1.2.7, BORATED WATER SOURCES - SHUTDOWN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.7 provides requirements on the borated water sources during shutdown. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.7 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Borated Water Sources - Shutdown Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Borated Water Sources - Shutdown Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Borated Water Sources - Shutdown Specification does not satisfy criterion 3.

4. As discussed in Section 4.0 (Appendix A, page A-10) and summarized in Table 1 of WCAP-11618, the loss of the CVCS System was found to be a

DISCUSSION OF CHANGES
CTS 3/4.1.2.7, BORATED WATER SOURCES - SHUTDOWN

non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Borated Water Sources - Shutdown Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Borated Water Sources - Shutdown LCO and Surveillances may be relocated out of the Technical Specifications. The Borated Water Sources - Shutdown Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.7, BORATED WATER SOURCES - SHUTDOWN

There are no specific NSHC discussions for this Specification.

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CTS 3/4.1.2.8, Borated Water Sources - Operations (Unit 1)/ Operating (Unit 2)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATIONS

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum usable borated water volume of 8,500 gallons,*
2. Between 6,550 and 6,990 ppm of boron, and
3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 375,500 gallons of water,
2. Between 2400 and 2600 ppm of boron, and
3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F. R.1

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

---

* Not required when borated water is injected into the RCS to meet SHUTDOWN MARGIN requirements of MODES 3 and 4.

CTS 3/4.1.2.8

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the water level of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

R.1

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CTS 3/4.1.2.8

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.1 REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 8500 gallons,*

2. Between 6,550 and 6,990 ppm of boron, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 375,500 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F. R.1

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

*Not required when borated water is injected into the RCS to meet SHUTDOWN MARGIN requirements of MODES 3 and 4.

CTS 3/4.1.2.8

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

R.1

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DISCUSSION OF CHANGES
CTS 3/4.1.2.8, BORATED WATER SOURCES - OPERATIONS (UNIT 1)/ OPERATING (UNIT 2)

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3/4.1.2.8 provides requirements on the borated water sources during operation. The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the CTS requires a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3/4.1.2.8 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The Borated Water Sources - Operations/Operating Specification does not satisfy criterion 1.

2. The CVCS is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. The Borated Water Sources - Operations/Operating Specification does not satisfy criterion 2.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient. The Borated Water Sources - Operations/Operating Specification does not satisfy criterion 3.

DISCUSSION OF CHANGES
CTS 3/4.1.2.8, BORATED WATER SOURCES - OPERATIONS (UNIT 1)/
OPERATING (UNIT 2)

4. As discussed in Section 4.0 (Appendix A, page A-10) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment. The Borated Water Sources - Operations/Operating Specification does not satisfy criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Borated Water Sources - Operations/Operating LCO and Surveillances may be relocated out of the Technical Specifications. The Borated Water Sources - Operations/Operating Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.1.2.8, BORATED WATER SOURCES - OPERATIONS (UNIT 1)/ OPERATING (UNIT 2)

There are no specific NSHC discussions for this Specification.

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CTS 3/4.10.1, Shutdown Margin

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 34 gpm of 6,550 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 34 gpm of 6,550 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored. M.1

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated OPERABLE by verifying its rod drop time to be $\leq$ 2.4 seconds within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion for OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 34 gpm of 6,550 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 34 gpm of 6,550 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored. M.1

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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DISCUSSION OF CHANGES
CTS 3/4.10.1, SHUTDOWN MARGIN

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 CTS 3/4.10.1 provides an exception to the SHUTDOWN MARGIN requirements in CTS 3.1.1.1 in MODE 2 for the purpose of measurement of rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s). According to the Bases, this special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. The ITS does not contain this special test exception. This changes the CTS by eliminating a special test exception.

This change is acceptable because this method of testing is no longer used. As a result, the CTS special test exception is not needed. Other rod worth measurement techniques that do not violate the SHUTDOWN MARGIN requirements are used. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.1, SHUTDOWN MARGIN

There are no specific NSHC discussions for this Specification.

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CTS 3/4.10.2, Group Height, Insertion, and Power Distribution Limits

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1 and 3.2.4 are suspended, either: M.1

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either: M.1

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

DISCUSSION OF CHANGES
CTS 3/4.10.2, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 CTS 3/4.10.2 provides an exception to the rod group height, rod insertion, and power distribution limits Specifications. This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth and, 2) determine the reactor stability index and damping factor under xenon oscillation conditions. The ITS does not contain this special test exception. This changes the CTS by eliminating a special test exception.

This change is acceptable because these types of PHYSICS TESTS (measurement of control rod worth and determination of the reactor stability index as well as the damping factor under xenon oscillation conditions) are only performed during initial plant startup test programs. These tests are never performed during post-refueling PHYSICS TESTS. As a result, the CTS special test exception is not needed. This change is designated as more restrictive because an exception to the CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.2, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

There are no specific NSHC discussions for this Specification.

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CTS 3/4.10.3, Pressure/Temperature Limitation - Reactor Criticality (Unit 1)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

CTS 3/4.10.3

M.1

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the region of acceptable operation shown on Figures 3.4-2 and 3.4-3.

APPLICABILITY: MODE 2

ACTION:

a. With the THERMAL POWER greater than 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figures 3.4-2 and 3.4-3, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figures 3.4-2 and 3.4-3 at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour.

CTS 3/4.10.3

M.1

SPECIAL TEST EXCEPTIONS

SURVEILLANCE REQUIREMENTS (Continued)

4.10.3.3 Each Intermediate and Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

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DISCUSSION OF CHANGES
CTS 3/4.10.3, PRESSURE/TEMPERATURE LIMITATION – REACTOR CRITICALITY

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 (Unit 1 only) CTS 3/4.10.3 provides an exception to the minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 during low temperature PHYSICS TESTS provided some other restrictions are enforced. These restrictions are that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, the reactor trip setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoints are set at ≤ 25% of RATED THERMAL POWER, and the Reactor Coolant System temperature and pressure relationship is maintained within the region of acceptable operation shown on Figures 3.4-2 and 3.4-3. The ITS does not contain this special test exception. This changes the Unit 1 CTS by eliminating a special test exception.

This change is acceptable because low temperature PHYSICS TESTS are no longer performed. This allowance is not available for Unit 2 and is not needed for Unit 1. Future PHYSICS TESTS will be performed under ITS 3.1.8, "PHYSICS TESTS Exceptions – MODE 2," which has been developed from CTS 3/4.10.4, PHYSICS TESTS. As a result, the CTS special test exception is not needed. This change is designated as more restrictive because an exception to the Unit 1 CTS is being deleted.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.10.3, PRESSURE/TEMPERATURE LIMITATION – REACTOR CRITICALITY

There are no specific NSHC discussions for this Specification.

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SUMMARY OF CHANGES
ITS SECTION 3.2

Change Description	Affected Pages
A self-identified change for ITS 3.2.1 has been made. This change adds "RTP" after the phrase "Reduce THERMAL POWER ≥ 1%" for ITS 3.2.1 Required Action B.1.	Pages 28 and 33 of 158.
A self-identified change for ITS 3.2.1 and 3.2.4 Bases has been made. This change revises ITS 3.2.1 and 3.2.4 Bases to incorporate miscellaneous editorial changes, and is administrative.	Pages 36 and 149 of 158.

VOLUME 7

CNP UNITS 1 AND 2
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS SECTION 3.2
POWER DISTRIBUTION LIMITS

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.2.1, Heat Flux Hot Channel Factor

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

$F_Q^C(Z)$ shall be within the limits specified in the COLR. LA.1

LCO 3.2.1

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq CFQ/P\ [K(Z)]$ $\quad P > 0.5$

$F_Q(Z) \leq CFQ/0.5\ [K(Z)]$ $\quad P \leq 0.5$

- CFQ is the F_Q limit at RATED THERMAL POWER specified in the COLR
- $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$
- $F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

ACTION A

With $F_Q(Z)$ exceeding its limit:

after each determination of $F_Q(Z)$ A.2

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next $\boxed{4}$ 72 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. L.1

b. $\boxed{\text{~~Identify and~~}}$ correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit. L.2

Add proposed ACTION C M.1

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A.1

ITS

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

~~4.2.2.1 The provisions of Specification 4.0.4 are not applicable.~~ A.3

SR 3.2.1.1 4.2.2.2 $F_Q(Z)$ shall be determined to be within its limit above 5% of RATED THERMAL POWER according to the following schedule:

~~a. Whenever $F_Q(Z)$ is measured for reasons other than meeting the requirement of 4.2.6.2, or~~ Add proposed 1st Frequency, Including Note to SR 3.2.1.1 M.2

b. At least once per 31 effective full power days, ~~whichever occurs first~~.

ITS 3.2.1

ITS

A.1

POWER DISTRIBUTION

LIMITS

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1

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ITS 3.2.1

ITS

A.1

Figure 3.2-3 intentionally deleted.

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ITS 3.2.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.2 POWER DISTRIBUTION LIMITS

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

$F_Q^W(Z)$ — A.4

within the limit specified in the COLR

LCO 3.2.1 3.2.6 ~~ALLOWABLE POWER LEVEL (APL) given by the following relationship, shall be greater than or equal to THERMAL POWER:~~

~~$APL = \min \text{ over } Z \text{ of } \frac{CFQ \times K(Z)}{F_Q(Z) \times V(Z) \times F_P} \times 100\%.$~~

- ~~CFQ is the F_Q limit at RATED THERMAL POWER specified in the COLR.~~
- ~~K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the COLR.~~ — LA.2
- ~~$F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.~~ — A.4
- ~~V(Z) is the function specified in the COLR.~~
- F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in max over Z of $\frac{F_Q(Z)}{K(Z)}$ with exposure

Note to SR 3.2.1.2

Then either of the penalties, F_p, shall be taken:

F_p = burnup dependent penalty specified in the COLR, or,

F_p = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until two successive maps indicate that the max over Z of $\frac{F_Q(Z)}{K(Z)}$ is not increasing.

- ~~The above limit is not applicable in the following core regions.~~
  1) ~~Lower core region 0% to 10% inclusive.~~
  2) ~~Upper core region 90% to 100% inclusive.~~

LA.2

APPLICABILITY: MODE 1

ITS 3.2.1

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.2 POWER DISTRIBUTION LIMITS

A.4

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

4 hours — L.3

ACTION B — With APL less than THERMAL POWER, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within ~~15 minutes~~. Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER within the ~~next 4~~ hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER.

72 — L.1

Add proposed Required Action B.4 — M.3

Add proposed ACTION C — M.1

SURVEILLANCE REQUIREMENTS

~~4.2.6.1 The provisions of Specification 4.0.4 are not applicable.~~ — A.3

$F_Q^W(Z)$ — A.4

SR 3.2.1.2 — 4.2.6.2 APL shall be determined by measurement ~~in conjunction with the target flux difference and target band determination~~ above 15% of RATED THERMAL POWER, according to the following schedule: — A.5

Within 24 hours — M.2

a. *Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which APL was last determined*[**], or

$F_Q^W(Z)$ — A.4

b. At least once per 31 effective full power days, whichever occurs first.

~~[*]APL can be redefined by remeasuring the target axial flux difference.~~ — A.4

SR 3.2.1.2 Note 1 — [**]During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved. — M.2

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ITS 3.2.1

ITS

A.1

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

$F^C_Q(Z)$ shall be within the limits specified in the COLR — LA.1

LCO 3.2.1

3.2.2 ~~$F_Q(Z)$ shall be limited by the following relationships:~~

~~$F_Q(Z) \leq CFQ/P\ [K(Z)]$ $P > 0.5$~~

~~$F_Q(Z) \leq CFQ/0.5\ [K(Z)]$ $P \leq 0.5$~~

- ~~CFQ is the F_Q limit at RATED THERMAL POWER specified in the COLR~~
- ~~$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$~~
- ~~$F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.~~
- ~~$K(Z)$ is the normalized $F_Q(Z)$ as a function of core height specified in the COLR.~~

APPLICABILITY: MODE 1

ACTION:

ACTION A

With $F_Q(Z)$ exceeding its limit: after each determination of $F_Q(Z)$ — A.2

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the ~~Power Range Neutron~~ Flux-High Trip Setpoints within the next $\boxed{4}$ 72 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. — L.1

b. ~~Identify and~~ correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit. — L.2

Add proposed ACTION C — M.1

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ITS

A.1

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 ~~The provisions of Specification 4.0.4 are not applicable.~~ A.3

SR 3.2.1.1 4.2.2.2 $F_Q(Z)$ shall be determined to be within its limit above 5% of RATED THERMAL POWER according to the following schedule:

a. ~~Whenever $F_Q(Z)$ is measured for reasons other than meeting the requirement of 4.2.6.2, or~~

b. At least once per 31 effective full power days, ~~whichever occurs first~~.

Add proposed 1st Frequency including Note to SR 3.2.1.1

M.2

D. C. COOK - UNIT 2 3/4 2-6 AMENDMENT NO. 82

A.1

ITS

POWER DISTRIBUTION LIMITS

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ITS

A.1

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A.1

ITS

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ITS

A.1

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ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.2 POWER DISTRIBUTION LIMITS

ALLOWABLE POWER LEVEL - APL

$F_Q^W(Z)$ — A.4

within the limit specified in the COLR

LIMITING CONDITION FOR OPERATION

LCO 3.2.1 3.2.6 ~~ALLOWABLE POWER LEVEL (APL)~~ ~~given by the following relationship~~ shall be ~~greater than or equal to THERMAL POWER:~~

~~$$APL = \text{min over Z of } \frac{CFQxK(Z)}{F_Q(Z)xV(Z)xF_P} \times 100\%$$~~

- ~~CFQ is the F_Q limit at RATED THERMAL POWER specified in the COLR.~~ — LA.2
- ~~K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the COLR.~~
- ~~$F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.~~ — A.4
- ~~V(Z) is the function specified in the COLR.~~
- ~~F_P = 1.00~~ except when successive steady-state power distribution maps indicate an increase in

Note to SR 3.2.1.2

max over Z of $\frac{F_Q(Z)}{K(Z)}$ with exposure.

Then either of the penalties, F_P, shall be taken:

F_P = burnup dependent penalty specified in the COLR, or

F_P = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until two successive maps indicate that the max over Z of $\frac{F_Q(Z)}{K(Z)}$ is not increasing.

- ~~The above limit is not applicable in the following core regions.~~ — LA.2
  1) ~~Lower core region 0% to 10% inclusive.~~
  2) ~~Upper core region 90% to 100% inclusive.~~

APPLICABILITY: MODE 1

A.1

ITS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.2 POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

4 hours L.3

A.4

ACTION B With APL less than THERMAL POWER, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes* Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER.

72 L.1

Add proposed Required Action B.4 M.3

Add proposed ACTION C M.1

SURVEILLANCE REQUIREMENTS

4.2.6.1 The provisions of Specification 4.0.4 are not applicable. A.3

$F_Q^M(Z)$ A.4

SR 3.2.1.2 4.2.6.2 APL shall be determined by measurement in conjunction with the target flux difference and target band determination* above 15% of RATED THERMAL POWER, according to the following schedule: A.5

within 24 hours M.2

a. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which APL was last determined**, or

$F_Q^M(Z)$ A.4

b. At least once per 31 effective full power days, whichever occurs first.

* APL can be redefined by remeasuring the target axial flux difference. A.4

SR 3.2.1.2 Note 1 ** During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved. M.2

DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 ITS 3.2.1, Required Actions A.2.1, A.2.2, and A.2.3 state that the Required Actions must be taken "after each $F_Q^C(Z)$ determination." CTS 3.2.2, Action a does not explicitly state this requirement.

This change is acceptable because it does not result in a technical change to the Technical Specifications. The CTS is understood to apply after each measurement of $F_Q(Z)$. This change is designated as administrative because it does not result in a technical change to the CTS.

A.3 CTS 4.2.2.1 states "The provisions of Specification 4.0.4 are not applicable." The ITS does not include this statement.

The purpose of a CTS 4.0.4 exception is to allow the plant to enter the MODE of Applicability without performing the required Surveillances. This change is acceptable because the CTS 4.0.4 exception is not necessary. The ITS SR 3.2.1.1 Frequencies are written to allow entry into MODE 1 following a reactor startup. This serves the same purpose as the CTS 4.0.4 exception. This change is designated as administrative because it eliminates a CTS provision which is covered in the ITS in an alternate manner.

A.4 CTS 3/4.2.6 provides a limit, Actions, and Surveillances for the Allowable Power Level (APL). The CTS requires the APL to be greater than THERMAL POWER, and if not, requires the THERMAL POWER to be reduced to APL or less of RATED THERMAL POWER (RTP). It further requires a reduction in the Power Range Neutron Flux - High and Overpower ΔT Trip Setpoints by the same percentage by which APL is below RTP. Surveillance Requirements are provided to periodically confirm APL is within limits. ITS 3.2.1 uses the term $F_Q^W(Z)$, consistent with NUREG-1431, Rev. 2, in lieu of the term APL. The ITS limit for $F_Q^W(Z)$ is provided in the COLR. If the $F_Q^W(Z)$ limit is not met, the ITS Required Actions are to reduce THERMAL POWER by $\geq$ 1% for each 1% that $F_Q^W(Z)$ exceeds its limit, and to reduce the trip setpoints by $\geq$ 1% for each 1% that $F_Q^W(Z)$ exceeds its limit. In addition, the ITS Surveillances periodically confirm $F_Q^W(Z)$ is within limit. This changes the CTS by substituting the term $F_Q^W(Z)$ for the term APL and modifies the Actions accordingly.

The purpose of monitoring and controlling APL is to protect the peaking factors to ensure $F_Q(Z)$ is within limits during transient conditions. The ITS term $F_Q^W(Z)$ performs the same function. As described in the AEP letter to the NRC (letter C0301-05) dated March 7, 2001, APL is analogous to $F_Q^W(Z)$. The letter described the formula for APL and how it related to the ITS term $F_Q^W(Z)$. This

DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

letter was reviewed by the NRC as part of an amendment request relating to APL, and was used as the basis for the NRC to approve Amendments 251 (Unit 1) and 233 (Unit 2) (SER letter dated March 29, 2001). Since the two terms are analogous, using the ITS term $F_Q^W(Z)$ is purely an editorial preference to conform to NUREG-1431, Revision 2. Therefore, this change is acceptable and is designated as an administrative change since conversion to the term $F_Q^W(Z)$ does not result in any technical changes.

A.5 CTS 4.2.6.2 requires APL (changed to $F_Q^W(Z)$ per DOC A.4) to be determined "in conjunction with the target flux difference and target band determination." The ITS does not include this cross-reference to the Surveillances of ITS 3.2.3 (the AFD Specification). This changes the CTS by deleting the cross-reference to the AFD Specification.

The AFD Specification (CTS 3.2.1) does not list the normal, periodic Surveillance Frequencies for determining the target flux difference and target band; it simply references the Frequencies of CTS 3.2.6. As such, the CTS Surveillances of the APL Specification also cross-reference back to the AFD Specification Surveillances. In the ITS, each SR has its own distinct Frequency; they do not normally cross-reference the Frequency in another ITS Specification. Therefore, this change is considered acceptable since it does not result in any technical changes to the CTS. Any changes to the CTS Frequencies are discussed in the appropriate Discussion of Changes. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.2.2 and CTS 3.2.6 do not contain an Action to follow if the provided Actions are not followed. Therefore, CTS 3.0.3 would be entered which would require the plant to be in MODE 2 within 7 hours. ITS 3.2.1 ACTION B states that when the Required Action and associated Completion Time is not met, the plant must be in MODE 2 within 6 hours. This changes the CTS by providing 6 hours instead of 7 hours to be in MODE 2.

This change is acceptable because, based on operating experience, 6 hours is a reasonable time to be in MODE 2 from full power operation in an orderly manner and without challenging plant systems. This change is designated as more restrictive because the ITS allows less time to be in MODE 2 than does the CTS.

M.2 CTS 4.2.2.2 requires $F_Q(Z)$ to be determined to be within its limit whenever $F_Q(Z)$ is measured for reasons other than meeting the requirement of CTS 4.2.6.2 or at least every 31 effective full power days (EFPD), whichever occurs first. ITS SR 3.2.1.1 requires a verification that $F_Q^C(Z)$ is within limit a) once within 24 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified; and b) every 31 EFPD thereafter. However, a Note is provided such that the SR is not required to be performed during power escalation at the beginning of each cycle until 24 hours after equilibrium conditions at a power level for extended operation are achieved. CTS 4.2.6.2 requires the APL to be determined to be within limit upon reaching equilibrium conditions after exceeding 10% or more of RTP, the THERMAL

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DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

POWER at which APL was last determined or at least once per 31 EFPD, whichever occurs first. CTS 4.2.6.2 footnote ** however, allows the Surveillance to be deferred during power escalation at the beginning of each cycle until a power level for extended operation has been achieved. ITS SR 3.2.1.2 requires the $F^W_Q(Z)$ to be verified within the limit: a) once within 24 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^W_Q(Z)$ was last verified; and b) every 31 EFPD thereafter. (It should be noted that the term APL has been changed to $F^W_Q(Z)$ per DOC A.4). The ITS also includes a Note (Note 1) that allows the SR not to be performed during power escalation at the beginning of each cycle until 24 hours after equilibrium conditions at a power level for extended operation are achieved. This changes the CTS by adding a new Frequency (first Frequency) and new time limit (24 hours for the applicable Note) for CTS 4.2.2.2 and adding a new time limit (24 hours) for CTS 4.2.6.2, including footnote **.

The first Frequency for CTS 4.2.2.2 does not really specify a specific time to verify $F_Q(Z)$ is within the limit; it essentially means to verify whenever CNP wants to. Thus, the only actual Frequency specified in CTS 4.2.2.2 is the 31 EFPD Frequency. This change is acceptable because it provides an appropriate verification (with a finite time to complete) at a power level for extended operation and if THERMAL POWER is substantially changed and equilibrium conditions are attained during the 31 EFPD window. This change is designated as more restrictive because it applies a Frequency and time limit which did not exist in the CTS.

M.3 The CTS 3.2.6 Action provides actions for when the APL is less than the THERMAL POWER. However, there are no requirements to recalculate APL prior to increasing power, once the APL is less than THERMAL POWER. ITS 3.2.1 Required Action B.4 requires performance of SR 3.2.1.1 and SR 3.2.1.2 when $F^W_Q(Z)$ is not within limit prior to increasing THERMAL POWER above the limit established in Required Action B.1. (It should be noted that APL has been changed to $F^W_Q(Z)$ per DOC A.4). This changes the CTS by adding a new requirement to verify $F^C_Q(Z)$ and $F^W_Q(Z)$ are within limits prior to increasing THERMAL POWER after restoring $F^W_Q(Z)$ to within the limit.

This change is acceptable because it requires a confirmation that $F^C_Q(Z)$ and $F^W_Q(Z)$ are within limits, similar to the confirmation required by CTS 3.2.2 Action b. This ensures that, prior to increasing THERMAL POWER after restoration of $F^W_Q(Z)$ to within the limit, that $F_Q(Z)$ is still within limits. This change is designated as more restrictive because it applies a new requirement which does not exist in the CTS.

RELOCATED SPECIFICATIONS

None

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DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.2.2 states that $F_Q(Z)$ shall be limited by an equation, which is contained in the LCO. Two of the four parameters in the CTS equation are already located in the CORE OPERATING LIMITS REPORT (COLR). The other two parameters are actually specified in the LCO. ITS LCO 3.2.1 states "$F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR." This changes the CTS by relocating the two parameters that are in the LCO, as well as the equation, to the COLR. This also changes the CTS by using the term " $F_Q^C(Z)$" in lieu of "$F_Q(Z)$".

While the two parameters (P and $F_Q(Z)$) are not cycle specific, the "P" parameter is based on actual THERMAL POWER divided by RTP (i.e., a measured value divided by a constant), and the "$F_Q(Z)$" value is the measured $F_Q(Z)$ multiplied by two constant uncertainty factors. Thus, the entire equation for $F_Q(Z)$ can be considered cycle specific. In addition, the ITS term " $F_Q^C(Z)$" is consistent with the CTS term "$F_Q(Z)$" since both are corrected for the uncertainty factors. The removal of these parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. The ITS requires that $F_Q^C(Z)$ be within the limits specified in the COLR. Two of the four parameters for the $F_Q^C(Z)$ limit are already located in the COLR. Moving the equation itself to the COLR does not change the requirement that the $F_Q^C(Z)$ limit be met. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as transient analysis limits and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.2.6 states that Allowable Power Level (APL) shall be limited by an equation, which is contained in the LCO. Three of the five parameters in the CTS equation are already located in the CORE OPERATING LIMITS REPORT (COLR). The other two parameters are actually specified in the LCO. ITS LCO 3.2.2 states "$F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR." This changes the CTS by relocating the two parameters that are in the LCO, as well as the equation and the allowance that the $F_Q^W(Z)$ limit is not applicable in certain core regions to the COLR. The change from APL to $F_Q^W(Z)$ is described in DOC A.4.

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DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

While the two parameters ($F_Q(Z)$ and F_p) are not normally cycle-specific, the "$F_Q(Z)$" value is the measured $F_Q(Z)$ multiplied by two constant uncertainty factors, and the "F_P" parameter is normally a constant (although, under certain circumstances, the parameter value is specified in the COLR). Thus, the entire equation for $F_Q^W(Z)$ can be considered cycle specific. The removal of these parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle specific parameter limits are being met. The ITS requires that $F_Q^W(Z)$ be within the limits specified in the COLR. Three of the five parameters for the $F_Q^W(Z)$ limit are already located in the COLR. Moving the equation itself to the COLR does not change the requirement that the $F_Q^W(Z)$ limit be met. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as transient analysis limits and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.2 Action a states the Power Range Neutron Flux - High trip setpoints must be reduced 1% for each 1% $F_Q(Z)$ exceeds its limit within 4 hours. The CTS 3.2.6 Action states the Power Range Neutron Flux - High trip setpoints must be reduced by the same percentage which APL is below RTP within 4 hours. (It should be noted that the term APL has been changed to $F_Q^W(Z)$ per DOC A.4). ITS 3.2.1 Required Actions A.2 and B.2 requires the Power Range Neutron Flux - High trip setpoints to be reduced $\geq$ 1% for each 1% $F_Q^C(Z)$ exceeds its limit or for each 1% that $F_Q^W(Z)$ exceeds its limit, respectively, within 72 hours. This changes the CTS by extending the Completion Time from 4 hours to 72 hours.

The purpose of CTS 3.2.2 Action a and the CTS 3.2.6 Action is to reduce the Power Range Neutron Flux - High trip setpoints when $F_Q(Z)$ or APL exceeds its limit to prevent inadvertently exceeding the maximum allowable power level. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. Following a significant power reduction, a time period of at least 24 hours is required to reestablish steady state xenon concentration and power distribution prior to taking a flux map, and a time period of approximately 8 hours to 12 hours is required to take and analyze a flux map. If it is determined

DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

that $F^C_Q(Z)$ or $F^W_Q(Z)$ is still not within its limit, reducing the Power Range Neutron Flux - High trip setpoints takes approximately 2 hours per channel, with additional time required for preparation and channel restoration. Furthermore, setpoint changes should only be required for extended operation in this condition because of the risk of a plant trip during the adjustment. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.2.2 Action b states that when $F_Q(Z)$ exceeds its limit, identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced power limit. ITS 3.2.1 Required Action A.4 requires verification that $F^C_Q(Z)$ is within its limit prior to increasing THERMAL POWER above the reduced power limit. This changes the CTS by eliminating the requirement to identify the cause of the out of limit condition prior to increasing power above the reduced power limit.

The purpose of CTS 3.2.2 Action b is to ensure $F_Q(Z)$ is within its limit prior to increasing THERMAL POWER above the reduced power limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation, while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Identifying the cause of the out of limit condition is not required to restore compliance with the LCO. Identifying the cause of the condition is a function of the corrective action program required by 10 CFR 50, Appendix B. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 3 – Relaxation of Completion Time)* The CTS 3.2.6 Action states that with APL less than THERMAL POWER, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes. ITS 3.2.1 Required Action B.1 requires, under the similar condition (It should be noted that APL has been changed to $F^W_Q(Z)$ per DOC A.4), 4 hours to complete the Required Action. This changes the CTS by extending the Completion Time from 15 minutes to 4 hours.

The purpose of the CTS 3.2.6 Action is to reduce the THERMAL POWER when APL is less than THERMAL POWER to help ensure the peaking factors are not exceeded if a transient were to occur. This change is acceptable because the Completion Time is consistent with safe operation under the specified condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a transient occurring during the allowed Completion Time. During the 4 hour Completion Time, the actual limit being protected by $F^W_Q(Z)$, $F_Q(Z)$, is not necessarily outside the required limits. If it is, then ITS 3.2.1 ACTION A would be entered, which requires a power reduction in 15 minutes. This change is designated as less

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DISCUSSION OF CHANGES
ITS 3.2.1, $F_Q(Z)$

restrictive because additional time is allowed to restore parameters to within LCO limits than was allowed in the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

$F_Q(Z)$ (CAOC-W(Z) Methodology) 1
3.2.1C

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.1C Heat Flux Hot Channel Factor ($F_Q(Z)$) (CAQC-W(Z) Methodology) 1 5

LCO 3.2.2, LCO 3.2.6

LCO 3.2.1C $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR. 1

APPLICABILITY: MODE 1.

ACTIONS

3.2.2 Actions a and b

CONDITION	REQUIRED ACTION	COMPLETION TIME
- NOTE - Required Action A.4 shall be completed whenever this Condition is entered.		
A. $F_Q^C(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	AND	
	A.2 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	AND	
	A.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

2

$F_Q(Z)$ ~~(CAOC-W(Z) Methodology)~~
3.2.1~~C~~ ①

CTS

ACTIONS (continued)

3.2.6 Action

CONDITION	REQUIRED ACTION		COMPLETION TIME
- NOTE - Required Action B.4 shall be completed whenever this Condition is entered.	B.1	Reduce ~~AFD limits~~ THERMAL POWER ≥ 1% RTP for each 1% $F^W_Q(Z)$ exceeds limit.	4 hours
	AND		
B. $F^W_Q(Z)$ not within limit~~s~~.	B.2	Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% ~~that the maximum allowable power of the AFD limits is reduced~~ $F^W_Q(Z)$ exceeds limit.	72 hours
	AND		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% ~~that the maximum allowable power of the AFD limits is reduced~~ $F^W_Q(Z)$ exceeds limit.	72 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

Doc M.1

⑥ ②

$F_Q(Z)$ (CAOC-W(Z) Methodology) (1)
3.2.1C

CTS

SURVEILLANCE REQUIREMENTS

- NOTE -
~~During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.~~ (3)

	SURVEILLANCE	FREQUENCY
4.2.2.2	SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit. INSERT 1	~~Once after each refueling prior to THERMAL POWER exceeding 75% RTP~~ (3) ~~AND~~ (3) Once within ~~12~~ (24) hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified (4) AND 31 EFPD thereafter

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3 **INSERT 1**

-NOTE-
Not required to be performed during power escalation at the beginning of each cycle until 24 hours after equilibrium conditions at a power level for extended operation are achieved.

$F_Q(Z)$ ~~(CAOC-W(Z) Methodology)~~ (1)
3.2.1C

CTS

SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE	FREQUENCY	
	INSERT 2		(3)
4.2.6.2 LCO 3.2.6	SR 3.2.1.2 1. -------- NOTES -------- 2. If measurements indicate that the maximum over z ~~[~~ $F_Q^C(Z) / K(Z)$ ~~]~~ has increased since the previous evaluation of $F_Q^C(Z)$ either: a. Increase $F_Q^W(Z)$ by the greater of a factor of ~~[~~ 1.02 ~~]~~ or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z ~~[~~ $F_Q^C(Z) / K(Z)$ ~~]~~ has not increased. --		(2) (4) (5) (2)
	Verify $F_Q^W(Z)$ is within limit.	~~Once after each refueling prior to THERMAL POWER exceeding 75% RTP~~ ~~AND~~ Once within ~~[12]~~ 24 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter	(3) (4)

3

INSERT 2

-NOTE-

Not required to be performed during power escalation at the beginning of each cycle until 24 hours after equilibrium conditions at a power level for extended operation are achieved.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, $F_Q(Z)$

1. The CAOC-W(Z) methodology and the Specification designator "C" are deleted since they are unnecessary (only one $F_Q(Z)$ Specification is used in the CNP ITS). This information is provided in NUREG-1431, Rev. 2 to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the CAOC-F_{XY} and RAOC-W(Z) methodology Specifications (ISTS 3.2.1A and 3.2.1B) are not used and are not shown.

2. Typographical/grammatical error corrected.

3. The first Frequency of both SR 3.2.1.1 and SR 3.2.1.2 has been deleted. In addition, Notes have been added to the two SRs stating when the SRs are required to be performed following startup at the beginning of a cycle. These two changes are consistent with the current licensing basis of CNP. Currently, CNP only requires the first performance of SR 3.2.1.2 (CTS 4.2.6.2 is the equivalent CTS Surveillance) after achieving equilibrium conditions after exceeding by 10% or more of RTP, the THERMAL POWER at which the parameter was last determined. The CTS modifies this requirement with a footnote (Footnote **) which states that during power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved. This essentially means that the first performance is not required until equilibrium conditions are reached at a power level for extended operation (normally 100% RTP). The ITS Note provides this allowance, and also establishes a finite time (24 hours) after equilibrium conditions are reached to perform the SR. The CTS equivalent to SR 3.2.1.1 (CTS 4.2.2.2) does not require the first performance until 31 EFPD. Thus, the addition of the entirely new ISTS second Frequency (ITS first Frequency), including the SR Note, is more restrictive than required by CTS. However, it is consistent with the Frequency of CTS 4.2.6.2. Finally, due to the addition of these specific SR Notes, the general Note at the beginning of the SRs is not needed and has also been deleted.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. This punctuation correction has been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

6. ISTS 3.2.1C Required Action B.1 requires AFD limits to be reduced $\geq$ 1% RTP for each 1% that $F^W_Q(Z)$ exceeds the limit. ISTS Required Actions B.2 and B.3 require a reduction in Power Range Neutron Flux - High and Overpower ΔT trip setpoints $\geq$ 1% for each 1% that the maximum allowable power of the AFD limits is reduced. The ISTS 3.2.1C Bases for Required Action B.1 state that reducing THERMAL POWER by $\geq$ 1% RTP for each 1% that $F^W_Q(Z)$ exceeds its limit maintains an acceptable power density and protects against the consequences of severe transients with unanalyzed power distributions. ISTS 3.2.1C Required Action B.1 has been modified to require a reduction of THERMAL POWER instead of a reduction of the AFD limits, and ISTS 3.2.1C Required Actions B.2 and B.3 have been modified to require the reduction of the associated trip setpoints to be based upon the amount $F^W_Q(Z)$ exceeds the limit instead of the maximum allowable power that the AFD limits is reduced. These changes establish consistency and clarity between ITS 3.2.1 Required Actions B.1, B.2, and B.3 and the ISTS 3.2.1C Bases.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, $F_Q(Z)$

This is acceptable since reducing THERMAL POWER an amount $\geq F_Q^W(Z)$ exceeds its limit ensures acceptable power distribution exists during a severe transient.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

 (1)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1C Heat Flux Hot Channel Factor ($F_Q(Z)$) (CAOC-W(Z) Methodology) (1)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near equilibrium conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a equilibrium condition, it does not include the variations in the value of $F_Q(Z)$ which are present during non-equilibrium situations such as load following or power ascension. (2)

To account for these possible variations, the equilibrium value of $F_Q(Z)$ is adjusted as $F_Q^W(Z)$ by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

 1

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1); 3

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition; 2 3

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and INSERT 1 2 3

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3). INSERT 2 4

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$F_Q(Z) \le$ (CFQ/P) K(Z) for P > 0.5; and 3

$F_Q(Z) \le$ (CFQ/0.5) K(Z) for $P \le 0.5$, 3

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR; 3

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR; and 3

P = THERMAL POWER/RTP. 3

2 **INSERT 1**

average fuel pellet enthalpy at hot spot is below 200 cal/gm for irradiated and unirradiated fuel

3 **INSERT 2**

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast enough to prevent exceeding acceptable fuel damage limits. SDM should assure subcriticality with the most restrictive RCCA fully withdrawn (Ref. 3).

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 (1)

BASES

LCO (continued)

~~For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of [2.32], and K(Z) is a function that looks like the one provided in Figure B 3.2.1C-1.~~ (5)

For Constant Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z)\ [1.0815]$$ (6)

where [1.0815] is a factor that accounts for fuel manufacturing tolerances (3%) and flux map measurement uncertainty (5%) (Ref. 4) (6) (2) (2)

$F_Q^C(Z)$ is an ~~excellent~~ approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken. (2)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

 1

BASES

APPLICABILITY	The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.
ACTIONS	A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. 7

A.2

A reduction of the Power Range Neutron Flux - High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels and future operation are consistent with safety analyses assumptions.

 (1)

BASES

ACTIONS (continued)

Condition A is modified by a Note that requires Required Action A.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action A.1, even when Condition A is exited prior to performing Required Action A.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER. (8)

B.1

If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, maintains an acceptable absolute power density such that even if a transient occurred, core peaking factors are not exceeded. (6)

B.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of ~~8~~ 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1. (8)

B.3

Reduction in the Overpower ΔT trip setpoints value of K_4 by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action B.1.

B.4

Verification that $F_Q^W(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action B.1 ensures that core conditions

 ①

BASES

ACTIONS (continued)

during operation at higher power levels and future operation are consistent with safety analyses assumptions.

Condition B is modified by a Note that requires Required Action B.4 to be performed whenever the Condition is entered. This ensures that SR 3.2.1.1 and SR 3.2.1.2 will be performed prior to increasing THERMAL POWER above the limit of Required Action B.1, even when Condition A is exited prior to perfoming Required Action B.4. Performance of SR 3.2.1.1 and SR 3.2.1.2 are necessary to assure $F_Q(Z)$ is properly evaluated prior to increasing THERMAL POWER. ⑧

C.1

ANY MODE ⑧ ⑨

If Required Actions ~~A.1 through A.4 or B.1 through B.4 are not met within their~~ and associated Completion Times, the ~~plant~~ unit must be placed in a ~~mode~~ MODE or condition in which the LCO requirements are not applicable. This is done by placing the ~~plant~~ unit in at least MODE 2 within 6 hours. ②

is not met

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging ~~plant~~ unit systems. ② ②

SURVEILLANCE REQUIREMENTS

~~SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ and $F_Q^W(Z)$ are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_Q^C(Z)$ and $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ and $F_Q^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ and $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$.~~ The Frequency condition is not intended to require ⑥

⑫ Move to Page B 3.2.1C-7 as INSERT SR 3.2.1.1 and Page B 3.2.1C-9 as INSERT SR 3.2.1.2

$F_Q(Z)$ (CAOC-W(Z) Methodology) 1
B 3.2.1C

BASES

SURVEILLANCE REQUIREMENTS (continued)

Move to Page B 3.2.1C-7 as INSERT SR 3.2.1.1 and Page B 3.2.1C-9 as INSERT SR 3.2.1.2

verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which $F_Q(Z)$ was last measured. 12

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. ~~Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ [1.0815] (Ref. 4).~~ $F_Q^C(Z)$ is then compared to its specified limits. 7

~~The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.~~ 7

~~Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.~~ 6

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required ~~12~~ 24 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits). 10

INSERT SR 3.2.1.1 from pages B 3.2.1C-6 and 7 12

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

INSERT 3 6

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

6

INSERT 3

SR 3.2.1.1 is modified by a Note, which applies during power escalation after a refueling. The Note states that the Surveillance is not required to be performed until 24 hours after equilibrium conditions at a power level for extended operation are achieved. This Note allows the unit to startup from a refueling outage and reach the power level for extended operation (normally 100% RTP) prior to requiring performance of the SR. Within 24 hours after equilibrium conditions are reached at the power level for extended operation, the SR must be performed.

 (1)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR. (7)

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive and (6) (3)

b. Upper core region, from 85 to 100% inclusive. (6)

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions. (6)

(Note 2)

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^W(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q^C(Z)$ evaluation. (8) (12)

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression INSERT 4 (12)

maximum over z $\left[F_Q^C(Z)/K(Z)\right]$ INSERT 5 (8) (12)

It is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR (Ref. 5). INSERT 6 (10) (12)

- REVIEWER'S NOTE -
WCAP-10216-P-A, Rev. 1A, Relaxation of Constant Axial Offset Control and F_Q Surveillance Technical Specification, February 1994, or other appropriate plant specific methodology, is to be listed in the COLR description in the Administrative Controls Section 5.0 to address the methodology used to derive this factor. (11)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased. (6)

12 **INSERT 4**

measurements indicate that the

12 **INSERT 5**

has increased since the previous evaluation of $F^C_Q(Z)$,

12 **INSERT 6**

and reverify $F^W_Q(Z)$ is within limits; or SR 3.2.1.2 must be repeated once per 7 EFPD until either $F^W_Q(Z)$ is within the limits or two successive flux maps indicate that the maximum over z $(F^C_Q(Z)/K(Z))$ has not increased.

$F_Q(Z)$ (CAOC-W(Z) Methodology) (1)
B 3.2.1C

BASES

SURVEILLANCE REQUIREMENTS (continued)

INSERT SR 3.2.1.2 from Pages B 3.2.1C-6 and 7

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels. (10) 24 (12)

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_{XY}(Z)$ evaluations. (8)

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances. (8) (6)

INSERT 7

REFERENCES

1. 10 CFR 50.46, 1974. (9)
2. Regulatory Guide 1.77, Rev. 0, May 1974. INSERT 8 (2)
3. 10 CFR 50, Appendix A, GDC 26. INSERT 9 (4)
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988. (8)
5. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.

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6 **INSERT 7**

SR 3.2.1.2 is modified by Note 1, which applies during power escalation after a refueling. The Note states that the Surveillance is not required to be performed until 24 hours after equilibrium conditions at a power level for extended operation are achieved. This Note allows the unit to startup from a refueling outage and reach the power level for extended operation (normally 100% RTP) prior to requiring performance of the SR. Within 24 hours after equilibrium conditions are reached at the power level for extended operation, the SR must be performed.

2 **INSERT 8**

UFSAR, Section 14.2.6.7 (Unit 1) and Section 14.2.6.1.2 (Unit 2).

4 **INSERT 9**

UFSAR, Section 1.4.5.

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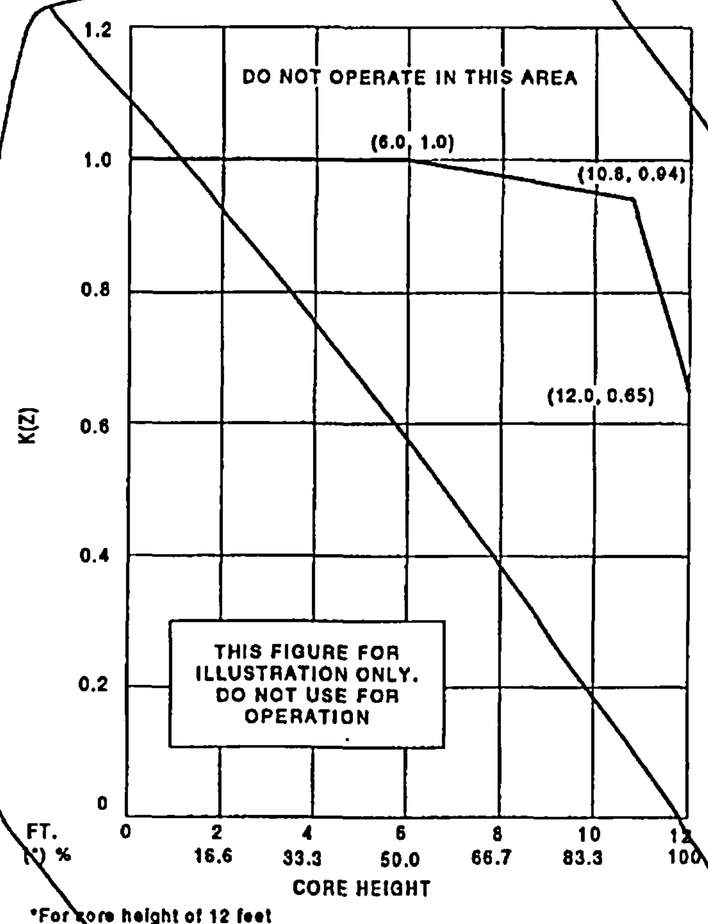
$F_Q(Z)$ (CAOC-W(Z) Methodology) B 3.2.1C

1



5

Figure B 3.2.1C-1 (page 1 of 1)
K(Z) - Normalized FQ(Z) as a Function of Core Height

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1 BASES, $F_Q(Z)$

1. The CAOC-W(Z) methodology and the Specification designator "C" are deleted since they are unnecessary (only one $F_Q(Z)$ Specification is used in the CNP ITS). This information is provided in NUREG-1431, Rev. 2 to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the CAOC-F_{XY} and RAOC-W(Z) methodology Specification Bases (ISTS B 3.2.1A and B 3.2.1B) are not used and are not shown.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

4. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

5. This generic statement is not necessary. The LCO Section of the Bases already states that certain values are specified in the COLR, and providing a "normal" value or Figure that is not the actual one in the COLR can lead to confusion that results in improper limits being applied. Therefore, the statement and the Figure B 3.2.1C-1 have been deleted.

6. The Bases have been changed to reflect changes made to the Specification.

7. These redundant statements in the ACTIONS A.1 Bases, the SR 3.2.1.1 Bases, and the SR 3.2.1.2 Bases have been deleted since the term is already defined and adequately described in the LCO Section of the Bases and does not need to be repeated in these Sections.

8. Typographical/grammatical error corrected.

9. These changes have been made to be consistent with similar phrases in other parts of the ITS Bases.

10. The brackets have been removed and the proper plant specific information/value has been provided.

11. This Reviewer's Note has been deleted. The appropriate methodology is listed in the COLR section of the ITS.

12. Changes have been made to be consistent with the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.1, $F_Q(Z)$

There are no specific NSHC discussions for this Specification.

ATTACHMENT 2

ITS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.2.2

ITS

A.1

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 $F^N_{\Delta H}$ shall be ~~limited by the following relationship:~~ within the limits specified in the COLR (LA.1)

~~$F^N_{\Delta H} \leq$ CFDH [1 + PFDH (1-P)]~~

~~where: P is the fraction of RATED THERMAL POWER~~

~~CFDH is the $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR~~

~~PFDH is the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR~~

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit: Add proposed Condition A Note (M.1)

Required Actions A.1 and A.3 — a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within ~~2~~ 4 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within ~~the next 4~~ 72 hours, (L.1)

Required Action A.2 — b. Demonstrate ~~through in-core mapping~~ that $F^N_{\Delta H}$ is within its limit within 24 hours ~~after exceeding the limit or~~ (LA.2)

ACTION B — reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~2~~ 6 hours, and (L.2)

Required Action A.4 — c. ~~Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER;~~ (A.2) subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated ~~through in-core mapping~~ (LA.2) to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER. Add proposed Required Action A.4 Note (A.3)

ITS 3.2.2

A.1

ITS

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 4.2.3 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map: LA.2

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days thereafter. A.4

c. ~~The provisions of Specification 4.0.4 are not applicable.~~

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ITS 3.2.2

ITS

A.1

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

LCO 3.2.2 3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$F^N_{\Delta H} \le CFDH\ [1 + PFDH\ (1-P)]$

where: P is the fraction of RATED THERMAL POWER

CFDH is the $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR

PFDH is the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR

within the limits specified in the COLR — LA.1

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit: — Add proposed Condition A Note — M.1

Required Actions A.1 and A.3 a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within ~~2~~ 4 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within ~~the next 4~~ 72 hours. — L.1

LA.2

Required Action A.2 b. Demonstrate ~~through in-core mapping~~ that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or

ACTION B — reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next ~~2~~ 6 hours, and — L.2

Required Action A.4 c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated ~~through in-core mapping~~ to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER. — A.2 — LA.2

Add proposed Required Action A.4 Note — A.3

ITS 3.2.2

A.1

ITS

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 4.2.3 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map: LA.2

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days. thereafter A.4

c. ~~The provisions of Specification 4.0.4 are not applicable.~~

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ITS

A.1

POWER DISTRIBUTION LIMITS

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A.1

ITS

POWER DISTRIBUTION LIMITS

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DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.2.3 Action c states that with $F^N_{\Delta H}$ exceeding its limit "identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER." ITS 3.2.2 does not include this requirement. This changes the CTS by eliminating the statement that the cause of the out-of-limit condition must be identified and corrected prior to increasing power.

This change is acceptable because the requirements have not changed. Stating that the cause of the $F^N_{\Delta H}$ limit violation must be identified and corrected prior to increasing power (i.e., exiting the Action which required power reduction) is unnecessary. Restoration of compliance with the LCO is always an option and allows exiting the ACTION per ITS 3.0.2. Therefore, it does not have to be stated. In addition, CTS 3.2.3 Action c and ITS 3.2.2 Required Action A.4 require $F^N_{\Delta H}$ to be within limit prior to exceeding 50% RTP and 75% RTP, which ensures $F^N_{\Delta H}$ limit is identified and corrected. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.2.3 Action c states that with $F^N_{\Delta H}$ exceeding its limit, $F^N_{\Delta H}$ must be demonstrated to be within its limit prior to exceeding 50% RTP and 75% RTP, and within 24 hours of exceeding 95% RTP. ITS 3.2.2 Required Action A.4 contains the same requirements. However, ITS 3.2.2 Required Action A.4 is modified by a Note which states "THERMAL POWER does not have to be reduced to comply with this Required Action." This modifies the CTS by adding a Note stating that THERMAL POWER does not have to be reduced to comply with the Required Action.

This change is acceptable because the requirements have not changed. The Note is included in the ITS to make clear that THERMAL POWER does not have to be reduced to perform the Required Action. For example, if $F^N_{\Delta H}$ exceeded its limit and power was reduced to 60% RTP before $F^N_{\Delta H}$ is demonstrated to be within its limit, under the Note THERMAL POWER does not have to be reduced to less than 50% RTP for a $F^N_{\Delta H}$ measurement. However, $F^N_{\Delta H}$ must still be measured prior to exceeding 75% RTP and within 24 hours of exceeding 95% RTP. The Note is needed because the ITS contains a Note in ITS 3.2.2 Condition A that states "Required Actions A.2 and A.4 must be completed whenever Condition A is entered." The Condition A Note does not exist in the CTS and could be construed as requiring THERMAL POWER to be reduced to comply with Required Action A.4. The Condition A Note is described in DOC M.1. As a result, the Required Action A.4 Note makes the ITS and CTS actions consistent.

DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 4.2.3.c states "The provisions of Specification 4.0.4 are not applicable." The ITS does not include this statement. In addition, CTS 4.2.3.b requires the $F^N_{\Delta H}$ to be determined at least once per 31 Effective Full Power Days. The ITS SR 3.2.2.1 Frequency is 31 EFPD thereafter. This changes the CTS by adding the word "thereafter" to the Frequency.

The purpose of a CTS 4.0.4 exception is to allow the plant to enter the MODE of Applicability without performing the required Surveillances. This change is acceptable because the CTS 4.0.4 exception is not required in the ITS. CTS 4.2.3 is required to be performed prior to operation above 75% RTP after each fuel loading and once per 31 EFPD. Without the CTS 4.0.4 exception, MODE 1 could not be entered without a measurement if the "once per 31 EFPD" Frequency was not met, because Surveillances must be met prior to entering the MODE of Applicability. However, the likelihood of this occurring (needing to enter MODE 1 with the 31 EFPD Frequency not met) is very small; the 31 EFPD Frequency only runs when the reactor is critical and a 25% grace period is allowed by CTS 4.0.2 (ITS SR 3.0.2). Also, the addition of the word "thereafter" in ITS SR 3.2.2.1 ensures that the 31 EFPD Frequency starts only after the first performance of the SR, which is required prior to exceeding 75% RTP after each fuel loading. This is essentially the way the CTS Frequencies work. Therefore, the deletion of the CTS 4.0.4 exception and addition of the word "thereafter" are considered acceptable. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.2.3 Action c states that with $F^N_{\Delta H}$ exceeding its limit "subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated through incore mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, and within 24 hours after attaining 95% or greater RATED THERMAL POWER." However, under CTS 3.0.2, these measurements do not have to be completed if compliance with the LCO is reestablished. ITS 3.2.2 Condition A contains a Note which states, "Required Actions A.2 and A.4 must be completed whenever Condition A is entered." ITS Required Actions A.2 and A.4 require performance of a $F^N_{\Delta H}$ measurement every 24 hours and prior to exceeding 50% RTP and 75% RTP, and within 24 hours after THERMAL POWER is $\geq$ 95% RTP. This changes the CTS by requiring the $F^N_{\Delta H}$ measurements to be made even if $F^N_{\Delta H}$ is restored to within its limit.

This change is acceptable because it establishes appropriate compensatory measurements for violation of the $F^N_{\Delta H}$ limit. As power is reduced under ITS Required Action A.1, the margin to the $F^N_{\Delta H}$ limit increases. Therefore, compliance with the LCO could be reestablished during the power reduction. Verifying that the limit is met as power is increased ensures that the limit

DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

continues to be met and does not remain unmeasured for up to 31 EFPD. This change is designated as more restrictive because it imposes requirements in addition to those in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.2.3 states that $F^N_{\Delta H}$ shall be limited by an equation, which is contained in the LCO. Two of the three parameters in the CTS equation are as specified in the CORE OPERATING LIMITS REPORT (COLR). ITS LCO 3.2.2 states "$F^N_{\Delta H}$ shall be within the limits specified in the COLR." This changes the CTS by relocating the entire equation to the COLR.

The removal of these cycle specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. The ITS requires that $F^N_{\Delta H}$ be within the limits specified in the COLR. Two of the three parameters for the $F^N_{\Delta H}$ limit are already located in the COLR. Moving the equation itself to the COLR does not change the requirement that the $F^N_{\Delta H}$ limit be met. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such transient analysis limits and accident analysis limits) of the safety analyses are met. This change is designated as a less restrictive removal of detail change because information relating to cycle specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.2.3 Actions b and c require $F^N_{\Delta H}$ to be determined to be within its limit through in-core mapping and CTS 4.2.3 requires $F^N_{\Delta H}$ to be determined to be within its limit by using the movable incore detectors to obtain a power distribution map. ITS SR 3.2.2.1 just requires verification that $F^N_{\Delta H}$ is within its limit. This changes the CTS by relocating to the ITS Bases the manner in which the $F^N_{\Delta H}$ determination is performed.

DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

The removal of these details for performing actions and a Surveillance Requirement from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to determine the $F^N_{\Delta H}$ is within its limit. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.3 Action a states that when $F^N_{\Delta H}$ exceeds its limit, reduce THERMAL POWER to less than 50% RTP within 2 hours and reduce the Power Range Neutron Flux - High Trip Setpoints to less than or equal to 55% of RTP within the next 4 hours. ITS 3.2.2 Required Actions A.1 and A.3 state than with $F^N_{\Delta H}$ not within this limit, reduce THERMAL POWER to < 50% RTP within 4 hours and reduce the Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP within 72 hours. This changes the CTS by allowing a 4 hour Completion Time to reduce power to < 50% RTP and 72 hours to reduce the trip setpoints.

The purpose of CTS 3.2.3 Action a is to reduce power, which increases the margin to the $F^N_{\Delta H}$ limit, and to lower the trip setpoints, which avoids inappropriately increasing power and violating the $F^N_{\Delta H}$ limit. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The revised Completion Times allow reactor power to be reduced in a controlled manner without challenging operators, technicians, or plant systems. Following a significant power reduction, a time period of 24 hours is allowed to reestablish steady state xenon concentration and power distribution and to take and analyze a flux map. If it is determined that $F^N_{\Delta H}$ is still not within its limit, reducing the Power Range Neutron Flux - High Trip Setpoints takes approximately 2 hours per channel, with additional time required for preparation and channel restoration. Furthermore, setpoint changes should only be required for extended operation in this condition because of the risk of a plant trip during the adjustment. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.2 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.3 Action b states that when $F^N_{\Delta H}$ exceeds its limit, demonstrate through incore mapping that $F^N_{\Delta H}$ is within its limit within 24 hours or reduce THERMAL POWER to less than 5% within the next 2 hours. ITS 3.2.2 ACTION B states that with the Required Action and

DISCUSSION OF CHANGES
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

associated Completion Time not met, be in MODE 2 within 6 hours. This changes the CTS by allowing a 6 hour Completion Time to reduce power to < 5% RTP in instead of the current 2 hour time limit.

The purpose of CTS 3.2.3, Action b is to reduce power when compliance with the $F^N_{\Delta H}$ limits cannot be obtained to a MODE in which the LCO is not applicable. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The revised Completion Times allow reactor power to be reduced in a controlled manner without challenging operators, technicians, or plant systems. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

$F^N_{\Delta H}$
3.2.2

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.3

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
	A. ----- - NOTE - Required Actions A.2 and A.3 must be completed whenever Condition A is entered. -----	A.1.1	~~Restore $F^N_{\Delta H}$ to within limit.~~	~~4 hours~~
		~~OR~~		
Action a		A.1.2.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	$F^N_{\Delta H}$ not within limit.	AND		
Action a		A.1.2.2	Reduce Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP.	72 hours
		AND		
Action b		A.2	Perform SR 3.2.2.1.	24 hours
		AND		

(4) (3) (1)

$F^N_{\Delta H}$
3.2.2

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
Action C		A.2 4	- NOTE - THERMAL POWER does not have to be reduced to comply with this Required Action. Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP	1
Action b	B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours	

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.2.3 SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

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JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

1. ISTS 3.2.2 Required Action A.1.1 requires restoration of $F^N_{\Delta H}$ to within limit within 4 hours or performance of a number of other actions, such as a power reduction to < 50% RTP. The Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 4.1.6.g, states "A Required Action which requires restoration, such that the Condition is no longer met, is considered superfluous. It is only included if it would be the only Required Action for the Condition or it is needed for presentation clarity." Neither exception applies in this case. If fact, the inclusion of Required Action A.1.1 requires an additional level of indenting and numbering for the remaining Required Actions in Condition A, which reduces its clarity. Therefore, Required Action A.1.1 is deleted and the subsequent Required Actions renumbered.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio ~~of [1.3] using the [W-3] CHF correlation~~. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. INSERT 1 (1)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible

1 **INSERT 1**

to a value greater than the design limits

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BASES

BACKGROUND (continued)

cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition; 2

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F; 3

c. During an ejected rod accident, the ~~energy deposition to the fuel must not exceed 280 cal/gm~~ (Ref. 1) and INSERT 2 5 2 INSERT 3 4

d. ~~Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.~~

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to ~~the~~ a ~~95/95 DNB criterion of [1.3] using the [W-3] CHF correlation. This~~ value that provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB. 1

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3). 5

3 **INSERT 2**

average fuel pellet enthalpy at hot spot is below 200 cal/gm for irradiated and unirradiated fuel

4 **INSERT 3**

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast enough to prevent exceeding acceptable fuel damage limits. SDM should assure subcriticality with the most restrictive rod cluster control assembly fully withdrawn (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER. 6

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

BASES

ACTIONS

~~A.1.1~~ A.1 and A.2

(7)

~~With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.~~

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, ~~if power is not reduced because this Required Action is completed within the 4 hour time period~~, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1. (4) (7)

INSERT 4

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours. (4) (7)

~~A.1.2.1 and A.1.2.2~~ (7)

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 ~~and reduce the Power Range Neutron Flux - High to ≤ 55% RTP in accordance with Required Action A.1.2.2~~. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 ~~is consistent with those allowed for in Required Action A.1.1 and~~ provides an acceptable time to reach the required power level from full power operation without allowing the ~~plant~~ unit to remain in an unacceptable condition for an extended period of time. ~~The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.~~ (7) (7) (7) (7) (3) (7)

Move to page B 3.2.2-5 as Insert A.3

INSERT A.2 from page B 3.2.2-5

 (7)

7 **INSERT 4**

even if $F^N_{\Delta H}$ is restored to within limits

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BASES

ACTIONS (continued)

3

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System. 7

Trip

Insert 4A

A.2 — 3 3

INSERT A.3 from page B 3.2.2-4

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$. 7

Required 8

7

Move to page B3.2.2-4 as Insert A.2

A.3 — 4

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

any — and associated — 9

is not met

When ~~Required Actions A.1.1 through A.3 cannot be completed within their required~~ Completion Time ~~s~~, the ~~plant~~ must be placed in a ~~mode~~ in which the LCO requirements are not applicable. This is done by placing the ~~plant~~ in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging ~~plant~~ systems.

unit — 3

8

3 — unit

7 **INSERT 4A**

If $F^N_{\Delta H}$ continues to be not within limits, the Power Range Neutron Flux - High trip setpoints must be reduced to $\leq$ 55% RTP per Required Action A.3.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit. INSERT 5 10

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. [0], May 1974. INSERT 6 3
2. 10 CFR 50, Appendix A, GDC 26. INSERT 7 4
3. 10 CFR 50.46.

10 **INSERT 5**

limit contains an allowance of

3 **INSERT 6**

UFSAR, Section 14.2.6.7 (Unit 1) and Section 14.2.6.1.2 (Unit 2).

4 **INSERT 7**

UFSAR, Section 1.4.5.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2 BASES, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

1. The Bases are revised to reflect the CNP DNB limits and correlation. The CNP safety analyses utilize different DNB limits for various analyses, so a specific value is not provided in the Bases. Also, the correlation used is subject to change and it is an analytical detail that does not add to the understanding of the $F^N_{\Delta H}$ Specification. Therefore, this information is not specified in the Bases.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

4. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. ISTS LCO 3.2.2 Bases state "The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER." This sentence is removed. The first sentence of the LCO Bases states "$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR." Part of the relationship specified in the COLR describes how the $F^N_{\Delta H}$ limit changes as a function of power. Describing part of the $F^N_{\Delta H}$ limit relationship in the Bases is inconsistent and does not provide any value without the rest of the relationship contained in the COLR. Therefore, the sentence is removed.

7. The Bases have been changed to reflect changes made to the Specification.

8. Typographical/grammatical error corrected.

9. These changes have been made to be consistent with similar phrases in other parts of the ITS Bases.

10. The Bases are revised to reflect the CNP $F^N_{\Delta H}$ limit. The Bases state that the measured value of $F^N_{\Delta H}$ must be increased by 1.04 to account for measurement uncertainty. At CNP, the $F^N_{\Delta H}$ limit includes 1.04 adjustment for measurement uncertainty. Therefore, adjusting the measured value is not necessary.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.2, NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 3

ITS 3.2.3, Axial Flux Difference

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.2.3

ITS

A.1

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

LCO 3.2.3.a 3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about a target flux difference. The target band is specified in the COLR. A.2

Add proposed LCO 3.2.3.c

APPLICABILITY: MODE 1 above ~~50%~~ RATED THERMAL POWER A.3

15% A.2

ACTION:

ACTION A

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. Above 90% or ~~0.9 x APL (whichever is less) of RATED THERMAL POWER~~, within 15 minutes: LA.1 L.1

a) Either restore the indicated AFD to within the target band limits, or

ACTION B

b) Reduce THERMAL POWER to ~~less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER~~. LA.1

LCO 3.2.3.b

ACTION C

2. Between 50% and ~~90% or 0.9 x APL (whichever is less)~~ of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes ~~and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours~~.

Add proposed Condition C Note M.1

L.2

LCO 3.2.3 Note 4

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

~~* See Special Test Exception 3.10.2~~ A.3

ITS 3.2.3

A.1

ITS

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

A.4

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status if the AFD has been outside of the target band for any period of time in the previous 24 hours of operation.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

L.3

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ITS 3.2.3

A.1

ITS

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

LCO 3.2.3 Note 1 — 4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band.

LCO 3.2.3 Notes 2 and 3 — Penalty deviation outside of the target band shall be accumulated on a time basis of:

LCO 3.2.3 Note 2 — a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

LCO 3.2.3 Note 3 — b. A penalty deviation of one-half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

SR 3.2.3.3 — 4.2.1.3 The target axial flux difference of each OPERABLE excore channel shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. ~~The provisions of Specification 4.0.4 are not applicable.~~ L.4

SR 3.2.3.2 — 4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. ~~The allowable values of the target band are specified in the COLR.~~ A.5 ~~The provisions of Specification 4.0.4 are not applicable.~~ L.4

ITS

A.1

Figure 3.2-1 intentionally deleted.

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ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.2 POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With APL less than THERMAL POWER, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes. Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER.

See ITS 3.2.1

SURVEILLANCE REQUIREMENTS

4.2.6.1 The provisions of Specification 4.0.4 are not applicable.

4.2.6.2 APL shall be determined by measurement in conjunction with the target flux difference and target band determination* above 15% of RATED THERMAL POWER, according to the following schedule:

SR 3.2.3.2, SR 3.2.3.3

a. Upon ~~achieving equilibrium conditions after exceeding by 10% or more~~ of RATED THERMAL POWER, the THERMAL POWER ~~at which APL was last determined~~**, or

once within 31 EFPD after each refueling

b. At least once per 31 effective full power days. ~~whichever occurs first~~

thereafter

L.4

*APL can be redefined by remeasuring the target axial flux difference.

See ITS 3.2.1

SR 3.2.3.3 Note

**During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

ITS 3.2.3

ITS

A.1

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

LCO 3.2.3.a

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about a target flux difference. The target band is specified in the COLR.

Add proposed LCO 3.2.3.c — A.2

APPLICABILITY: MODE 1 above ~~50%~~ 15 RATED THERMAL POWER* — A.3, A.2

ACTION:

ACTION A

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. ~~Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER~~, within 15 minutes: — LA.1, L.1

a) Either restore the indicated AFD to within the target band limits, or

ACTION B

b) Reduce THERMAL POWER to ~~less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER~~. — LA.1

LCO 3.2.3.b

ACTION C

2. Between 50% and ~~90% or 0.9 x APL (whichever is less)~~ of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes ~~and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~ — L.2

Add proposed Condition C Note — M.1

LCO 3.2.3 Note 4

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

~~* See Special Test Exception 3.10.2~~ — A.3

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ITS 3.2.3

A.1

ITS

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

A.4

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status if the AFD has been outside of the target band for any period of time in the previous 24 hours of operation.

L.3

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

ITS 3.2.3

A.1

ITS

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

LCO 3.2.3 Note 1 — LCO 3.2.3 Notes 2 and 3

4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

LCO 3.2.3 Note 2

a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

LCO 3.2.3 Note 3

b. A penalty deviation of one half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

SR 3.2.3.3

4.2.1.3 The target axial flux difference for the OPERABLE excore channels shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. ~~The provisions of Specification 4.0.4 are not applicable.~~ L.4

SR 3.2.3.2

4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. ~~The allowable values of the target band are specified in the COLR.~~ A.5 ~~The provisions of Specification 4.0.4 are not applicable.~~ L.4

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A.1

ITS

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ITS 3.2.3

ITS

A.1

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS
3/4.2 POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With APL less than THERMAL POWER, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes. Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER. See ITS 3.2.1

SURVEILLANCE REQUIREMENTS

4.2.6.1 The provisions of Specification 4.0.4 are not applicable.

4.2.6.2 APL shall be determined by measurement in conjunction with the target flux difference and target band determination* above 15% of RATED THERMAL POWER, according to the following schedule:

SR 3.2.3.2, SR 3.2.3.3 a. ~~Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which APL was last determined**, or~~

b. At least once per 31 effective full power days thereafter, ~~whichever occurs first.~~ Once within 31 EFPD after each refueling

L.4

* ~~APL can be redefined by remeasuring the target axial flux difference.~~ See ITS 3.2.1

SR 3.2.3.3 Note ** During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

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DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 The CTS 3.2.1 Applicability is MODE 1 above 50% RATED THERMAL POWER. However, CTS 4.2.1.2.b provides a penalty deviation for operation outside of the target band at THERMAL POWER levels between 15% RTP and 50% RTP. The ITS 3.2.3 Applicability is MODE 1 with THERMAL POWER > 15% RTP, and ITS LCO 3.2.3.c states that the AFD may deviate outside the target band with THERMAL POWER < 50% RTP. This changes the CTS by clearly stating that the AFD limit is Applicable between 15% RTP and 50% RTP, but that there is no maximum time limit it can be outside the limit; only the time has to be tracked (so that it can be used for the LCO 3.2.3.b limit).

The purpose of tracking the time limit as required by CTS 4.2.1.2.b is to ensure that power is not increased above 50% RTP if the AFD has been outside its target band longer than allowed in the last 24 hours. The ITS continues to track this time, but properly displays the requirements in the format consistent with other ITS requirements. These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.3 The Applicability of CTS 3.2.1 is modified by a footnote * stating "See Special Test Exception 3.10.2." ITS 3.2.3 Applicability does not contain the footnote or a reference to the Special Test Exception.

The purpose of the CTS 3.2.1 footnote * reference is to alert the user that a Special Test Exception exists which may modify the Applicability of the Technical Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as an administrative change because it does not result in technical changes to the CTS.

A.4 CTS 3.2.1 Action b states "THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a)1), above has been satisfied." CTS 3.2.1 Action c states "THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours." ITS 3.2.3 does not contain similar requirements. This changes the CTS by eliminating prohibitions contained in the CTS.

This change is acceptable because the requirements have not changed. CTS 3.0.4 and ITS LCO 3.0.4 prohibit entering the MODE of Applicability of a

DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

Technical Specification unless the requirements of the LCO are met. CTS 3.2.1 and ITS 3.2.3 are applicable in MODE 1 with THERMAL POWER > 50% RTP (CTS) and > 15% RTP (ITS). Therefore, both the CTS and ITS prohibit exceeding 50% RTP without the LCO requirements being met. CTS 3.2.1 Actions b and c are duplicative of CTS 3.0.4 and ITS LCO 3.0.4, and their elimination does not make a technical change to the Specification. This change is designated as an administrative change because it does not result in a technical change to the CTS.

A.5 CTS 4.2.1.4 states that the allowable values of the target band are specified in the COLR. The ITS does not include this statement in ITS SR 3.2.3.2. This change deletes the statement from the CTS Surveillance concerning where the target band limits are located.

The CTS 4.2.1.4 statement identifies the location of the target band limit. However, this statement is duplicative of ITS LCO 3.2.1, which already identifies the location of the target band limit (the COLR). Therefore, the deletion of the duplicative and redundant statement from the Surveillance Requirement is acceptable, since it remains in the LCO statement (CTS 3.2.1 and ITS LCO 3.2.3). This change is designated as an administrative change since it does not result in any technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.2.1 Action a.2.a)2) requires THERMAL POWER to be reduced to < 50% RTP within 30 minutes if the AFD limits are not met when between 50% RTP and 90% RTP or 0.9 of APL (whichever is less). However, if the AFD limits are met during the 30 minute time limit, the CTS does not require continuation of the power reduction (as allowed by CTS 3.0.2). ITS 3.2.3 ACTION C (as stated in the Note to Condition C) requires completion of the power reduction to < 50% RTP, even if the AFD is restored to within limits prior to the expiration of the 30 minute time limit. The CTS is changed by now requiring power to be reduced to < 50% RTP when the Action is entered, even if the AFD is restored to within limits prior to expiration of the 30 minute time limit.

The purpose of the CTS Action is to restore compliance with the LCO. Since unanalyzed xenon axial distributions could result from a different pattern of xenon buildup and decay when ≥ 50% RTP, power must be reduced to below 50% RTP. Completion of the power reduction is required to ensure that the reactor is at a THERMAL POWER level at which AFD is not a significant accident analysis parameter. This change is designated as more restrictive because it imposes a requirement in addition to those in the CTS.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.2.1 Actions a.1, a.1.b), and a.2 specify Actions to be taken based upon 90% or 0.9 of Allowable Power Level (APL) (whichever is less) RTP. In ITS LCO 3.2.3.b and ACTIONS A, B, and C, the power level point is defined as "upper limit specified in the COLR." This changes the CTS by relocating the specific power level (with the CTS term APL changed to the appropriate term as described in ITS 3.2.1 DOC A.4) to the COLR.

The purpose of the APL value in this Specification is to ensure that if APL is more limiting than THERMAL POWER, then the THERMAL POWER will be reduced to the APL limit. CTS 3.2.1 specifies that the AFD limits are provided in the COLR. The 90% RTP limit is the normal upper limit of the AFD curve, as is shown in the typical AFD curve provided in NUREG-1431, Rev. 2, Bases Figure B 3.2.3A-1. Thus, the 90% RTP limit is already provided in the COLR. In addition, if the $F_Q^W(Z)$ limit is not met, the CNP AFD curve in the COLR already reflects the adjustment (since the y-axis is based on % RTP or 0.9 of APL RTP, whichever is less). The removal of this power level point from the Technical Specifications and the relocation into the COLR is acceptable because this power level point is developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements that ensure the proper power level is utilized in the determination of penalty deviation time and the proper power level to which THERMAL POWER must be reduced. In addition, the actual AFD curve from which the power level is determined is already located in the COLR. Moving the power level point (90% or 0.9 of APL (whichever is less)) RTP does not change the requirements. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, "Core Operating Limits Report." ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as transient analysis limits and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 - Relaxation of Completion Time)* CTS 3.2.1 Action a.1 provides two options if the AFD is outside the target band and THERMAL POWER is above 90% or 0.9 of APL (whichever is less) RTP: a) to restore the AFD to within limits in 15 minutes; or b) to reduce THERMAL POWER to less than the upper limit specified in the COLR in 15 minutes. Under the same conditions, ITS 3.2.3 ACTION A maintains the 15 minute time limit for the restoration of AFD to within limits, but ITS 3.2.3 ACTION B provides 30 minutes to reduce the THERMAL

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DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

POWER to < 90% RTP. This changes the CTS by allowing an additional 15 minutes to reduce power.

The purpose of CTS 3.2.1 Action a.1 is to restore AFD to within the limits of the LCO. The revised Completion Time allows reactor power to be reduced in a controlled manner without challenging operators or plant systems. It also prioritizes the CTS actions, such that restoring AFD to within the limits should be the first attempted action, followed by (if the first action is not successful) a reduction in THERMAL POWER. The change is acceptable since actions are still provided to reduce power, and only a short time extension (15 minutes) is allowed. In addition, the ITS still provides an action to restore the AFD to within limits in the same time as is currently allowed. This change is designated as less restrictive because additional time is allowed to reduce power than was allowed in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.2.1 Action a.2.a)2) states that when AFD is not within its limit between 50% RTP and 90% or 0.9 of APL (whichever is less) RTP, reduce THERMAL POWER to less than 50% RTP within 30 minutes and reduce the Power Range Neutron Flux - High Trip Setpoints to ≤ 55% of RTP within the next 4 hours. Under the same conditions, ITS 3.2.3 Required Action C.1 only requires THERMAL POWER to be reduced to less than 50% RTP within 30 minutes when AFD is outside of its limit. This changes the CTS by eliminating the requirement to reduce the Power Range Neutron Flux - High Trip Setpoints to ≤ 55% of RTP within the next 4 hours.

The purpose of CTS 3.2.1 Action a.2.a)2) is to reduce THERMAL POWER to the point at which the LCO is met if AFD is not restored within its limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. With the AFD meeting the Technical Specification requirements, further actions are not required to ensure that the assumptions of the safety analyses are met. Increases in THERMAL POWER are governed by ITS LCO 3.0.4, which requires the LCO to be met prior to entering a MODE or other specified condition in which the LCO applies. Therefore, power increases are prohibited while avoiding the risk of changing Reactor Trip System setpoints during operation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.2.1.1 requires the indicated AFD for each OPERABLE excore channel to be determined to be within its limits once per 7 days when the AFD Monitor Alarm is OPERABLE, at least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status if the AFD has been outside the target band in the previous 24 hours, and once per hour for the first 24 hours and once per 30 minutes thereafter when the AFD Monitor Alarm is

DISCUSSION OF CHANGES
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

inoperable. ITS SR 3.2.3.1 requires AFD to be verified within its limits for each OPERABLE excore channel every 7 days. This changes the CTS by eliminating all AFD Surveillance Frequencies based on the OPERABILITY of the AFD Monitor Alarm.

The purpose of ITS 3.2.3 is to ensure that AFD is within its limit. This change is acceptable because the remaining Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the Frequency of monitoring AFD when the AFD Monitor Alarm is inoperable is unnecessary as inoperability of the alarm does not increase the probability that AFD is outside its limit. The AFD Monitor Alarm is for indication only. Its use is not credited in any safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.4 *(Category 7 - Relaxation of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.2.1.3 and CTS 4.2.1.4 require two AFD Surveillances to be performed at the same Frequency as the Allowable Power Level Surveillances in CTS 4.2.6.2. In addition, CTS 4.2.1.3 and CTS 4.2.1.4 state that the provisions of Specification 4.0.4 are not applicable. The CTS 4.2.6.2 Frequency is the first Frequency that occurs of the following: a) upon achieving equilibrium conditions after exceeding 10% or more of RTP, the THERMAL POWER at which the Allowable Power Level was last determined; or b) at least once per 31 EFPD. ITS SR 3.2.3.2 and SR 3.2.3.3 Frequencies for the same Surveillances are: a) once within 31 EFPD after each refueling; and b) 31 EFPD thereafter. The Frequencies of the CTS are changed to be 31 EFPD after a refueling and every 31 EFPD thereafter, and the Specification 4.0.4 allowance is deleted.

The purpose of CTS 4.2.1.3 and CTS 4.2.1.4 is to determine that the AFD is within limits. The change is acceptable due to the slow rate of change of the AFD, and the fact that in most cases during steady state operation, the 31 EFPD Frequency in the CTS will be the more limiting of the two CTS Frequencies. In addition, since the first ITS Frequency is based on being required at a given point (once with 31 EFPD after each refueling), the Specification 4.0.4 allowance is not necessary as entering MODE 1 is allowed in accordance with ITS SR 3.0.4. This change is designated as less restrictive because Surveillances could be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

 (1)

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.3A AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

LCO3.2.1 a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

Action a.2.a) INSERT 1 b. May deviate outside the target band with THERMAL POWER ~~< 90% RTP~~ but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR. (6)

DOC A.2 c. May deviate outside the target band with THERMAL POWER < 50% RTP.

- NOTES -

4.2.1.2 1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

4.2.1.2.a 2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

4.2.1.2.b 3. With THERMAL POWER < 50% RTP and > 15 % RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

Action 2.b) 4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.~~6~~ 7, provided AFD is maintained within acceptable operation limits. (5)

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

6 **INSERT 1**

less than the upper limit specified in the COLR

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AFD (CAOC Methodology)
3.2.3A

1

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action a.1.a	A. THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1 Restore AFD to within target band. INSERT 2	15 minutes	6
Action a.1.b	B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP.	15 minutes INSERT 3	6
Action a.2.a	C. ---NOTE--- Required Action C.1 must be completed whenever Condition C is entered. --- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes	6 6
	~~D. Required Action and associated Completion Time for Condition C not met.~~	~~D.1 Reduce THERMAL POWER to < 15% RTP.~~	~~9 hours~~	2

6 **INSERT 2**

greater than or equal to the upper limit specified in the COLR

6 **INSERT 3**

less than the upper limit specified in the COLR

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CTS

AFD (CAOC Methodology) ①
3.2.3A

SURVEILLANCE REQUIREMENTS

		SURVEILLANCE	FREQUENCY	
4.2.1.1	SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days	
4.2.1.4 4.2.6.2	SR 3.2.3.2	~~Update target flux difference of each OPERABLE excore channel by:~~ ~~a. Determining the target flux difference in accordance with SR 3.2.3.3, or~~ ~~b. Using linear interpolation between the most recently measured value, and either the predicted value for the end of cycle or 0% AFD.~~	Once within 31 EFPD after each refueling AND 31 EFPD thereafter	③
4.2.1.3 4.2.6.2	SR 3.2.3.3	- NOTE - The initial target flux difference after each refueling may be determined from design predictions. Determine, by measurement, the target flux difference.	Once within 31 EFPD after each refueling AND ~~92~~ 31 EFPD thereafter	④

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

1. The CAOC methodology and the Specification designator "A" are deleted since they are unnecessary (only one AFD Specification is used in the CNP ITS). This information is provided in NUREG-1431, Rev. 2 to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the RAOC methodology Specification (ISTS 3.2.3B) is not used and is not shown.

2. ISTS 3.2.3 ACTION D has been deleted. With the AFD limits not met when between 50% RTP and 90% or 0.9 of APL (whichever is less) RTP, the CTS only requires a power reduction to < 50% RTP. The only restriction between 15% RTP and 50% RTP is to measure the penalty deviation time. This penalty deviation time is only used when at 50% RTP or greater. The CTS allows an unlimited amount of time to operate below 50% RTP with the AFD limits not met. Therefore, the requirement to reduce power to < 15% RTP within 9 hours has been deleted. In addition, since ACTIONS do not have to be completed if the Condition is not applicable (as described in LCO 3.0.2), once power is reduced to below 50% RTP Condition D does not apply. Thus, completion of the reduction of power to < 15% RTP would not be required. The only way the reduction to < 15% RTP would be required is if Condition D had a Note similar to the Note in Condition C.

3. TSTF-24, Rev. 1 was approved by the NRC on April 22, 1998. However, when NUREG-1431, Rev. 2 was issued, this TSTF was not completely incorporated. Therefore, changes approved by TSTF-24, Rev. 1 have been made.

4. The CTS requires the AFD to be determined at least every 31 EFPD. Therefore, ITS SR 3.2.3.3 will maintain the current Frequency, in lieu of extending it to 92 EFPD as allowed by the NUREG-1431, Rev. 2.

5. Changed due to change made in ITS 3.3.1.

6. ISTS 3.2.3A uses 90% RTP as a decision point in determining penalty deviation time and ACTIONS to be entered when the AFD limits are not met. The decision point used in CTS 3.2.1 is 90% RTP or 0.9 of APL, whichever is less. As discussed in ITS 3.2.1 DOC A.4, the APL limit requirement has been changed to $F^W_Q(Z)$; thus the corrected term, if used in the ITS, would be 0.9 of [(CFQ) x K(Z)]/$F^W_Q(Z)$. The extra CTS decision point (0.9 of [(CFQ) x K(Z)]/$F^W_Q(Z)$, whichever is less) would only be effective if the $F^W_Q(Z)$ is not within limit, which is not a common occurrence. Instead of using this modified decision point (i.e., 90% RTP or 0.9 of [(CFQ) x K(Z)]/$F^W_Q(Z)$, whichever is less) in the various locations of ITS 3.2.3, the term "upper limit specified in the COLR" is used. The nominal value of 90% RTP and 0.9 of APL RTP, whichever is less, is part of the AFD limit curve that is currently allowed to be in the COLR, as described in CTS 3.2.1 (and continues to be allowed to be in the COLR by the ISTS).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

1

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3A AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, (CAOC) involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers. INSERT 1 1

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

2

plant

The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is ≥ 90% RTP. During operation at THERMAL POWER levels < 90% RTP but ≥ 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours. INSERT 3 2

6 INSERT 1A

INSERT 1B 6

50 INSERT 2

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

1 **INSERT 1**

Constant Axial Offset Control

6 **INSERT 1A**

greater than or equal to the upper limit specified in the COLR (normally 90% RTP)

6 **INSERT 1B**

less than the upper limit specified in the COLR

2 **INSERT 2**

and at THERMAL POWER levels < 50% RTP but > 15% RTP,

2 **INSERT 3**

and > 2 hours, respectively,

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 (1)

BASES

BACKGROUND (continued)

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Ref~~s~~. 1~~, 2, and 3~~) entails: (2)

a. Establishing an envelope of allowed power shapes and power densities~~;~~

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes~~;~~. (8)

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics~~;~~ and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled ~~bank~~ withdrawal and rod (2) boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

The ~~limits on the~~ AFD satisf~~y~~ ies Criterion 2 of 10 CFR 50.36(c)(2)(ii). (2)

BASES

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 4). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI. (2) (2)

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated. (6) (6)

INSERT 3A

INSERT 3B

plant

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

~~Figure B 3.2.3A-1 shows a typical target band and typical AFD acceptable operation limits.~~ (3)

The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band. Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return

6 **INSERT 3A**

greater than or equal to the upper limit specified in the COLR (normally 90% RTP)

6 **INSERT 3B**

greater than or equal to the upper limit specified in the COLR

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 (1)

BASES

LCO (continued)

to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part b of this LCO (i.e., THERMAL POWER ≥50% RTP). The cumulative penalty time is the sum of penalty times from Parts b and c of this LCO.

INSERT 3C (6) (4)

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

(7) For surveillance of the power range channels performed according to SR 3.3.1.6, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. ~~This calibration is performed every 92 days.~~ (4) (6) (2) (5)

may be

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1).

INSERT 3D (6)

Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

6 **INSERT 3C**

less than the upper limit specified in the COLR

6 **INSERT 3D**

the upper limit specified in the COLR

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 (1)

BASES

APPLICABILITY (continued)

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

INSERT 3E

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time. (6)

B.1

INSERT 3F

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR. (6)

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time. (6)

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter. (6)

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

6 **INSERT 3E**

greater than or equal to the upper limit specified in the COLR

6 **INSERT 3F**

less than the upper limit specified in the COLR

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BASES

ACTIONS (continued)

Condition C is modified by a Note that requires that Required Actions C.1 and C.2 must be completed whenever this Condition is entered. (4)

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid. (6)

Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band. (9)

SR 3.2.3.2

This Surveillance requires that the target flux difference is updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.3. Insert 4 (2)

2 INSERT 4

Updating the target flux difference includes updating the target band.

 (1)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.~~ (7)

SR 3.2.3.3

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and ~~92~~ 31 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore excore calibrations that may have occurred in the interim. (6)

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling. INSERT 5 (2)

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

~~2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980.~~ (2)

~~3. C. Eicheldinger to D. B. Vassallo (Chief of Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975.~~

(2) ~~4~~ 2 UFSAR, ~~Chapter 15~~ Section 7.4 (2)

2 **INSERT 5**

WCAP-8385 (Westinghouse proprietary) and

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AFD (CAOC Methodology)
B 3.2.3A

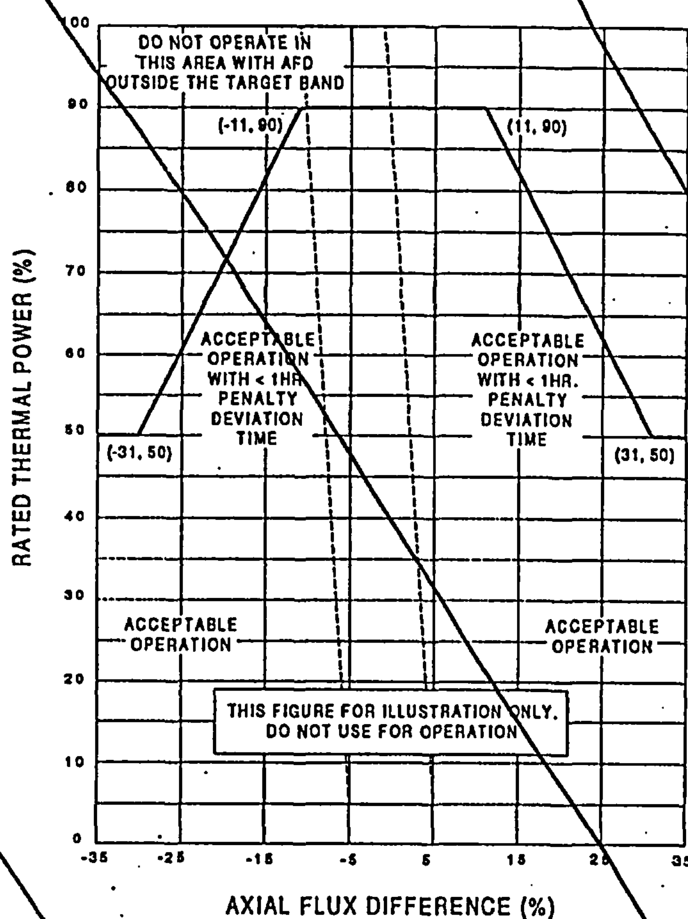

Figure B 3.2.3A-1 (Page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits and Target Band Limits as a Function of RATED THERMAL POWER

3

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3 BASES, AXIAL FLUX DIFFERENCE (AFD)

1. The methodology (CAOC) and the Specification designator "A" are deleted since they are unnecessary (only one AFD Specification is used in the CNP ITS). This information is provided in NUREG-1431, Rev. 2 to assist in identifying the appropriate Specification to be used as a model for the plant specific ITS conversion, but serves no purpose in a plant specific implementation. In addition, the RAOC methodology Specification Bases (ISTS B 3.2.3B) is not used and is not shown.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. Since the ITS states the actual AFD target band and operation limits are specified in the COLR, the "typical" example is not needed in the Bases and has been deleted.

4. Typographical/grammatical error corrected.

5. The Frequency of the CHANNEL CALIBRATION is not necessary in these Bases, since the Bases for ITS SR 3.3.1.6 are sufficient.

6. Changes are made to reflect those changes made to the ITS.

7. This option has been deleted since it is not used at CNP.

8. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

9. This Bases statement, which adds an additional requirement above that required by the actual SR, has been deleted. The actual SR provides only a 7 day Frequency for verifying AFD. Any requirement for more frequent checks of AFD is more appropriately located in plant procedures, not the Bases.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.3, AXIAL FLUX DIFFERENCE (AFD)

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 4

ITS 3.2.4, Quadrant Power Tilt Ratio

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

ITS 3.2.4

ITS

A.1

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

LCO 3.2.4 — 3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER~~*~~ — A.2

ACTION:

ACTION A, ACTION B

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Within 2 hours:

a) ~~Either reduce the QUADRANT POWER TILT RATIO to within its limit, or~~ — A.3

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 ~~and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.~~ — L.1

~~2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

Add proposed Required Actions A.2, A.3, A.4, A.5, A.6, and ACTION B — L.2

ACTION A, ACTION B

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within ~~30 minutes~~ 2 hours — L.3

~~2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or~~

Add proposed Required Actions A.2, A.3, A.4, A.5, A.6, and ACTION B — L.2

~~*See Special Test Exception 3.10.2~~ — A.2

ITS 3.2.4

A.1

ITS

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

~~reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

L.2

ACTION A, ACTION B

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

~~1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.~~

~~2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.~~

Add proposed Required Actions A.1, A.2, A.3, A.4, A.5, A.6, and ACTION B

L.2

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.2.4.1 Note 2

L.4

SR 3.2.4.1

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days ~~when the alarm is OPERABLE.~~

~~b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.~~

L.5

SR 3.2.4.2

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is greater than 75 percent of RATED THERMAL POWER.

D. C. COOK - UNIT 1 3/4 2-12 AMENDMENT NO. 155, 120

ITS 3.2.4

ITS

A.1

DEFINITIONS

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

See ITS Chapter 1.0

QUADRANT POWER TILT RATIO

SR 3.2.4.1 Note 1

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average. and THERMAL POWER ≤ 75% RTP

A.4

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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 those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," or in NRC Regulatory Guide 1.109 Rev 1, October 1977.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

See ITS Chapter 1.0

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ITS 3.2.4

A.1

ITS

POWER DISTRIBUTION LIMITS

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

LCO 3.2.4 — 3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER* — A.2

ACTION:

ACTION A, ACTION B

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09:

1. Within 2 hours:

a) ~~Either reduce the QUADRANT POWER TILT RATIO to within its limit, or~~ — A.3

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 ~~and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.~~ — L.1

2. ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~

3. ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

Add proposed Required Actions A.2, A.3, A.4, A.5, A.6, and ACTION B — L.2

ACTION A, ACTION B

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within ~~30 minutes~~ 2 hours — L.3

2. ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or~~

Add proposed Required Actions A.2, A.3, A.4, A.5, A.6, and ACTION B — L.2

~~*See Special Test Exception 3.10.2.~~ — A.2

ITS 3.2.4

ITS

A.1

POWER DISTRIBUTION

ACTION: (Continued)

~~reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~

~~3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~ L.2

ACTION A, ACTION B — c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

~~1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.~~

~~2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.~~

Add proposed Required Actions A.1, A.2, A.3, A.4, A.5, A.6, and ACTION B — L.2

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.2.4.1 Note 2 — L.4

SR 3.2.4.1 — 4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER By:

a. Calculating the ratio at least once per 7 days ~~when the alarm is OPERABLE.~~

~~b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.~~ L.5

SR 3.2.4.2 — c. Using the movable incore detectors to confirm that the power distribution is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

D. C. COOK - UNIT 2 3/4 2-14 Amendment No. 10

ITS 3.2.4

ITS

A.1

DEFINITIONS

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

See ITS Chapter 1.0

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, and THERMAL POWER ≤ 75% RTP the remaining three detectors shall be used for computing the average.

SR 3.2.4.1 Note 1

A.4

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/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 those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," or in NRC Regulatory Guide 1.109 Rev. 1, October 1977.

See ITS Chapter 1.0

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the CNP Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 2, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 The Applicability of CTS 3.2.4 is modified by footnote * stating "See Special Test Exception 3.10.2." ITS 3.2.4 Applicability does not contain the footnote or a reference to the Special Test Exception.

The purpose of the CTS 3.2.4 footnote * reference is to alert the user that a Special Test Exception exists which may modify the Applicability of the Specification. It is an ITS convention to not include these types of footnotes or cross-references. This change is designated as an administrative change since it does not result in technical changes to the CTS.

A.3 CTS 3.2.4 Action a.1.a) states that with QPTR > 1.02 and ≤ 1.09, within 2 hours reduce the QPTR to within its limit. ITS 3.2.4 does not contain a Required Action stating QPTR must be reduced to within its limit.

This change is acceptable because the technical requirements have not changed. Restoration of compliance with the LCO is always an available Required Action and it is the convention in the ITS to not state such "restore" options explicitly unless it is the only action or is required for clarity. This change is designated as an administrative change since it does not result in technical changes to the CTS.

A.4 CTS 1.18, the definition of QPTR, states, in part, that "With one excore detector inoperable, the remaining three detectors shall be used for computing the average." ITS SR 3.2.4.1 Note 1, which incorporates the QPTR definition portion described above, states that when one Power Range Neutron Flux channel (i.e., an excore detector) is inoperable and THERMAL POWER is ≤ 75% RTP, the remaining three Power Range Neutron Flux channels can be used for calculating QPTR. This changes the CTS by specifying the allowance can only be used when ≤ 75% RTP.

The purpose of the CTS is to state when fewer than the normal complement of excore detectors can be used to determine QPTR. CTS 4.2.4.a requires the QPTR to be calculated once per 7 days. CTS 4.2.4.c requires the QPTR to be determined using the incore detectors every 12 hours when an excore detector is inoperable and THERMAL POWER is > 75% RTP. Thus, this effectively means that the one excore detector inoperable allowance can only be used when ≤ 75% RTP. When > 75% RTP with one excore detector inoperable, CTS 4.2.4.c must be performed to determine QPTR. Therefore, this change is designated as an administrative change and is acceptable since it does not result in a technical change to the CTS.

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 Action a.1.b) states that when QPTR is > 1.02 but ≤ 1.09, reduce THERMAL POWER at least 3% from RTP for each 1% of indicated QPTR in excess of 1.0, and similarly reduce the Power Range Neutron Flux - High Trip Setpoints within the next 4 hours. ITS 3.2.4 Required Action A.1 includes the requirement to reduce THERMAL POWER similar to the CTS, but does not include a requirement to reduce the Power Range Neutron Flux - High Trip Setpoints. This changes the CTS by eliminating the requirement to reduce the Power Range Neutron Flux - High Trip Setpoints.

The purpose of CTS 3.2.4 Action a.1.b) is to reduce THERMAL POWER to increase the margin to the core power distribution limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. With THERMAL POWER reduced by 3% from RTP for each 1% QPTR > 1.00, further actions are not required to ensure that THERMAL POWER is not increased. Power increases are administratively prohibited by the Technical Specifications while avoiding the risk of changing Reactor Trip System setpoints during operation. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.2.4 Action a.2 states that with QPTR > 1.02 and ≤ 1.09, verify that QPTR is within its limit within 24 hours or reduce THERMAL POWER to < 50% RTP within the next 2 hours and reduce the Power Range Neutron Flux - High Trip Setpoints to ≤ 55% RTP within the next 4 hours. CTS 3.2.4 Action b.2 states that when QPTR is > 1.09 due to misalignment of a RCCA, verify that QPTR is within its limit within 2 hours or reduce THERMAL POWER to < 50% RTP within the next 2 hours and reduce the

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DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

Power Range Neutron Flux - High Trip Setpoints to ≤ 55% RTP within the next 4 hours. CTS 3.2.4 Action c.1 states that when QPTR is > 1.09 for reasons other than misalignment of a RCCA, reduce THERMAL POWER to < 50% RTP within the next 2 hours and reduce the Power Range Neutron Flux - High Trip Setpoints to ≤ 55% RTP within the next 4 hours. CTS 3.2.4 Actions a.3, b.3, and c.2 state that the cause of the out of limit QPTR must be identified and corrected prior to increasing THERMAL POWER, and that subsequent operation above 50% RTP may proceed provided that the QPTR is verified to be within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RTP. ITS 3.2.4, Required Action A.2 requires the QPTR to be determined within 12 hours, Required Action A.3 requires $F_Q(Z)$ and $F^N_{\Delta H}$ to be verified to be within limit within 24 hours of achieving equilibrium conditions after the power reduction and every 7 days thereafter, Required Action A.4 requires the safety analyses to be reevaluated to confirm the results are still valid for the duration of operation under this condition prior to increasing power, Required Action A.5 requires (after completion of Required Action A.4) the excore detectors to be normalized to restore QPTR within limit prior to increasing power, and Required Action A.6 requires $F_Q(Z)$ and $F^N_{\Delta H}$ to be verified to be within limits within 24 hours after achieving equilibrium condition at RTP not to exceed 48 hours after increasing power. In addition, for the condition of QPTR > 1.09 for reasons other than misalignment of a RCCA, ITS 3.2.4 Required Action A.1 requires THERMAL POWER to be reduced ≥ 3% from RTP for each 1% of QPTR > 1.00, similar to the CTS Actions a.1.b) and b.1. Furthermore, ITS 3.2.4 ACTION B states that with a Required Action and associated Completion Time (of Condition A) not met, reduce THERMAL POWER to ≤ 50% RTP within 4 hours. This changes the CTS by eliminating requirements to be ≤ 50% RTP within a specified time of exceeding the LCO and substituting compensatory measures in ITS ACTION A, which if not met, result in a reduction in power per ITS ACTION B.

The purpose of the CTS actions is to lower reactor power to less than 50% when QPTR is not within its limit and cannot be restored to within its limit within a reasonable time period. In addition, the Power Range Neutron Flux - High Trip Setpoints are reduced to ≤ 55% to ensure that reactor power is not inadvertently increased without QPTR within its limit. This action is taken because with QPTR not within limit, the core power distribution is not within the analyzed assumptions, and critical core parameters such as $F_Q(Z)$ and $F^N_{\Delta H}$ may not be within their limits. A QPTR not within limit may not be an unacceptable condition if the critical core parameters such as $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features or restore out of limit parameters. The Required Actions are consistent with safe operation under the specified Condition, considering the status of the redundant indications, the capacity and capability of remaining features, a reasonable time for repairs or restoration of required features, and the low probability of a DBA occurring during the repair period. The ITS requires measurement of $F_Q(Z)$ and $F^N_{\Delta H}$ within 24 hours and every 7 days thereafter to verify that those parameters are within limit. In addition, the ITS requires the safety analyses to be reevaluated to ensure that the results remain valid. Assuming that these actions are successful, the ITS allows indefinite

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

operation with QPTR out of its limit and allows the excore nuclear detectors to be normalized to eliminate the indicated QPTR. This ensures that the core is operated within the safety analyses. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 3 – Relaxation of Completion Time)* CTS 3.2.4 Action b.1, which applies when QPTR is > 1.09 due to misalignment of a RCCA, requires a THERMAL POWER reduction of 3% from RTP for every 1% QPTR exceeds 1.0 within 30 minutes. ITS 3.2.4 Required Action A.1 requires a THERMAL POWER reduction of 3% from RTP for every 1% QPTR exceeds 1.0 within 2 hours. This changes the CTS by allowing 2 hours to perform the required power reduction.

The purpose of CTS 3.2.4 is provide appropriate compensatory measures for QPTR greater than that assumed in the safety analyses. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering other indications available to the operator, a reasonable time for restoring compliance with the LCO, and the low probability of a DBA occurring during the restoration period. Under the ITS, a QPTR of 1.09 would require THERMAL POWER to be reduced to ≤ 73% RTP. This will provide sufficient thermal margin to account for the radial power distribution. In addition, the 2 hour time limit is consistent with the CTS time allowed when QPTR is > 1.02 but ≤ 1.09. This change is designated as less restrictive because additional time is allowed to decrease power than was allowed in the CTS.

L.4 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.2.4.a states that QPTR shall be determined to be within the limit by calculating the ratio at least once per 7 days. ITS SR 3.2.4.1 Note 2 states that SR 3.2.4.2, which requires verification of QPTR using the movable incore detectors, may be performed in lieu of SR 3.2.4.1. This changes the CTS by allowing the movable incore detectors to be used to determine QPTR instead of the excore detectors.

The purpose of CTS 4.2.4.a is to periodically verify that QPTR is within limit. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are sufficient for verification that the parameters meet the LCO. The movable incore detector system provides a more accurate indication of QPTR than the excore detectors. In fact, the movable incore detector system is used to calibrate the excore detectors. Therefore, allowing the use of the movable incore detector system or the excore detectors is appropriate. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L.5 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.2.4.a requires the QPTR to be verified to be within limit every 7 days when the QPTR alarm is OPERABLE and CTS 4.2.4.b requires the verification every 12 hours when the QPTR alarm is inoperable. ITS SR 3.2.4.1 requires verification that QPTR is within limit every 7 days. This changes the CTS by eliminating the requirement to verify QPTR more frequently when the QPTR alarm is inoperable.

DISCUSSION OF CHANGES
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

The purpose of CTS 4.2.4.a and CTS 4.2.4.b is to periodically verify that QPTR is within limit. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Increasing the frequency of QPTR verification when the QPTR alarm is inoperable is unnecessary as inoperability of the alarm does not increase the probability that QPTR is outside its limit. The QPTR alarm is for indication only. Its use is not credited in any of the safety analyses. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

QPTR
3.2.4

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

Actions a, b, c

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	AND	
Determine QPTR	A.2 Perform SR 3.2.4.1.	Once per 12 hours (1)
	AND	
	A.3 Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 (2) AND Once per 7 days thereafter
	AND	

QPTR
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CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Actions a, b, c	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
	A.5	- NOTE - 5 1. Perform Required Action A.5 only after Required Action A.4 is completed. 4 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed.	3
		Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
	A.6	- NOTE 8 - Perform Required Action A.6 only after Required Action A.5 is completed.	3
		Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
Actions a, b, c	B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

		SURVEILLANCE	FREQUENCY
QPTR definition DOC L.4 4.2.4.a	SR 3.2.4.1	- NOTES - 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. (4) 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. Verify QPTR is within limit by calculation.	7 days
4.2.4.c	SR 3.2.4.2	- NOTE - Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. Verify QPTR is within limit using the movable incore detectors.	12 hours

3

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

1. TSTF-109, Rev. 0 was approved by the NRC on October 28, 1996. However, when NUREG-1431, Rev. 2 was issued, this TSTF was not completely incorporated. Therefore, this change approved by TSTF-109, Rev. 0 has been made.

2. TSTF-314, Rev. 0 was approved by the NRC on January, 11 1999. However, when NUREG-1431, Rev. 2 was issued, this TSTF was not completely incorporated. Therefore, this change approved by TSTF-314, Rev. 0 has been made.

3. Typographical/grammatical errors corrected.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1); 1

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 *departure from nucleate boiling (DNB) criterion)* that the hot fuel rod in the core does not experience a DNB condition; 1

c. During an ejected rod accident, the ~~energy deposition to the fuel must not exceed 280 cal/gm~~ (Ref. 2); and INSERT 1 2 1

d. ~~The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).~~ INSERT 2 3

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

2 **INSERT 1**

average fuel pellet enthalpy at hot spot is below 200 cal/gm for irradiated and unirradiated fuel

3 **INSERT 2**

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast enough to prevent exceeding acceptable fuel damage limits. SDM should assure subcriticality with the most restrictive rod cluster control assembly fully withdrawn (Ref. 3).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses. 9

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged. 6

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

2 from 4

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level and increasing power up to this revised limit.

BASES

ACTIONS (continued)

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F_Q(Z)$, ~~as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$~~, and $F_{\Delta H}^N$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction ~~under~~ per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from Thermal Power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the ~~plant~~ unit and perform a flux map. If these peaking factors are not within their limits, the Required Actions of ~~these Surveillances~~ applicable LCOs provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

(5) (6) (a) (6) (2) (6) (6)

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best

QPTR
B 3.2.4

BASES

ACTIONS (continued)

characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

is still ing but

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the ~~excore detectors are~~ normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR. INSERT 3 (7)

can be

Required Action A.5 is modified by two Notes. Note 1 states that the QPTR is not restored to within limits ~~until after the re-evaluation of the~~ safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions. INSERT 4 (7) (6)

cannot be

A.6

QPTR INSERT 4 (7)

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$, ~~as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$~~, and $F^N_{\Delta H}$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time. (5) (7) (6)

INSERT 5

7 **INSERT 3**

. Any normalization must be performed

7 **INSERT 4**

via normalization of the excore detectors

7 **INSERT 5**

by performing SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1

BASES

ACTIONS (continued)

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power. 6 6

B.1

any and

If Required Actions ~~A.1 through A.6 are not completed within their~~ associated Completion Times the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems. 8 2

is not met, other specified

unit

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 ~~for three and four loop cores~~. (2)

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. Regulatory Guide 1.77, Rev 0, May 1974. INSERT 6 (2)
3. 10 CFR 50, Appendix A, GDC 26. INSERT 7 (3)

2 **INSERT 6**

UFSAR, Section 14.2.6.7 (Unit 1) and Section 14.2.6.1.2 (Unit 2).

3 **INSERT 7**

UFSAR, Section 1.4.5.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.2.4 BASES, QUADRANT POWER TILT RATIO (QPTR)

1. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. CNP Units 1 and 2 were designed and under construction prior to the promulgation of 10 CFR 50, Appendix A. CNP Units 1 and 2 were designed and constructed to meet the intent of the proposed General Design Criteria, published in 1967. However, the CNP UFSAR contains discussions of the Plant Specific Design Criteria (PSDCs) used in the design of CNP Units 1 and 2. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section and description in the UFSAR.

4. Editorial changes are made for consistency with the ITS. ITS 3.2.4 Required Action A.1 requires that THERMAL POWER be reduced "$\geq$ 3% from RTP" for each 1% of QPTR > 1.00. The ISTS Bases state that power is reduced "3% RTP" for each 1% of QPTR > 1.00. The Bases are revised to be consistent with the Specification.

5. The peaking factor $F_Q(Z)$ is sufficient. There is no need to state how it is approximated.

6. Typographical/grammatical error corrected.

7. Changes added for clarity and to be consistent with the Specification.

8. These changes have been made to be consistent with similar phrases in other parts of the ITS Bases and to be consistent with the Specification.

9. The statement concerning why $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained has been deleted, since it is duplicative of statements provided in the individual Bases for the two factors (ITS 3.2.2 and ITS 3.2.1). The Bases for QPTR is not appropriate for describing why other factors, covered by their own Technical Specifications, are required.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.4, QUADRANT POWER TILT RATIO (QPTR)

There are no specific NSHC discussions for this Specification.

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