PACKAGE 3.0

LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

CROSS - REFERENCE

CURRENT TECHNICAL SPECIFICATIONS

TO

IMPROVED TECHNICAL SPECIFICATIONS

List of Section Cross - References

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

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Conversion Submittal

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Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 3.0				
3.0.A		LCO	3.0.1	
3.0.A		LCO	3.0.2	
3.0.B		LCO	3.0.2	
3.0.C		LCO	3.0.3	
New		LCO	3.0.4	
New		LCO	3.0.5	
New		LCO	3.0.6	
New		LCO	3.0.7	
New		LCO	3.0.8	
New		LCO	3.0.9	

Current Technical Specification Cross-Reference

CTS Section	CTS Table Item Number	Section Type	ITS Section	ITS Table Item Number
CTS Section 4.0				
4.0.A		SR	3.0.2	
4.0.B		SR	3.0.1	
4.0.B		SR	3.0.3	
New		SR	3.0.4	

PACKAGE 3.0

LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

CROSS - REFERENCE

IMPROVED TECHNICAL SPECIFICATIONS

TO

CURRENT TECHNICAL SPECIFICATIONS

Section Cross - Reference

3.0

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
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Improved Technical Specification Cross-Reference

ITS Section	ITS Table Item Number	Section Type	CTS Section	CTS Table Item Number
ITS Section 3.0				
3.0.1		LCO	3.0.A	
3.0.2		LCO	3.0.A	
3.0.2		LCO	3.0.B	
3.0.3		LCO	3.0.C	
3.0.3		LCO	New	
3.0.4		LCO	New	
3.0.5		LCO	New	
3.0.6		LCO	New	
3.0.7		LCO	New	
3.0.8		LCO	New	
3.0.9		LCO	New	
3.0.1		SR	4.0.B	
3.0.2		SR	4.0.A	
3.0.2		SR	New	
3.0.3		SR	4.0.B	
3.0.4		SR	New	

ITS PACKAGE CONTENTS

Package:

3.1

1.	Part A	Introduction
2.	Part B	Proposed PI ITS and Bases
3.	Part C	Markup of PI CTS
4.	Part D	DOC to PI CTS
5.	Part E	Markup of ISTS and Bases
6.	Part F	JD from ISTS
7.	Part G	NSHD for changes to PI CTS
8.	Cross-Reference CTS to ITS	
9.	Cross-Reference ITS to CTS	

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART A

INTRODUCTION

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

LICENSE AMENDMENT REQUEST DATED December 11, 2000
Conversion to Improved Standard Technical Specifications

3.1
PART A

Introduction to the Discussion of the proposed Changes to the Current Technical Specifications, Justification of Differences from the Improved Standard Technical Specifications, and the supporting No Significant Hazards Determination

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose changes to the Facility Operating Licenses and Appendix A, Technical Specifications, as follows and as presented in the accompanying Parts B through G of this Package.

BACKGROUND

Over the past several years the nuclear industry and the Nuclear Regulatory Commission (NRC) have jointly developed Improved Standard Technical Specifications (ISTS). The NRC has encouraged licensees to implement these improved technical specifications as a means for improving plant safety through the more operator-oriented technical specifications, improved and expanded bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

This License Amendment Request (LAR) is submitted to conform the Prairie Island Nuclear Generating Plant (PINGP) Current Technical Specifications (CTS) to NUREG-1431, Improved Standard Technical Specifications, Westinghouse plants, Revision 1 issued April 1995 (ISTS). The resulting new Technical Specifications (TS) for Prairie Island (PI) are the PI Improved Technical Specifications (ITS) which incorporates the PI plant specific information.

NUREG-1431 is based on a hypothetical four loop Westinghouse plant. Since PI is similar in design and vintage to the R.E. Ginna Nuclear Power Plant which has already completed conversion to improved technical specifications, this amendment request relies on the Ginna ITS.

This LAR is also supported by Parts B through G. Part B contains a "clean" copy of the proposed PI ITS and Bases. Part C contains a mark-up of the PI CTS. Part D is the Description of Changes (DOC) to the PI CTS. Part E is a mark-up of the ISTS and Bases which shows the deviations from the standard incorporated to meet PI plant specific requirements. Part F gives the Justification for Deviations (JFD) from the ISTS and Part G provides the No Significant Hazards Determinations (NSHD) for changes to the PI CTS. To facilitate review of this LAR, cross-reference numbers from changes and deviations to the corresponding DOC, JFD and NSHD are provided. The methodology for mark-up and cross-references are described in the next section.

MARK-UP METHODOLOGY

The TS conversion package includes mark-ups of the CTS, the ISTS and the ISTS Bases in accordance with this guidance. Mark-up may be electronic or by hand as indicated.

Current Technical Specifications

The mark-up of the CTS is provided to show where current requirements are placed in the ITS, to show the major changes resulting from the conversion process, and to allow reviewers to evaluate significant differences between the CTS and ITS.

This ITS conversion LAR has been prepared in 14 packages following the Chapter/Section outline of the ITS as follows: 1.0, 2.0, 3.0, 3.1 . . . 3.9, 4.0 and 5.0. Accordingly, each package contains all the elements of Parts A through G as described above. The CTS Bases are not included in the CTS mark-up packages since the Bases have been rewritten in their entirety.

The current Specifications addressed by the associated ITS Chapter/Section are cross-referenced in the left margin to the new ITS location by Specification number and type (G-General, SL-Safety Limit, LCO-Limiting Condition for Operation or SR-Surveillance Requirements). Those portions of each CTS page which are not addressed in the associated ITS Chapter/Section are shadowed (electronic) or clouded and crossed out (by hand) and in the right margin is the comment, "Addressed Elsewhere".

The CTS are marked-up to incorporate the substance of NUREG-1431 Revision 1. It is not the intent to mark every nuance required to make the format change from CTS to ITS.

In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.

Some apparent changes result from the different conventions and philosophies used in the ITS. Generally these apparent changes will not be marked-up in the CTS if there is no resulting change in plant operating requirements.

Changes are identified by a change number in the right margin which map the changed specification requirement to Part D, Discussion of Changes, and Part G, No Significant Hazards Determination (NSHD) and indicate the NSHD category. The change number form is R3.4-02 where the first two numbers, 3.4 in this example, refer to ITS Chapter/Section number 3.4, and the second number, 02 in this example, is a sequentially assigned number for changes within that Chapter/Section, starting with 01. The prefix letter(s) indicates the classification of the change impact. For CTS changes this is also the NSHD category.

The change impact categories defined below conveniently group the type of changes for consideration of the effect of the change on the current plant license in Part D and are also useful for efficient discussion in Part G the "No Significant Hazards Determination" (NSHD) section. If the same change is made in Part E, then the change impact category will also show up in the change number in Part F. These categories are:

A - Administrative changes, editorial in nature that do not involve technical issues. These include reformatting, renaming (terminology changes), renumbering, and rewording of requirements.

L - Less restrictive requirements included in the PI ITS in order to conform to the guidance of NUREG-1431. Generally these are technical changes to existing TS which may include items such as extending Completion Times or reducing Surveillance Frequencies (extended time interval between surveillances). The less restrictive requirements necessitate individual justification. Each is provided with its specific NSHD.

LR - Less restrictive Removal of details and information from otherwise retained specifications which are removed from the CTS and placed in the Bases, Technical Requirements Manual (TRM), Updated Safety Analysis Report (USAR) or other licensee controlled documents. These changes include details of system design and function, procedural details or methods of conducting surveillances, or alarm or indication-only instrumentation.

M - More restrictive requirements included in the PI ITS in order to provide a complete set of Specifications conforming to the guidance of NUREG-1431. Changes in this category may be completely new requirements or they may be technical changes made to current requirements in the CTS.

R - Relocation of Current Specifications to other controlled documents or deletion of current Specifications which duplicate existing regulatory requirements.

Current requirements in the LCOs or SRs that do not meet the 10 CFR 50.36 selection criteria and may be relocated to the Bases, USAR, Core Operating Limits Report (COLR), Operational Quality Assurance Plan (OQAP), plant procedures or other licensee controlled documents. Relocating requirements to these licensee controlled documents does not eliminate the requirement, but rather, places them under more appropriate regulatory controls, such as 10CFR 50.54 (a)(3) and 10 CFR 50.59, to manage their implementation and future changes. Maintenance of these requirements in the TS commands resources which are not commensurate with their importance to safety and distract resources from more important requirements. Relocation of these items will enable more efficient maintenance of requirements under existing regulations and reduce the need to request TS changes for issues which do not affect public safety.

Deletion of Specifications which duplicate regulations eliminates the need to change Technical Specifications when changes in regulations occur. By law, licensees shall meet applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions; therefore, restatement in the Technical Specifications is unnecessary.

The methodology for marking-up these changes is as follows:

As discussed above, administrative changes may not be marked-up in detail. Portions of the specifications which are no longer included are identified by use of the electronic strike-out feature (or crossed out by hand). Information being added is inserted into the specification in the appropriate location and is identified by use of shading features (or handwritten/insert pages).

Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS mark-up is to identify changes from the ISTS required to create a plant specific ITS by incorporating plant specific values in bracketed fields and identifying other changes with cross-reference to the Part F Justification For Differences.

All deviations from the ISTS are cross-referenced to the Part F justification for differences by a change number in the right margin. The change number form is CL3.4-05 where the prefix letter(s), CL in this example, indicate the classification of the reason for the difference, the first two numbers, 3.4 in this example, refer to the ITS Chapter/Section number 3.4, and the second number, 05 in this example, is a sequentially assigned number for deviations within that Chapter/Section, starting with a number which is larger than the last number from the Part C CTS mark-up. In some instances where a change has been made to the CTS and ISTS, the Part D change number is given since the justification for difference is the same as the discussion of change. The following categories are used as prefixes to indicate the general reason for each difference:

CL - Current Licensing basis. Issues that have been previously licensed for PI and have been retained in the ITS. This includes Specifications dictated by plant design features or the design basis. Since no plant modifications have been or will be made to accommodate conversion to ITS, the plant design basis features shall be incorporated into the PI ITS.

PA - Plant, Administrative. Plant specific wording preference or minor editorial improvements made to facilitate operator understanding.

TA - Traveler, Approved. Deviations made to incorporate an industry traveler which has been approved by the NRC.

TP - Traveler, Proposed. Deviation made to incorporate a proposed industry traveler which as of the time of submittal has not been approved by the NRC.

X - Other, Deviation from the ISTS for any other reason than those given above.

Material which is deleted from the ISTS is identified by use of the WordPerfect strike-out feature (or crossed out by hand). Information being added to the ISTS to generate the PI ITS due to any of the deviations discussed above is identified by use of WordPerfect red-line features (or handwritten/insert pages).

Bracketed Information

Many parameters, conditions, notes, surveillances, and portions of sections are bracketed in the ISTS recognizing that plant specific values are likely to vary from the "generic" values provided in the standard.

If the bracketed value applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. In some instances, bracketed material is not discussed. If bracketed material is discussed, a change number is provided which includes the appropriate prefix as described above. When bracketed "generic" material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand), the plant specific information is substituted for the bracketed information and a change number is provided which includes the appropriate prefix. Information added is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages).

Optional Sections

Due to differing Westinghouse plant designs and methodologies, some ISTS section numbers include a letter suffix indicating that only one of these sections is applicable to any specific plant. The appropriate section is indicated in the Table of Contents, the suffix letter is deleted, and justification, if required, is included in the appropriate Chapter/Section package.

<u>Bases, Improved Standard Technical Specifications (NUREG-1431, Rev. 1)</u>

The ISTS Bases have been marked-up to support the plant specific PI ITS and allow reviewers to identify changes from NUREG-1431. To the extent possible, the words of NUREG-1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading with respect to Prairie Island, they have been revised. In addition, descriptions have been added to cover plant specific portions of the specifications. Change numbers have been provided for the ISTS Bases with the same format as the ISTS Specification mark-up. In some instances, the same change number is used to describe the change.

Material which is deleted from the ISTS Bases is identified by use of the strike-out feature of WordPerfect (or crossed out by hand). Information being added to the ISTS Bases to generate the PI ITS is identified by use of the red-line (shading) feature of WordPerfect (or handwritten/insert pages).

Bracketed Material

Many parameters and portions of Bases are bracketed in the ISTS recognizing that plant specific values and discussions are likely to vary from the "generic" information provided in the standard.

If the bracketed information applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. No change number or justification is provided for use of bracketed material, unless special circumstances warrant discussion.

When bracketed "generic" Bases material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand) and the plant specific information substituted for the bracketed information is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages). A change number with the same format as those used for the ISTS Specification mark-up is provided.

ACRONYMS

Many acronyms are used throughout this submittal. The intent of the final ITS (Part B) is that in general acronyms be written in full prior to the first use. Commonly used acronyms may not be written in full. Other parts of this package may not always write in full each acronym prior to first use; therefore, a list of acronyms is attached to assist in the review of this package.

Attachment to Part A

LIST OF ACRONYMS

AB	Auxiliary Building
ABSVS	Auxiliary Building Special Ventilation System
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ALARA	As Low As Reasonably Achievable
ALT	Actuation Logic Test
ASA	Applicable Safety Analyses
ASME	American Society of Mechanical Engineers
AOO	Anticipated Operational Occurrences
AOT	Allowed Outage Time
BAST	Boric Acid Storage Tank
BIT	Boron Injection Tank
BOC	Beginning of Cycle
CC	Component Cooling
COT	CHANNEL OPERATIONAL TEST
CAOC	Constant Axial Offset Control
CET	Core Exit Thermocouple
CL	Cooling Water
CLB	Current Licensing Basis
COLR	Core Operating Limits Reports
CRDM	Control Rod Drive Mechanism
CRSVS	Control Room Special Ventilation System
CS	Containment Spray
CST	Condensate Storage Tanks
CTS	Current Technical Specification(s)
DBA	Design Basis Accident
DDCL	Diesel Driven Cooling Water
DG	Diesel Generator
DNB	Departure from Nucleate Boiling
DNBR	Departure from nucleate boiling ratio
ECCS	Emergency Core Cooling System

EDG	Emergency Diesel Generators
EFPD	Effective Full Power Days
EOC	End of Cycle
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FWLB	Feedwater Line Break
GDC	General Design Criteria
GITS	Ginna Improved Technical Specifications
HELB	High Energy Line Break
HZP	Hot Zero Power
IPE	Individual Plant Evaluation
ISTS	Improved Standard Technical Specifications
ITC	Isothermal Temperature Coefficient
ITS	Improved Technical Specifications
LA	License Amendment
LAR	License Amendment Request
LBLOCA	Large Break LOCA
LCO	Limiting Conditions for Operation
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Overpressure Protection
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MFW	Main Feedwater
MOSCA	MODE or Other Specified Condition of Applicability
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
MSLB	Main Steam Line Break
MSLI	Main Steam Line Isolation
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
NMC	Nuclear Management Company
NPSH	Net Positive Suction Head

NRCV	Non-Return Check Valve
NUREG-1431	The ISTS for Westinghouse plants
OPPS	OverPressure Protection System
PCT	Peak Cladding Temperature
PI	Prairie Island
PITS	Prairie Island Technical Specifications
PIV	Pressure Isolation Valve
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valve
PTLR	Pressure and Temperature Limits Report
QTPR	Quadrant Power Tilt Ratio
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RPI	Rod Position Indication
RPS	Reactor Protection System
RTB	Reactor Trip Breaker
RTBB	Reactor Trip Bypass Breaker
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SBVS	Shield Building Ventilation System
SCWS	Safeguards Chilled Water System
SDM	Shut Down Margin
SFDP	Safety Function Determination Program
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SL	Safety Limit

SLB	Steam Line Break
SR	Surveillance Requirements
SSC	Structures, Systems and Components
TADOT	Trip Actuating Device Operational Test
TDAFW	Turbine Driven Auxiliary Feedwater
TRM	Technical Requirements Manual
TS	Technical Specifications
TSSC	Technical Specification Selection Criteria
TSTF	Term used for a NUREG change (traveler)
VCT	Volume Control Tank
VFTP	Ventilation Filter Test Program
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
WCAP	Westinghouse technical report

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART B

PROPOSED PRAIRIE ISLAND IMPROVED TECHNICAL SPECIFICATIONS AND BASES

List of Pages

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is within limits.	24 Hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODE 1 and 2.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1	Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Prior to entering MODE 1 after each refueling
SR 3.1.2.2	-----------------------------NOTES----------------------------- 1. Only required to be performed after 60 effective full power days (EFPD). 2. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. --- Verify measured core reactivity is within ± 1% Δk/k of predicted values.	31 EFPD

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Isothermal Temperature Coefficient (ITC)

LCO 3.1.3 The ITC shall be maintained within the limits specified in the COLR. The maximum COLR upper limit shall be:

a. < 5 pcm/°F for power levels ≤ 70% RTP; and

b. < 0 pcm/°F for power levels > 70% RTP.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper ITC limit, MODES 1, 2, and 3 for the lower ITC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ITC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain ITC within limit.	24 hours
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Required Action C.1 must be completed whenever Condition C is entered. --------------------------- Projected end of cycle (EOC) ITC not within lower limit	---------------NOTE---------------- LCO 3.0.4 is not applicable. --- C.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	Once prior to reaching the equivalent of an equilibrium RTP all rods out boron concentration of 300 ppm
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify ITC is within upper limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.3.2	Confirm ITC will be within limits at 70% RTP.	Once after each refueling prior to THERMAL POWER exceeding 70% RTP
SR 3.1.3.3	Confirm that ITC will be within limits at EOC.	Once after each refueling prior to THERMAL POWER exceeding 70% RTP

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits.

LCO 3.1.4 All shutdown and control rods shall be OPERABLE and, individual actual rod positions shall be within 24 steps of their group step counter demand position when the demand position is between 30 and 215 steps, or within 36 steps of their group step counter demand position when the demand position ≤ 30 steps, or ≥ 215 steps.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limits.	B.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2.1.1 Perform SR 3.2.1.1 and SR 3.2.1.2..	2 hours
	<u>AND</u>	
	B.2.1.2 Perform SR 3.2.2.1.	2 hours
	<u>OR</u>	
	B.2.2 Reduce THERMAL POWER to ≤ 85% RTP.	2 hours
	<u>AND</u>	
	B.3 Verify SDM is within the limits provided in the COLR.	Once per 12 hours
	<u>AND</u>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Re-evaluate safety analyses and determine the THERMAL POWER for which the results remain valid for duration of operation under these conditions.	30 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	---------------------------NOTE------------------------- If RPI differs by > 12 steps from the group step counter demand position, enter LCO 3.1.7 to determine RPI OPERABILITY. --- Verify individual rod positions within alignment limit.	12 hours
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.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. Both reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

-----------------------------------NOTE-----------------------------------
This LCO is not applicable while performing SR 3.1.4.2.

APPLICABILITY: MODES 1 and 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
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown banks to within limits.	2 hours

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

---------------------------------------NOTE---------------------------------------
This LCO is not applicable while performing SR 3.1.4.2.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

ACTIONS

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
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore control bank(s) to within limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits provided in the COLR .	1 hour
	OR	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2 Restore control bank sequence and overlap to within limits.	2 hours
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
SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours
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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Rod Position Indication (RPI) System and demand position indication shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RPI per group inoperable for one or more groups.	A.1 Verify the position of the rod(s) with inoperable position indicators by using movable incore detectors.	Once per 8 hours
	<u>OR</u>	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. More than one RPI per group inoperable for one or more groups.	B.1 Monitor and record demand position indication for rods with inoperable RPI.	Once per hour
	<u>AND</u>	
	B.2 Monitor and record reactor coolant system average temperature.	Once per hour
	<u>AND</u>	
	B.3 Verify, using movable incore detectors, position of rods with inoperable RPIs which have been moved in excess of 24 steps in one direction since last determination of their position.	Once per 4 hours
	<u>AND</u>	
	B.4 Restore inoperable RPIs to OPERABLE status such that a maximum of one RPI per group is inoperable.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Indication for one demand position per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all RPIs for the affected bank(s) are OPERABLE.	Once per 8 hours
	AND	
	C.1.2 Verify RPI of the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart.	Once per 8 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify each RPI agrees within 12 steps of the group demand position between 30 and 215 steps, or within 24 steps of the group demand position when the demand position is ≥ 215 steps or ≤ 30 steps.	Once prior to criticality after each removal of the reactor head

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, "Isothermal Temperature Coefficient (ITC)";
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"

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e may be reduced to "3" required channels, provided:

a. RCS lowest loop average temperature is ≥ 535°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

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.	15 minutes
	<u>AND</u>	
	A.2 Suspend PHYSICS TESTS exceptions.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

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
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 535°F.	30 minutes

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.8.4	Verify SDM is within the limits provided in the COLR.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to AEC GDC Criteria 27 and 28 (Ref. 1), two independent reactivity control systems must be provided which are capable of holding the reactor core subcritical from any hot standby or hot operating condition. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

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 control assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperatures are changed to the nominal hot zero power temperature, 547°F.

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 Rod Control 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 Rod Control 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.

BASES

BACKGROUND (continued)

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.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an 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. The primary safety analyses that rely on the SDM limits are the boron dilution and MSLB analyses.

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;

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 $\leq$ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements, at end of cycle (EOC), 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 an RCS cooldown. This results in a reduction of

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 the initial RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB 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 return to power may occur; however, fuel damage as a result of the return to power will not cause offsite doses to exceed the 10 CFR 100 limits.

The most limiting accident at beginning of cycle (BOC) is the boron dilution accident. 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, that is, the time available to operators to stop the dilution event. As the unit changes MODES the volume being diluted may change, i.e., if RHR is in service, as well as the critical boron concentration due to the different temperature ranges. Thus different SDMs may be required for the different modes and dilution flow rates. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

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.

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 in the RCS.

BASES

LCO (continued)

The COLR provides the shutdown margin requirements. The MSLB and the boron dilution accidents (Ref. 2) are the most limiting analyses that establish the SDM requirements in the COLR. For MSLB accidents, if the LCO is violated, there is a potential to exceed 10 CFR 100, "Reactor Site Criteria," limits. 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.

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 $K_{eff} \geq 1.0$, the SDM requirements specified in the COLR are ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

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, and the probability of a design basis accident (DBA) occurring during this time is very low. 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 operator should borate with the best source available for the plant conditions.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, 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.

In MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5, the SDM is verified by comparing the RCS boron concentration to a Shutdown Boron Concentration requirement curve that was generated by taking into account:

a. Required SDM;

b. Shutdown and control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration.

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 comparison.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27 and 28, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to AEC GDC Criteria 27, 28, 29, and 30 (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 SHUTDOWN MARGIN (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.

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, 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

BASES

BACKGROUND (continued)	other variables fixed or stable (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 RATED THERMAL POWER (RTP) and normal operating temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration. 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.
APPLICABLE SAFETY ANALYSES	The acceptance criteria for core reactivity are that the uncertainties in the nuclear design methods are within the expected range and that the calculational models used to generate the safety analyses are adequate. 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

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 early in the cycle (≤ 60 effective full power days (EFPD)) 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 early in the cycle, 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 curve that develop during fuel depletion may be an indication that the calculational model is not adequate, 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 early in the cycle, 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).

BASES (continued)

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 methods 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, 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

BASES

APPLICABILITY (continued)	Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. Verification of measured core reactivity (SR 3.1.2.1) 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

BASES

ACTIONS

A.1 and A.2 (continued)

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.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant 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. 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity must be verified following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The comparison is made when the core conditions such as control rod position, moderator temperature, and samarium concentration are fixed or stable. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at beginning of cycle (BOC).

SR 3.1.2.2

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made,

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.2 (continued)

considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The required Frequency of 31 effective full power days (EFPD) 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. The SR is modified by two Notes. Note 1 states that the SR is only required to be performed after 60 EFPD. Note 2 indicates that the normalization of predicted core reactivity to the measured value may take place within the first 60 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.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27, 28, 29 and 30, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 14.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Isothermal Temperature Coefficient (ITC)

BASES

BACKGROUND

According to AEC GDC Criterion 8 (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 moderator temperature coefficient (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 ITC is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the ITC is the sum of the MTC and fuel temperature coefficient. The ITC is measured directly during low power PHYSICS TEST in order to verify analytical prediction of the MTC. The units of the isothermal temperature coefficient are pcm/°F, where 1 pcm = 1×10^{-5} $\Delta k/k$.

The reactor is designed to operate with a negative ITC 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.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements at beginning of cycle (BOC). Reactor cores are designed so that the BOC ITC is less than zero when THERMAL POWER is at RTP. The actual value of the ITC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed

BASES

BACKGROUND (continued)

distributed poisons to yield an ITC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure that the ITC does not exceed the limits.

The limitations on ITC are provided to ensure that the value of MTC remains within the limiting conditions assumed in the USAR accident and transient analyses.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified ITC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The ITC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The USAR (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 for the cycle exposure being evaluated to ensure that the accident results are bounding.

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive (i.e., upper limit). Such accidents include the rod withdrawal transient from either zero or RTP, 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 the main steam line break.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. 2).

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, ITC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the ITC 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 will remain within the bounds of the original accident analyses during operation. Assumptions made in safety analyses require that the ITC be less positive than a given upper bound and more positive than a given lower bound. The ITC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit usually occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the ITC 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.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance check at BOC on ITC provides confirmation that the ITC is behaving as anticipated and will be within limits at 70% RTP, full power, and EOC so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are

BASES

LCO (continued)

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.

If the LCO limits are not met, the assumptions of the safety analysis may not be met. The core could violate criteria that prohibit a return to criticality, or the DNBR ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

APPLICABILITY

Technical Specifications place both LCO and SR values on ITC, based on the safety analysis assumptions described above.

In MODE 1, the limits on ITC 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 accidents (such as the uncontrolled rod cluster control withdrawal) will not violate the assumptions of the accident analysis. The lower ITC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents at EOC will not violate the assumptions of the accident analysis since ITC becomes more negative as the cycle burnup increases and the RCS boron concentration is reduced. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

ITC must be kept within the upper limit specified in LCO 3.1.3 to ensure that assumptions made in the safety analysis remain valid. The upper limit of Condition A is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO.

BASES

ACTIONS

A.1 (continued)

If the upper ITC limit is violated at BOC, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits in the future. The ITC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the ITC measurement and computing the required bank withdrawal limits.

The control rods are maintained within the administrative withdrawal limits until a subsequent calculation verifies that ITC has been restored with its limit. As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the ITC to become more negative. Using physics calculations, the time in cycle life at which the calculated ITC 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.

BASES

ACTIONS (continued)

C.1

Exceeding the EOC ITC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If it is determined during PHYSICS TESTS that the EOC ITC value will exceed the most negative ITC limit specified in the COLR, the safety analysis and core design must be re-evaluated prior to reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm to ensure that operation near the EOC remains acceptable. The 300 ppm limit is sufficient to prevent EOC operation at or below the accident analysis MTC assumptions.

Condition C has been modified by a NOTE that requires Required Action C.1 to be completed whenever this Condition is entered. This is necessary to ensure that the plant does not operate at conditions where the ITC would be below the most negative limit specified in the COLR.

Required Action C.1 is modified by a Note which states that LCO 3.0.4 is not applicable. This Note is provided since the requirement to re-evaluate the core design and safety analysis prior to reaching an equivalent RTP ARO boron concentration of 300 ppm is adequate action without restricting entry into MODE 1.

D.1

If the re-evaluation of the safety analysis cannot support the predicted EOC ITC lower limit, or if the Required Actions of Condition C are not completed within the associated Completion Time the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 4 within 12 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 systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the ITC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive ITC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC ITC value for ARO will be obtained from measurements during the physics tests after refueling. The ARO value can be directly compared to the BOC ITC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

Measurement of the ITC at the beginning of the fuel cycle is adequate to confirm that the ITC remains within its upper limit.

SR 3.1.3.2

This SR requires measurement of ITC at BOC prior to exceeding 70% RTP after each refueling in order to confirm compliance with the 70% RTP ITC limit. The Frequency of "Once after each refueling prior to THERMAL POWER exceeding 70% RTP" ensures the limit will be met prior to being applicable.

SR 3.1.3.3

This SR requires measurement of ITC at BOC prior to exceeding 70% RTP after each refueling in order to confirm compliance with the most negative ITC LCO. Meeting this limit prior to exceeding 70% RTP ensures that the limit will also be met at EOC.

The ITC value for EOC is derived from the ITC low power PHYSICS TESTS. The EOC value is calculated using the predicted EOC ITC from the core design report and the difference between the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.3 (continued)

measured and predicted BOC ITC. The predicted EOC value is directly compared to the most negative EOC value established in the COLR to ensure that the predicted EOC negative ITC value is within the safety analysis assumptions.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 8, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Sections 14.4 and 14.5.

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.

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC Criteria 6, 14, 27, and 28 (Ref. 1), and 10 CFR 50.46 (Ref. 2).

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 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 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.

BASES

BACKGROUND (continued)

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. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Both units have four control banks and two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, 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 indications, which are the bank demand position indication (usually the group step counters) and the individual Rod Position Indication (RPI) System.

The bank demand position indication 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

BASES

BACKGROUND (continued)

demand position indication 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 RPI System provides a highly reliable indication of rod position, but at a lower accuracy than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The RPI System is designed with an accuracy of ± 5% (approximately 12 steps) of full rod travel. There are inaccuracies arising from the normal range of coolant temperature variation from hot shutdown to full power which are compensated for by allowing ± 24 steps at the lower and upper ends of rod travel.

With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. At the lower and upper ends of rod travel with an indicated deviation of 24 steps between the group step counter and RPI, the deviation between actual rod position and the demand position could be 36 steps, or 22.5 inches.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment assure that:

a. There are no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. 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.

The safety analysis regarding static rod misalignment considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on DNBR in this case bounds the situation when a rod is misaligned from its group by 24 steps.

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 fully withdrawn (Ref. 3).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected, that the linear heat rates (LHRs) are not significantly affected, or that THERMAL POWER will be adjusted so that excessive local LHRs will not occur, and that the requirements on SDM and ejected rod worth are preserved.

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 assumed power distribution used in the safety analysis may not be 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

APPLICABLE SAFETY ANALYSES (continued)

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.

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.

The rod alignment requirements are satisfied when individual actual rod positions are within 24 steps of their group step counter demand position when the demand position is between 30 and 215 steps, or within 36 steps of their group step counter demand position when the demand position is ≤ 30 steps, or ≥ 215 steps.

The requirement to maintain the rod alignment to within plus or minus 12 steps when the group step counter demand position is between 30 and 215 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.

BASES

LCO (continued)	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. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are normally 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 (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.
ACTIONS	<u>A.1.1 and A.1.2</u> 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 boration and restoring SDM. In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

BASES

ACTIONS (continued)

A.2

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.

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.

B.1.1 and B.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

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.

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 unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.1.1, B.2.1.2, B.2.2, B.3, and B.4

For continued operation with a misaligned rod, hot channel factors ($F_Q(Z)$ and ($F^N_{\Delta H}$) must be verified within limits or reactor power

BASES

ACTIONS

B.2.1.1, B.2.1.2, B.2.2, B.3, and B.4 (continued)

must be reduced, SDM must periodically be verified within limits, the safety analyses must be re-evaluated to confirm continued operation is permissible, and, if necessary, the power level must be reduced to a level consistent with the safety analysis. Considerations in these analyses include the potential ejected rod worth and associated transient power distribution peaking factors.

Verifying that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within the required limits (i.e., SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1) ensures that current operation at RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 2 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 $F_{\Delta H}^N$.

In lieu of determining hot channel factors ($F_Q(Z)$ and $F_{\Delta H}^N$) within the Completion Time of 2 hours, reducing power to 85% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

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.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analyses 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 Ref. 3 that may be adversely affected will be

BASES

ACTIONS

B.2.1.1, B.2.1.2, B.2.2, B.3, and B.4 (continued)

evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions.

If the analyses do not support continued operation at RTP, then the power must be reduced to a level consistent with the safety analyses.

A Completion Time of 30 days is sufficient time to obtain the required input data and to perform the analysis and adjust power level.

C.1

When Required Actions cannot be completed within their 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 3 within 6 hours, which eliminates 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 systems.

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 for LCO 3.1.1. The required Completion Time of 1 hour for

BASES

ACTIONS

D.1.1 and D.1.2 (continued)

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 initiate boration. Boration will continue until the required SDM is restored.

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. 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 3 within 6 hours.

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.

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 3.1.4.1 is modified by a Note which direct the operators to Specification 3.1.7, "Rod Position Indication," if a rod appears to be misaligned by more than 12 steps. If the rod position indication is

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 (continued)

determined to be correct in accordance with Specification 3.1.7, then the operator must return to Specification 3.1.4 and enter the appropriate Conditions for rod misalignment.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, 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 providing rod alignment limits are not exceeded. 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.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

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 moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions. Actual rod drop time is measured from opening of the RTB which is conservative with respect to beginning of decay of stationary gripper coil voltage.

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.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 6, 14, 27, and 28, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants".

3. USAR, Section 14.4.

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 define the deepest insertion into the core with respect to core power which is allowed and 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, SHUTDOWN MARGIN (SDM) and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC Criteria 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (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, reactivity limits, and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Some banks 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 that consists of two groups are moved in a staggered fashion, but always within one step of each other. Each reactor has four control banks and two 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

BASES

BACKGROUND (continued)

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 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 (SDM)") 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

power (547°F), 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.

The acceptance criteria for addressing shutdown and control rod bank insertion limits assure that:

a. There are no violations of:

   1. specified acceptable fuel design limits, or

   2. RCS pressure boundary integrity; and

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).

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. 3).

Operation at the insertion limits assures that the maximum linear heat generation rate or peaking factor will be less than that used in the misaligned rod analysis. Operation at the insertion limit also assures that the maximum ejected RCCA worth will be less than the limiting value used in the ejected RCCA analysis.

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).

BASES (continued)

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.

The LCO is modified by a Note indicating that a shutdown bank may be below the insertion limit when required for performance of 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.

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. In MODE 3, 4, 5, or 6, the shutdown bank insertion limit does not apply because the reactor is not producing fission power. In shutdown MODES the OPERABILITY of the shutdown 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. 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.

ACTIONS

A.1.1, A.1.2 and A.2

With one or more shutdown banks not within insertion limits 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.

BASES

ACTIONS

A.1.1, A.1.2 and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take appropriate action because the simultaneous occurrence of either an accident or transient 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 provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, 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 systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours 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.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 27, 28, 29, and 32, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

3. USAR, Sections 14.4 and 14.5.

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 define the deepest insertion into the core with respect to core power which is allowed and 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 SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate. 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.

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation ($K_{eff} \geq 1.0$) to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Some banks 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 that consists of two groups are moved in a staggered fashion, but always within one step of each other. Each reactor has four control banks and two 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.

BASES

BACKGROUND (continued)

Insertion Limits

The control bank insertion limits are specified in a figure in the COLR. The control banks are required to be at or above the insertion limit lines.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. The fully withdrawn position is defined in the COLR. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature.

Overlap and Sequence

The insertion limits Figure in the COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. By overlapping control bank movements, the small reactivity addition at the beginning and end of control bank travel will be compensated for; that is, the overlapping sequential movement of control banks makes the reactivity addition more uniform.

Control banks are moved in an overlap pattern, 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 fully withdrawn position, 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,

BASES

BACKGROUND

Overlap and Sequence (continued)

and C are at the fully withdrawn position, and control bank D is near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. 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.

General

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, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, "Control Bank Insertion Limits," 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 assures fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant will be bounded by the safety analysis results in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other transient requiring termination by a Reactor Trip System (RTS) trip function.

BASES (continued)

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 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 (SDM)") 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 (547°F), and to maintain the required SDM at rated no load temperature (Ref. 3). The control bank insertion limit also limits the reactivity worth of an ejected control rod.

The acceptance criteria for addressing shutdown and control bank insertion limits assure that:

a. There are no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System pressure boundary integrity; and

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).

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. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)	Operation at the insertion limits assures that the maximum linear heat generation rate or peaking factor will be less than that used in the misaligned rod analysis. Operation at the insertion limit also assures that the maximum ejected RCCA worth will be less than the limiting value used in the ejected RCCA analysis. The control bank insertion, sequence and overlap limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis.
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 limited, and ensuring adequate negative reactivity insertion is available on a 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. The LCO is modified by a Note indicating that a control bank may be below the insertion limit when required for performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would normally violate the LCO.
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, and SDM. Applicability in MODE 2 with $K_{eff} < 1.0$, and in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

BASES (continued)

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

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 is normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). 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 an accident or transient 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 limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

BASES

ACTIONS (continued)

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant 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 systems.

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. Prior to achieving criticality, the estimated critical position calculation appropriate for the time at which criticality is achieved shall be verified for control bank position.

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. This verification may be performed manually by an operator or through a computer insertion limit monitoring program.

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

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.6.3 (continued) limit check above in SR 3.1.6.2. This verification may be performed manually by an operator or through a computer sequence and overlap monitoring program.
REFERENCES	1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 27, 28, 29, and 32, issued for comment July 10, 1967, as referenced in USAR Section 1.2. 2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". 3. USAR, Sections 14.4 and 14.5.

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to AEC GDC Criteria 12 and 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.

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 SHUTDOWN MARGIN (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

BASES

BACKGROUND (continued)

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 positions of shutdown rods and control rods are determined by two separate and independent systems: the bank demand position indication (commonly called group step counters) and the individual Rod Position Indication (RPI) System.

The bank demand position indication counts the pulses from the Rod Control System that move 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 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 RPI System provides a highly reliable indication of actual control rod position, but at a lower accuracy than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. The RPI System is designed with an accuracy of ± 5 % (approximately 12 steps) of full rod travel. There are inaccuracies arising from the normal range of coolant temperature variation from hot shutdown to full power which are compensated for by allowing ± 24 steps at the lower and upper ends of rod travel. With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. At the lower and upper ends of rod travel with an indicated deviation of 24 steps between the group counter and RPI, the deviation between actual rod position and the demand position could be 36 steps, or 22.5 inches.

BASES (continued)

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.

The control rod position indicator channels satisfy 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.

LCO

LCO 3.1.7 specifies that the RPI System and bank demand position indication be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires the following:

a. The RPI System indicates within 12 steps of the group step counter demand position when the demand position is between 30 and 215 steps, or within 24 steps of their group step counter demand position when the demand position is greater than or equal to 215 steps, or less than or equal to 30 steps;

b. Bank demand indication has been calibrated either in the fully inserted position or to the RPI System. Demand position indication may be provided by various means such as step counters, Emergency Response Computer System (ERCS), calculations using rod drive cabinet counters or Pulse to Analog counters.

BASES

LCO (continued)

The 12 step agreement limit between bank demand position indication and the RPI System when the demand position is between 30 and 215 steps indicates that the bank demand position indication is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given above, 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.

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 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. 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. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

BASES (continued)

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable RPI 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

When one RPI channel per group fails, the position of the rod may still be determined indirectly by use of the moveable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. 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. Verification may determine that the RPI is OPERABLE and the rod is misaligned, then the Conditions of 3.1.4, "Rod Group Alignment Limits" must be entered.

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.

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.

BASES

ACTIONS (continued)

B.1, B.2, B.3, and B.4

When more than one RPI channel per group fails, additional monitoring shall be performed to assure that the reactor remains in a safe condition. The demand position from the group step counters associated with the rods with inoperable position indicators shall be monitored and recorded on an hourly basis. This ensures a periodic assessment of rod position to determine if rod movement in excess of 24 steps has occurred since the last determination of rod position. If rod movement in excess of 24 steps has occurred since the last determination of rod position, the Required Action of B.3 below is required.

The reactor coolant system average temperature shall be monitored and recorded on an hourly basis. Monitoring and recording of the reactor coolant system average temperature may provide early detection of mispositioned or dropped rods.

When one or more rods have been moved in excess of 24 steps in one direction, since the position was last determined, action is initiated sooner to begin verifying that these rods are still properly positioned relative to their group positions. The four hour allowance for completion of this action allows adequate time to complete the rod position verification using the moveable incore detectors.

The position of rods with inoperable RPIs will also continue to be verified indirectly using the moveable incore detectors every 8 hours in accordance with Required Action A.1. Using the moveable incore detectors provides further assurance that the rods have not moved.

Based on experience, normal power operation does not require excessive movement of banks. Therefore, the actions specified in this condition are adequate for continued full plant operation for up to 24 hours 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 allowed out of service time also

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

provides sufficient time to troubleshoot and restore the RPI system to operation following a component failure in the system, while avoiding the challenges associated with a plant shutdown.

C.1.1 and C.1.2

Demand position indication is provided by any of the following means: step counters; Emergency Response Computer System (ERCS); calculations using rod drive cabinet counters and Pulse to Analog counters. With all indication for one demand position per bank inoperable, the rod positions can be determined by the RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the RPI of the most withdrawn rod and the RPI of the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This ensures that the most withdrawn and least withdrawn rod are no more than 24 steps apart (including instrument uncertainty) which bounds the accident analysis assumptions. This verification can be an examination of logs, administrative controls, or other information that shows that all RPIs in the affected bank are OPERABLE.

C.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. 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.

BASES

ACTIONS (continued)

D.1

If the Required Actions 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 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 systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the RPI agrees with the demand position within 12 steps (between 30 and 215 steps) or within 24 steps (when $\leq$ 30 steps or $\geq$ 215 steps) ensures that the RPI is operating correctly.

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.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 12 and 13, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Sections 14.4 and 14.5.

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, 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.

The key objectives of a test program are to:

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.

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. 1).

BASES

BACKGROUND (continued)

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.

The PHYSICS TESTS required for reload fuel cycles (Ref. 1) in MODE 2 are listed below:

a. Critical Boron Concentration-Control Rods Withdrawn;

b. Critical Boron Concentration-Control Rods Inserted;

c. Control Rod Worth; and

d. Isothermal Temperature Coefficient (ITC).

Low power physics tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration-Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, bank D 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 could violate LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)."

b. The Critical Boron Concentration-Control Rods Inserted Test measures the critical boron concentration at HZP, with the highest worth rod bank fully inserted into the core. This test is used to give an indication of the boron reactivity coefficient.

BASES

BACKGROUND (continued)

With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered. 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 gives an indication of the Boron Reactivity Coefficient compared to the measured bank worth. Performance of this test could violate 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."

c. The Control Rod 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 Dilution 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 Dilution 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

BASES

BACKGROUND (continued)

worth of the selected control bank. 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.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP using the Slope 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 ITC at BOC, 70% RTP and at EOC is determined from the ITC measured in this test. This test satisfies the requirements of SR 3.1.3.1, SR 3.1.3.2 and SR 3.1.3.3. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The above mentioned PHYSICS TESTS may require the operating control or process variables to deviate from their LCO limitations.

The USAR defines requirements for initial testing of the facility, including PHYSICS TESTS. USAR Appendix J summarizes the initial plant startup zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in Reference 1. 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. The requirements specified in the following LCOs may be suspended for PHYSICS TESTING:

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)";
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".

When these LCOs 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 ≥ 535°F, and SDM is within the limits provided in the COLR.

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 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.

LCO

This LCO allows the reactor parameters of ITC and minimum temperature for criticality to be outside their specified limits to conduct PHYSICS TESTS in MODE 2, to verify certain core physics parameters. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from "4" to "3". 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.

BASES

LCO (continued)

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 and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 535°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

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 maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

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 conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour.

BASES

ACTIONS (continued)

B.1

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 < 535°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 to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 535°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If Required Action C.1 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 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 systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1 (continued)

TEST is 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.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is $\geq$ 535°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

Verification that the THERMAL POWER is $\leq$ 5% RTP will ensure that the plant 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.

SR 3.1.8.4

Prior to achieving criticality, the SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.4 (continued)

b. Control and shutdown bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration.

After achieving criticality, this SR is met by determining the reactivity insertion available from tripping the shutdown and control banks.

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. ANSI/ANS-19.6.1-1985, Reload Startup Physics Tests for Pressurized Water Reactors.

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART C

MARKUP OF PRAIRIE ISLAND CURRENT TECHNICAL SPECIFICATIONS

List of Pages

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications Conversion Submittal

3.1.F. ISOTHERMAL TEMPERATURE COEFFICIENT (ITC)

LCO3.1.3

1. When the reactor is in MODE 1 and 2 with $K_{eff} \geq 1.0$ ~~critical~~, the isothermal temperature coefficient shall be within the limits specified in the COLR, the maximum upper limit shall be A3.1-01 A3.1-02

less than 5 pcm/°F ~~with all rods withdrawn~~, except ~~during low power PHYSICS TESTS and~~ as specified in 3.1.F.2 and 3. LR3.1-03

When the reactor is in MODES 1, 2 and 3, the lower ITC limit shall be met. M3.1-04

LCO3.1.3

2. When the reactor is above 70 percent RATED THERMAL POWER ~~with all rods withdrawn~~, the isothermal temperature coefficient shall be negative, except as specified in 3.1.F.3. LR3.1-03

3. If the limits of 3.1.F.1 or 2 cannot be met, POWER OPERATION may continue provided the following actions are taken:

LCO3.1.3
Action A
Action B

a. Establish and maintain control rod withdrawal limits sufficient to restore the ITC to less than the upper limits specified in A3.1-02

the COLR ~~Specification 3.1.F.1 and 2 above~~ within 24 hours or be in MODE 2 with $K_{eff} < 1.0$ ~~HOT SHUTDOWN~~ within the next 6 hours. A3.1-01

~~These withdrawal limits shall be in addition to the insertion limits specified in the CORE OPERATING LIMITS REPORT.~~ A3.1-06

~~b. Maintain the control rods within the withdrawal limits established above until a subsequent calculation verifies that the ITC has been restored to within its limit for the all rods withdrawn condition.~~ LR3.1-07

A3.1-08

~~c. Submit a special report to the Commission within 30 days, describing the value of the measured ITC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the ITC to within its limit for the all rods withdrawn condition.~~

LCO3.1.3
Action C
Action D

If the projected EOC ITC is not within the lower limit, prior to reaching the equivalent of an equilibrium RTP all rods out boron concentration 300 ppm, re-evaluate the core design and safety analysis and determine that the reactor core is acceptable for continued operation. If this action or completion time are not met, be in MODE 4 within 12 hours. M3.1-09

SR3.1.3.1
SR3.1.3.2
SR3.1.3.3

M3.1-11

New SRs, Verify ITC within upper limit, confirm ITC will be within limits at 70% power, and confirm that ITC is within limits at EOC.

3.10 ~~CONTROL ROD AND POWER DISTRIBUTION LIMITS~~

~~Applicability~~

A3.1-12

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

~~Objective~~

~~To assure 1) core subcriticality after reactor trip, 2) acceptable core power distributions during POWER OPERATION, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.~~

Specification

A. Shutdown Margin

LCO3.1.1

1. The SHUTDOWN MARGIN shall be maintained within the limits specified in the Core Operating Limits Report when in MODES 2 with $K_{eff} < 1.0$, 3, 4 and 5 ~~HOT SHUTDOWN, INTERMEDIATE SHUTDOWN and COLD SHUTDOWN.~~

A3.1-01

LCO3.1.1

2. With the SHUTDOWN MARGIN less than the applicable limit specified in 3.10.A.1 above, within 15 minutes initiate boration to restore SHUTDOWN MARGIN to within the applicable limit.

SR3.1.1.1

New SR, Verify SDM is within limits.

M3.1-17

LCO3.1.2 Action B

New Action Statement, Be in Mode 3 within 6 hours if core re-evaluation or operating restrictions are not provided in 7 days.

M3.1-18

SR3.1.2.1

New SR, Verify measured core reactivity is within ± 1%ΔK/K of predicted values prior to entering MODE 1 after each refueling.

M3.1-19

Addressed Elsewhere

B. Power Distribution Limits

1. At all times, except during low power PHYSICS TESTING, measured hot channel factors, F^N_Q and $F^N_{\Delta H}$, as defined below and in the bases, shall meet the following limits:

$F^N_Q \times 1.03 \times 1.05* \leq (F_Q^{RTP} / P) \times K(Z)$

$F^N_{\Delta H} \times 1.04** \leq F_{\Delta H}^{RTP} \times [1 + PFDH(1-P)]$

Where the following definitions apply:

Addressed Elsewhere

3.10.C.2 If the QUADRANT POWER TILT RATIO exceeds 1.02 but is less than 1.07 for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for PHYSICS TESTS if the QUADRANT POWER TILT RATIO exceeds 1.07, the reactor shall be brought to the HOT SHUTDOWN condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel inoperable, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

LCO3.1.5 1. The shutdown banks ~~rods~~ shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT when the reactor is in MODES 1 and 2 ~~critical or approaching criticality~~. A3.1-01

LCO3.1.5 New Action Statements, A and B, if limits are not met, verify SDM within limits or initiate boration, restore shutdown banks to limits in 2 hours, or be in MODE 3 in next 6 hours. L3.1-21

SR3.1.5.1 New SR, verify shutdown banks are within the limits specified in the COLR. M3.1-22

LCO3.1.6 2. When the reactor is in MODES 1 and 2 with $K_{eff} \geq 1.0$ ~~critical or approaching criticality~~, the control banks L3.1-23

shall be limited in physical insertion, sequence and overlap limits as specified in the CORE OPERATING LIMITS REPORT. M3.1-24

LCO3.1.6 New Action Statements, A, B and C, if limits are not met, within 1 hour verify SDM within limits or initiate boration, restore control banks to limits in 2 hours, or be in MODE 2 with $K_{eff} < 1.0$ in next 6 hours. L3.1-26

SR3.1.6.1
SR3.1.6.2
SR3.1.6.3

New SRs, verify estimated critical control bank position is within limits in the COLR, verify control bank insertion is within limits in the COLR, verify sequence and overlap limits are met. M3.1-27

LCO3.1.8
LCO3.1.5
LCO Note
LCO3.1.6
LCO Note

3. Insertion limits do not apply during PHYSICS TESTS provided the PHYSICS TEST requirements below are met or during periodic exercise of individual rods. L3.1-28

LCO3.1.8

ITC, Rod Group Alignment Limits, and RCS Minimum Temperature for Criticality are not required to be met during PHYSICS TESTING provided: L3.1-28

RCS lowest loop average temperature is ≥ 535 °F, SDM is within limits provided in the COLR and THERMAL POWER is ≤ 5% RTP. Associated Action Statements are also provided.

SR3.1.8.1
SR3.1.8.2
SR3.1.8.3
SR3.1.8.4

New SRs, perform COT on power range and intermediate range channels, verify RCS lowest loop average temperature, verify THERMAL POWER, verify SDM is within limits. M3.1-29

~~The shutdown margin specified in the Core Operating Limits Report must be maintained except for low power PHYSICS TESTING. For this test the reactor may be critical with all but one high worth full-length control rod inserted for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power PHYSICS TEST.~~ M3.1-31

3.10.E. Rod Misalignment Limitations

LCO3.1.4 Action B

1. If a rod cluster control assembly (RCCA) is misaligned from its bank by more than 24 steps, the rod will be realigned or, verify SDM is within limits or initiate boration, and the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B (ITS SR 3.2.1.1 and SR 3.2.2.1) applied. If peaking factors are not determined within 2 hours, the THERMAL POWER ~~high neutron flux trip setpoint~~ shall be reduced to 85 percent of rating. M3.1-32 L3.1-33

~~2. If the misaligned RCCA is not realigned within a total of 8 hours, the RCCA shall be declared inoperable.~~ A3.1-34

3.10.F. Rod Position Indication System

LCO3.1.7 Actions Note

Note: Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. A3.1-36

1. In MODE 1 and 2 each channel of the Rod Position Indication System shall be OPERABLE, ~~capable of determining the control rod positions within the following (except as specified in 3.10.F.2 or 3.10.F.3 below):~~ M3.1-38 LR3.1-37

   ~~a. With bank demand position greater than or equal to 215 steps, or less than or equal to 30 steps, the difference between the individual rod position indication and the demand position for the corresponding group step counter shall be no greater than ± 24 steps, or~~

   ~~b. With bank demand position between 30 and 215 steps, the difference between the individual rod position indication and the demand position for the corresponding group step counter shall be no greater than ± 12 steps.~~

2. In MODE 1 and 2 with one rod position indicator per group inoperable for one or more groups either: M3.1-38

   LCO3.1.7 Action A

   a. Verify the position of rod(s) with inoperable position indicator(s) indirectly using the moveable incore detectors at least once per 8 hours, or

   LCO3.1.7 Action A Action C

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

3. In MODE 1 and 2 with more than one rod position indicator per group inoperable for one or more groups: M3.1-38

   LCO3.1.7 Action B

   a. Verify the position of rods with inoperable position indicators indirectly using the moveable incore detectors at least once per 8 hours, and

   LCO3.1.7 Action B

   b. Verify the position of rods with inoperable position indicators indirectly using the moveable incore detectors within 4 hours after rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of their position, and

   LCO3.1.7 Action B

   c. Monitor and record the demand position for the corresponding group step counters for rods with inoperable position indicators at least once per hour, and

   LCO3.1.7 Action B

   d. Monitor and record reactor coolant system average temperature at least once per hour, and

LCO3.1.7 Action B

e. Restore inoperable position indicators to OPERABLE status within 24 hours such that a maximum of one rod position indicator per group is inoperable.

LCO3.1.7 Action D

4. If the requirements of Specification 3.10.F.3 cannot be met, then place the affected unit in at least HOT SHUTDOWN within the following 6 hours.

LCO3.1.4

5. If a control rod with an inoperable rod position indicator is found to be misaligned during the verification of rod position required by Specifications 3.10.F.2.a, 3.10.F.3.a or 3.10.F.3.b above, then apply the requirements of Specification 3.10.E.

LCO3.1.7 Action C

Demand position indication shall be operable. If all demand position indication per bank is inoperable for one or more banks, within 8 hours verify administratively all RPIs for the affected bank(s) are operable and verify the most withdrawn and least withdrawn rod of the affected bank(s) are ≤ 12 steps apart, OR reduce power to ≤ 50% RTP. M3.1-42

3.10.G. Control Rod Operability Limitations

LR3.1-43

1. ~~An inoperable rod is a rod which (a) does not trip, (b) cannot be moved as a result of excessive friction or mechanical interference, or (c) is declared inoperable under specification 3.10.E or 3.10.H.~~

LCO3.1.4 Action A

2. The reactor shall be brought to the HOT SHUTDOWN condition within 6 hours should one or more ~~than one~~ inoperable rod(s) be discovered during POWER OPERATION. M3.1-44

LCO3.1.4 Action A

3. If a rod is ~~the~~ inoperable ~~rod is located below the 200 step level and is capable of being tripped, or if the rod is located below the 30 step level whether or not it is capable of being tripped~~, then the insertion limits (SDM limits) specified in the CORE OPERATING LIMITS REPORT shall be met within one hour or initiate boration ~~apply~~. M3.1-47

LCO3.1.4 Action A

4. If a rod is ~~the~~ inoperable ~~rod cannot be located, or if the inoperable rod is located above the 30 step level and cannot be tripped~~, then the insertion limits (SDM limits) specified in the CORE OPERATING LIMITS REPORT shall be met within one hour or initiate boration ~~apply~~. M3.1-47

A3.1-48

LCO3.1.4 Action B

5. If POWER OPERATION is continued with one ~~inoperable~~ rod not within alignment limits,

LCO3.1.4 Action B

re-evaluate the safety analyses ~~the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis~~ within 30 days unless the rod is earlier made OPERABLE. ~~The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod~~. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, THERMAL POWER shall be reduced to a level consistent with the safety analysis. LR3.1-51

LCO3.1.4 Action B

6. With one or more rod(s) trippable, but immovable due to an electrical problem in the rod control system, verify the rod is within alignment ~~within one hour verify that control rod position is within the rod insertion limits specified in section 3.10.D~~. If the rod is not in alignment, then apply LCO 3.1.4 Action B ~~Restore the Rod Control System to OPERABLE status within 72 hours or declare the affected rod(s) inoperable and apply the limitations specified in sections 3.10.G.2 through 3.10.G.5~~. A3.1-52

~~TS.3.10-7 Overflow~~

LCO3.1.4 Action C	New Action Statement, if Condition B requirements are not met, be in MODE 3 in 6 hours.	M3.1-53
LCO3.1.4 Action D	Action Statement, if more than one rod not within alignment requirements are not met, be in MODE 3 in 6 hours.	A3.1-54

H. Rod Drop Time

SR3.1.4.3	At operating temperature (T_{ave} > 500°F) and full flow (both RCPs operating), the drop time of each RCCA shall be no greater than 1.8 seconds from beginning of decay~~loss~~ of stationary gripper coil voltage to dashpot entry. If the time is greater than 1.8 seconds, the rod shall be declared inoperable.	A3.1-57

3.10.I. Monitor Inoperability Requirements

1. ~~If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of RATED THERMAL POWER, and after 30 inches or more of rod motion.~~

LR3.1-59

Addressed Elsewhere

2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of RATED THERMAL POWER in addition to the increased surveillance requirements.

3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of RATED THERMAL POWER.

J. DNB Parameters

The following DNB related parameters limits shall be maintained during POWER OPERATION:

a. Reactor Coolant System Tavg	≤564°F
b. Pressurizer Pressure	≥2220 psia*
c. Reactor Coolant Flow	≥the value specified in the CORE OPERATING LIMITS REPORT

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours. Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

*Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

4.9 Core REACTIVITY ~~ANOMALIES~~

Applicability

~~Applies to potential reactivity anomalies.~~ A3.1-12

Objective

~~To require evaluation of reactivity anomalies within the reactor.~~

Specification

SR3.1.2.2 Note 2: ~~Following a~~ normalization of the computed boron concentration as a function of burnup may be performed prior to 60 EFPD;~~,~~ T~~t~~he actual boron concentration of the coolant shall be compared monthly with the predicted value. Note 1: SR only required to be performed after 60 EFPD. M3.1-62

LCO3.1.2 Action A If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, Re-evaluate the core design and establish operating restrictions and SRs ~~submit a special report to the Commission~~ within 7~~30~~ days. M3.1-18

TABLE TS.4.1-1C (Page 1 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

LR3.1-59

SR3.1.7.1

SR3.1.4.1

L3.1-63

LR3.1-59

Addressed Elsewhere

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~1. Control Rod Insertion Monitor~~	~~M~~	~~R~~	~~S/U(30)~~	~~N.A.~~	~~1, 2~~
2. Analog Rod Position	S	R	S/U(30)	N.A.	1, 2, 3(31)~~, 4(31), 5(31)~~
~~3. Rod Position Deviation Monitor~~	~~M~~	~~N.A.~~	~~S/U(30)~~	~~N.A.~~	~~1, 2~~
~~4. Rod Position Bank Counters~~	S(32)	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3(31), 4(31), 5(31)~~
5. Charging Flow	S	R	N.A.	N.A.	1, 2, 3, 4
6. Residual Heat Removal Pump Flow	S	R	N.A.	N.A.	4(37), 5(37), 6(37)
7. Boric Acid Tank Level	D	R(33)	M(33)	N.A.	1, 2, 3, 4
8. Refueling Water Storage Tank Level	W	R	M	N.A.	1, 2, 3, 4
9. Volume Control Tank Level	S	R	N.A.	N.A.	1, 2, 3, 4
10. Annulus Pressure (Vacuum Breaker)	N.A.	R	R	N.A.	See Note (39)
11. Auto Load Sequencers	N.A.	N.A.	M	N.A.	1, 2, 3, 4
12. Boric Acid Make-up Flow Channel	N.A.	R	N.A.	N.A.	1, 2, 3, 4

~~TABLE TS.4.1-1C (Page 1 of 4) REV 111 8/10/94~~

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
W	Weekly
M	Monthly
Q	Quarterly
S/U	Prior to each reactor startup
Y	Yearly
R	Each Refueling Shutdown
N.A.	Not applicable.

TABLE NOTATION

L3.1-64

SR3.1.7.1

(30) Prior to criticality after each removal of the reactor head~~each startup following shutdown in excess of two days if not done in previous 30 days.~~

L3.1-63

(31) ~~When the reactor trip system breakers are closed and the control rod drive system is capable of rod withdrawal.~~

(32) ~~Following rod motion in excess of six inches when the computer is out of service.~~ LR3.1-59

(33) Transfer logic to Refueling Water Storage Tank

(34) When either main steam isolation valve is open.

(35) Includes those instruments named in the emergency procedure.

(36) Except for containment hydrogen monitors and refueling water storage tank level which are separately specified in this table.

(37) When RHR is in operation.

(38) When the reactor coolant system average temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

(39) Whenever CONTAINMENT INTEGRITY is required.

Addressed Elsewhere

~~TABLE TS.4.1-1C~~
~~(Page 4 of 4)~~
~~REV 135 5/4/98~~

Table TS.4.1-2A (Page 1 of 2)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

Equipment	Test	Frequency	FSAR Sect. Reference
SR3.1.4.3			7
1 Control Rod Assemblies	Rod Drop Times of full length rods	~~All rods during each refueling shutdown or~~ Prior to criticality following each removal of the reactor vessel head; ~~affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods~~	LR3.1-65
SR3.1.4.2			
2. Control Rod Assemblies	Partial movement of all rods not fully inserted ≥ 10 steps	Every Quarter	7 M3.1-66
		Addressed Elsewhere	
3. Pressurizer Safety Valves	Verify OPERABLE in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Verify each required lift setpoint in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	-
5. Reactor Cavity	Water Level	Prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded	
6. Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1).(b).2 or 3.1.A.2.c.(1).(b).3	-
7. Pressurizer PORVs	Functional	Every 18 months	-

~~TABLE TS.4.1-2A (Page 1 of 2) REV 123 5/21/96~~

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART D

DISCUSSION OF CHANGES
(DOC)

to

PRAIRIE ISLAND
CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART D

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

DISCUSSION OF CHANGES TO CURRENT TECHNICAL SPECIFICATIONS

The proposed changes to PI Operating License Appendix A, TS are discussed below and the specific wording changes are shown in Parts B, C and E.

For ease of review, all package parts and discussions are organized according to the proposed PI ITS Table of Contents.

NSHD category	Change number 3.1-	Discussion Of Change
A	01	CTS 3.1.F.1, 3.1.F.3.a, 3.10.A.1, 3.10.A.2 and 3.10.D.1. The CTS contains prose descriptions of the Modes for which the specification is applicable. This description has been replaced with the equivalent MODES of applicability for the ITS. Since the plant Modes to which this specification apply have not changed, this is an administrative change.
A	02	CTS 3.1.F.1 and 3.1.F.3.a. The CTS contains specific maximum upper limit requirements for ITC which have been included in the LCO. This change will require new ITS limits required by the ITS to be located in the COLR. This change is consistent with the guidance of NUREG-1431. Since no new limits are added to the ITS and the CTS limits are stated in the ITS, this is an administrative change.

NSHD category	Change number 3.1-	Discussion Of Change
LR	03	CTS 3.1.F.1 and 3.1.F.2. The term, "with all rods withdrawn" is relocated to the COLR which will define the conditions under which the specific limits apply. This change is consistent with the guidance of NUREG-1431. This change is acceptable since the COLR is governed by the requirements of the Administrative Controls Section 5.6. Since changes to the ITC may be made within the COLR without prior NRC approval, this change is less restrictive.
M	04	CTS 3.1.F.1. In conformance with the guidance of NUREG-1431, the LCO requires the lower limit for ITC to be met. Since the CTS does not explicitly state that the lower limit for the ITC is required to be met, this is a more restrictive change. This change is acceptable and does not cause an unsafe condition because the plant currently assures that the lower ITC limits are met. This change is included to make the PI ITS complete.
	05	Not used.
A	06	3.1.F.3.a. The limits contained in CTS are repeated in the LCO for ITS 3.1.3 in accordance with the guidance in NUREG-1431. Therefore, this statement is not meaningful or necessary and is not included in the ITS. Since this change does not affect any operating limits or conditions, this is an administrative change.

NSHD category	Change number 3.1-	Discussion Of Change
LR	07	CTS 3.1.F.3.b. Action Statements are provided in accordance with the guidance of NUREG-1431 which, along with the Bases and Use and Application section, provide the necessary guidance for complying with conditions which deviate from the LCO. The details associated with the method of establishing compliance with the limit are not necessary to ensure restoration is accomplished in a timely manner and are not required to be in the TS to provide adequate protection of the public health and safety. Thus the guidance provided by this statement is relocated to the Bases. Since the ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) are licensee controlled and can be revised without prior NRC approval, this change is less restrictive.

NSHD category	Change number 3.1-	Discussion Of Change
A	08	CTS 3.1.F.c. ITS Specification 3.1.3, in conformance with NUREG-1431, requires establishment of administrative withdrawal limits for control banks to maintain ITC within limits. Once these limits are established, the plant is in a safe operating configuration and further action is not necessary. ITS LCO 3.1.3 Required Actions do not require special reporting. This change is acceptable because the special reporting requirements of CTS 3.1.F.3.c are not necessary to assure operation in a safe manner. In development of NUREG-1431, TS reporting requirements that are redundant to regulations have been deleted from the TS. The NRC modified 10CFR50.72 and 10CFR50.73 to more clearly identify which plant conditions need to be reported to the NRC. These regulations currently would require a report if the TS are violated or if the condition is outside accident analysis. If a reactivity anomaly is identified which meets these conditions, a report to the NRC would be required by these revised regulations. Thus the CTS requirement to submit a report to the NRC within 30 days is not necessary to assure that a report is submitted to the NRC and thus is not included in the ITS. Since reporting of safety significant conditions is still required, this is an administrative change.
M	09	New Action Statements are included which are consistent with the guidance of NUREG-1431. These action statements provide requirements for the conditions if the projected EOC ITC is not within the lower limit. Since this change provides additional limitations on plant operation, this is a more restrictive change. This change is acceptable because it assures that plant operations maintain the reactor core in a safe operating configuration. This change is included to make the PI ITS complete.

NSHD category	Change number 3.1-	Discussion Of Change
	10	Not used.
M	11	New SRs, 3.1.3.1, 3.1.3.2, and 3.1.3.3 are included to verify that ITC is within the upper limit, confirm it will be within limits at 70% power, and confirm that it wil be within the limits at EOC. This change is consistent with the guidance of NUREG-1431. Since these SRs are new requirements, this change is more restrictive. This change is acceptable since the act of performing these SRs does not impact normal plant operations. This change is included to make the PI ITS complete.
A	12	CTS 3.10 and 4.9. The beginning of each CTS section contains general statements of Applicability and Objectives for that TS section which are not included in the ITS. This Applicability states the plant design features or systems to which the specifications apply which is a different meaning than the Applicability in NUREG-1431. Since the ITS clearly states within each specification, the plant design features or systems to which it applies, administratively these statements have been incorporated. Likewise, the CTS Objectives statement provides an overall purpose for the specifications within the section. These objectives are administratively incorporated in general through the statement of the ITS specification LCO and the supporting Bases. Since these general CTS statements do not establish any regulatory requirements and are incorporated in a broad sense in the ITS, these are considered administrative changes.
	13	Not used.

NSHD category	Change number 3.1-	Discussion Of Change
	14	Not used.
	15	Not used.
	16	Not used.
M	17	A new SR, 3.1.1.1 is included to verify that SDM is within limits. This change is consistent with the guidance of NUREG-1431. Since this SR is a new requirement, this change is more restrictive. This change is acceptable since the act of performing this SR does not impact normal plant operations. This change is included to make the PI ITS complete.
M	18	CTS 3.10.A.3 and 4.9. CTS 4.9 requires a special report to the NRC within 30 days with no further required actions when the core reactivity differs from the predicted value. Under the provisions of the ITS which is consistent with the guidance of NUREG-1431, core re-evaluation and operating restrictions are required to be prepared within 7 days. If these actions are not completed within 7 days, then this new Action Statement requires the plant to be in MODE 2 within 6 hours. Thus, this change is more restrictive. This change is acceptable since it assures that the plant is maintained in a safe condition if core reactivity is not within ± 1% $\Delta k/k$. This change is included to make the PI ITS complete.

NSHD category	Change number 3.1-	Discussion Of Change
M	19	A new SR, 3.1.2.1 is included to verify that core reactivity is within limits. This change is consistent with the guidance of NUREG-1431. Since this SR is a new requirement, this change is more restrictive. This change is acceptable since the act of performing this SR does not impact normal plant operations. This change is included to make the PI ITS complete.
	20	Not used.
L	21	New Action Statements are included which provide remedial actions if the rod insertion limits are not met in MODES 1 and 2. CTS do not provide Action Statements for this condition, thus the plant would be required to enter CTS 3.0.C (ITS 3.0.3) which would require the plant to be in MODE 3 in 6 hours and MODE 5 in 36 hours. Since the new ITS Action Statements only require the plant to go to MODE 3 in 6 hours, this change is less restrictive. This change is acceptable since the control rods are fully inserted when the plant is in MODE 3 and further shutdown to limit the effect of not meeting insertion limits is unnecessary. The plant is maintained in a safe condition in MODES 3, 4 and 5 due to boration which provides the required SDM . Also rod insertion is not assumed in any analyses in these modes. This change is consistent with the guidance of NUREG-1431.

NSHD category	Change number 3.1-	Discussion Of Change
M	22	A new SR, 3.1.5.1 is included to verify that shutdown banks are within their insertion limits. This change is consistent with the guidance of NUREG-1431. Since this SR is a new requirement, this change is more restrictive. This change is acceptable since the act of performing this SR does not impact normal plant operations. This change is included to make the PI ITS complete.
L	23	CTS 3.10.D.2. The CTS contains prose descriptions of the Modes for which the specification is applicable. In conformance with the guidance of NUREG-1431, this description has been replaced with MODES 1 and 2 with $K_{eff} \geq 1.0$. This applicability is nearly the same as CTS except that currently, control bank insertion limits apply when "approaching criticality". Control banks are required to meet insertion limits when at power (MODES 1 and 2 with $K_{eff} \geq 1.0$) to preserve the assumed power distribution, ejected rod worth, SDM, and reactivity insertion rate assumptions. Control bank insertion limits are not required during the period of approach to criticality since the shutdown banks provide adequate protection for shutdown margin, ejected rods and reactivity insertion rate. When approaching criticality, there is no assumed power distribution. Thus, this change is acceptable. This change is consistent with the guidance of NUREG-1431.

NSHD category	Change number 3.1-	Discussion Of Change
M	24	CTS 3.10.D.2. In conformance with the guidance of NUREG-1431, control bank sequence and overlap limits shall be met as a TS requirement. CTS only require the insertion limits to be met. This change is acceptable since current plant practice requires the sequence and overlap limits to be met and no new plant operating restrictions are imposed. However, since compliance with these limits is now a TS requirement, this is a more restrictive change. This change is included to make the PI ITS complete.
	25	Not used.
L	26	New Action Statements are included which provide remedial actions if the rod insertion limits are not met in MODES 1 and 2 with $K_{eff} \geq 1.0$. CTS do not provide Action Statements for this condition, thus the plant would be required to enter CTS 3.0.C (ITS 3.0.3) which would require the plant to be in MODE 3 in 6 hours and MODE 5 in 36 hours. Since the new ITS Action Statements only require the plant to go to MODE 2 with $K_{eff} < 1.0$ in 6 hours, this change is less restrictive. This change is acceptable since the control rods are fully inserted when the plant is in MODE 2 with Keff < 1.0 and further shutdown to limit the effect of not meeting insertion limits is unnecessary. The plant is maintained in a safe condition in MODE 2 with $K_{eff} \leq 1.0$, and in MODES 3, 4 and 5 due to boration which provides the required SDM . Also rod insertion is not assumed in any analyses in these modes. This change is consistent with the guidance of NUREG-1431.

NSHD category	Change number 3.1-	Discussion Of Change
M	27	New SRs, 3.1.6.1, 3.1.6.2 and 3.1.6.3 are included to verify estimated critical control bank position is within limits in the COLR, verify control banks are within their insertion limits and verify sequence and overlap limits are met. These new SRs are consistent with the guidance of NUREG-1431. Since these SRs are new requirements, these changes are more restrictive. These changes are acceptable since the act of performing these SRs does not impact normal plant operations. This change is included to make the PI ITS complete.
L	28	CTS 3.10.D.3. In conformance with the guidance of NUREG-1431, the ITS provides exceptions from the specification requirements for ITC, Rod Group Alignment, and RCS Minimum Temperature for Criticality when Physics Tests are performed. To assure plant safety, additional restrictions are placed on RCS lowest loop average temperature, SDM and Thermal Power. These exceptions and additional restrictions are not included in the CTS. Since this change includes both new restrictions and new exceptions, this change is considered less restrictive. This change is acceptable because the Physics Test exceptions do not pose a threat to fuel integrity provided the new restrictions are met.

NSHD category	Change number 3.1-	Discussion Of Change
M	29	New SRs, 3.1.8.1, 3.1.8.2, 3.1.8.3 and 3.1.8.4 are included to perform COT on power and intermediate range NIS channels, verify RCS lowest loop average temperature, verify Thermal Power and verify SDM is within limits. These new SRs are consistent with the guidance of NUREG-1431. Since these SRs are new requirements, these changes are more restrictive. The SR 3.1.8.1 is acceptable since these are activities that are currently performed in support of Physics Tests. The SRs 3.1.8.2, 3.1.8.3 and 3.1.8.4 are acceptable because their performance does not impact normal plant operations. Therefore, the changes imposed by these new SRs are acceptable. This change is included to make the PI ITS complete.
	30	Not used.
M	31	CTS 3.10.D.3. CTS provisions for SDM exceptions during Physics Tests are not included in the ITS. NUREG-1431 as modified by approved TSTF-12, Rev. 1 does not include SDM exceptions. The CTS exceptions were required to perform the rod worth measurement in the N-1 condition. The use of other rod worth measurement techniques will maintain the SDM during the entire verification. Since this measurement technique is no longer used, the SDM test exception can be deleted. Since this change removes operational flexibility it is a more restrictive change. This change is acceptable since the plant will continue to be operated within the TS requirements without exception to SDM requirements. This change is included to make the PI ITS complete.

NSHD category	Change number 3.1-	Discussion Of Change
M	32	CTS 3.10.E.1. A new Action Statement is included consistent with the guidance of NUREG-1431 which requires verification that SDM requirements are met or initiate boration. This change is acceptable since it requires conservative operator actions in response to a possible abnormal situation. This change is included to make the PI ITS complete.
L	33	CTS 3.10.E.1. CTS requires the high neutron flux trip setpoint to be reduced to 85 percent of rating. This change allows the plant power level to be reduced to 85 percent of rating and the high flux trip setpoint remains unchanged. This change is acceptable since the power will be reduced and most of the safety benefits will be achieved. Adjusting the high neutron flux trip setpoint may introduce plant transients which may negate any further benefit which could be gained. This change is consistent with the guidance of NUREG-1431.
A	34	CTS 3.10.E.2. In conformance with the guidance of NUREG-1431, when a rod is misaligned, the SDM is verified and remedial action is taken, but the rod is NOT declared inoperable. Under the provisions of CTS 3.10.E.2, the misaligned rod is eventually declared inoperable and under CTS 3.10.G.2 the plant is allowed to continue to operate with ONE inoperable rod. The effect of declaring the rod inoperable is to require verification of SDM and take remedial action in accordance with the requirements of CTS 3.10.G.3 and 4. Thus removing this paragraph does not have any net effect on plant operations under ITS requirements. This change is therefore considered an administrative change.

NSHD category	Change number 3.1-	Discussion Of Change
	35	Not used.
A	36	CTS 3.10.F. A new note which explicitly allows Separate Condition entry for each inoperable rod position indicator and each demand position indicator is included. This change is consistent with the guidance of NUREG-1431. CTS allow multiple rod position indicators to be inoperable as indicated by 3.10.F.2 and 3.10.F.3 which provide Action Statements when more than one RPI is inoperable per group; i.e., more than one group may have an inoperable RPI. Thus this change does not allow additional flexibility and is therefore considered an administrative change.
LR	37	CTS 3.10.F.1. The statements which define the capabilities of the control rod position indication system have been relocated to the Bases. This change is consistent with the guidance of NUREG-1431. This change is acceptable since the definition will be available in the Bases and is not needed in the TS. Since the ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) are licensee controlled, this change is less restrictive.
M	38	CTS 3.10.F.1, 3.10.F.2 and 3.10.F.3.In conformance with the guidance of NUREG-1431, the MODES of applicability for this specification are extended to MODE 2. This change is acceptable since it requires RPI to be operable over a greater range of operation and is thus conservative. This change does not cause any unsafe plant conditions. Since this change requires the specification to apply more extensively, it is a more restrictive change.

NSHD category	Change number 3.1-	Discussion Of Change
	39	Not used.
	40	Not used.
	41	Not used.
M	42	A new Action Statement is included for demand position indications. Since CTS do not include actions for these indications, this is a more restrictive change. This change is acceptable since it assures the plant is in a safe condition when all demand position indication methods are inoperable. This change is consistent with the guidance of NUREG-1431. This change is included to make the PI ITS complete.
LR	43	CTS 3.10.G.1. The CTS definition of an inoperable rod is not included in the ITS. The ITS Bases defines an inoperable rod for the associated Action Statements; therefore, the definition is not necessary in the TS. This change is consistent with the guidance of NUREG-1431. Since the ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) are licensee controlled, this change is less restrictive.
M	44	CTS 3.10.G.2. In conformance with NUREG-1431, the plant is required to be shutdown if one rod is inoperable. Since CTS allow operations to continue with one rod inoperable, this change is more restrictive. This change is acceptable since shutting down the plant will maintain it in a safe condition with one rod inoperable. This change is included to make the PI ITS complete.

NSHD category	Change number 3.1-	Discussion Of Change
	45	Not used.
	46	Not used.
M	47	CTS 3.10.G.3 and 3.10.G.4. In conformance with the guidance of NUREG-1431, whenever a rod is inoperable, the SDM requirements shall be met within one hour or boration initiated. Since the SDM limits are required to be met within a specific time, this change is more restrictive. This change is acceptable because verification of SDM and the associated remedial actions within 1 hour assure the plant is maintained in a safe condition. This change is included to make the PI ITS complete.
A	48	CTS 3.10.G.5. The Action Statements of this CTS specification are consistent with NUREG-1431 Action Statements for one misaligned rod. Since CTS declares a misaligned rod to be inoperable and operations to continue, the applicability for this specification has been changed to apply to a misaligned rod. The impact on plant operations remains the same with this change; therefore, this is an administrative change.
	49	Not used.
	50	Not used.

NSHD category	Change number 3.1-	Discussion Of Change
LR	51	CTS 3.10.G.5. Both CTS and ITS require performance of safety analyses when a rod is misaligned. Specific CTS details for consideration in these analyses have been relocated to the Bases. There are many parameters that must be considered in these analyses of which only a few were included in the TS. This change is acceptable because it is sufficient in TS to require the analyses and an incomplete set of additional details such as those in CTS are unnecessary. Since the ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) are licensee controlled, this change is less restrictive. This change is consistent with the guidance of NUREG-1431.
A	52	CTS 3.10.G.6. For consistency with the provisions of the ITS, if a rod is trippable but not movable, rod alignment is required to be verified to determine which specifications apply. If the rod is not in alignment then ITS LCO3.1.4 Action Statement B applies. The changes to this specification clarify how the CTS requirements relate to ITS. Since this wording change does not change the impact on plant operations, this is an administrative change.

NSHD category	Change number 3.1-	Discussion Of Change
M	53	A new Action Statement is included which requires the unit to be in MODE 3 in 6 hours if ITS LCO 3.1.4 Action B requirements are not met. Under CTS, an immovable, unrestored rod would be declared inoperable but continued plant operation would be allowed. Since this new Action Statement requires unit shutdown, this is a more restrictive change. This change is acceptable because it conservatively shuts the plant down to maintain it in a safe condition when rod misalignment Actions are not met. This change is consistent with the guidance of NUREG-1431. This change is included to make the PI ITS complete.
A	54	CTS 3.10.G.5. CTS require plant shutdown if more than one rod is misaligned. The CTS specification which applied for this condition was revised to apply to inoperable rods. Thus, for clarity this Action was restated to apply to misaligned rods. Since this change does not affect plant operations, it is an administrative change.
	55	Not used.
	56	Not used.

NSHD category	Change number 3.1-	Discussion Of Change
A	57	CTS 3.10.H. CTS terminology "loss of stationary gripper coil voltage" is replaced by NUREG-1431 terminology "beginning of decay of stationary gripper coil voltage". In practice, at PI the time is measured from the loss of current since the coil voltage is not monitored. Thus use of NUREG-1431 terminology does not change the SR performance and therefore this is an administrative change.
	58	Not used.
LR	59	CTS 3.10.I.1 and Table 4.1-1C F.U. 1, 2 and 4. The CTS requirements for instrument surveillances on the rod bank insertion limit monitor and rod position deviation monitor are relocated to the TRM. CTS provisions for additional logging of rod positions when these instruments are inoperable are also relocated to the TRM. This is acceptable since these alarms do not directly relate to any LCO limits. These alarms are for indication purposes only and there is no adverse effect in permitting the normal surveillance frequency to be used instead of the frequency associated with these alarms. Since the TRM is part of the USAR it is under licensee control in accordance with 10CFR50.59 and can be changed without prior NRC approval. Therefore this change is less restrictive. This change is consistent with the guidance of NUREG-1431 as modified by approved TSTF-110, Rev. 2.
	60	Not used.
	61	Not used.

NSHD category	Change number 3.1-	Discussion Of Change
M	62	CTS 4.9. CTS require monthly verification of core reactivity "Following a normalization of the computed boron concentration as a function of burnup". The time for this normalization is not specified in the CTS. In conformance with the guidance of NUREG-1431, this normalization is required to be met within 60 EFPD. Since a time restraint is specified, this change is more restrictive. 60 EFPD is sufficient time for core conditions to reach steady state, but prevents operation for a large fraction the fuel cycle without establishing a benchmark for the design calculations. This change is acceptable since it assures that the core is properly evaluated and the plant is maintained in a safe condition. This change is included to make the PI ITS complete.
L	63	CTS Table 4.1-1C F.U. 2 and Note 31. CTS requirements for operability of the rod position indication system in MODES 3, 4, and 5 have not been included in the PI ITS. This change is acceptable because no fission power is generated in these modes and alignment limits do not apply because the control rods are bottomed and the reactor is shutdown. In the shutdown modes, the operation 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. Since this change allows equipment to be out of service in additional modes it is a less restrictive change. This change is consistent with the guidance of NUREG-1431.

NSHD category	Change number 3.1-	Discussion Of Change
L	64	CTS Table 4.1-1C Note 30. CTS requirements to functionally test the rod position indication "prior to each startup following shutdown in excess of two days if not done in the previous 30 days" has been replaced by "prior to criticality after each removal of the reactor head". This change is consistent with NUREG-1431 as modified by approved TSTF-89. This change is acceptable because the plant activity which may affect the rod position indication system is reactor head removal. Since this change could result in performing this surveillance less often, this change is less restrictive.
LR	65	CTS Table 4.1-2A, Item 1. CTS requirements to measure rod drop times after each refueling or following maintenance or modification to the control rod drive system will be relocated to the TRM. Normal plant practices dictate that post-maintenance and post-modification testing is performed to assure the proper performance for the affected equipment. Thus, this change is acceptable because these details are unnecessary in the TS. Since the TRM is part of the USAR it is controlled under 10CFR50.59. Since changes to the TRM can be made without prior NRC approval, this change is less restrictive. This change is consistent with the guidance of NUREG-1431.

NSHD category	Change number 3.1-	Discussion Of Change
M	66	CTS Table 4.1-2A, Item 2. In conformance with the guidance of NUREG-1431, the control rods are required to be moved 10 or more steps to demonstrate operability every quarter. CTS does not specify a minimum number of steps, thus this change is more restrictive. This change is acceptable because movement of an individual rod has been evaluated for complete insertion or withdrawal; thus, movement more than 10 steps will not cause an unsafe condition. This change also exempts rods which are fully inserted from being moved. This is not a significant change since PI does not normally operate with any rods fully inserted and if the rod is fully inserted, then it is not of concern that it may not move if the reactor trips. This change is included to make the PI ITS complete.

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART E

MARKUP OF NUREG-1431 IMPROVED STANDARD TECHNICAL SPECIFICATIONS AND BASES

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

~~ML003772000~~ 4

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM) - ~~T_{avg} > 200°F~~

TA3.1-76

LCO 3.1.1 SDM shall be within the limits provided in the COLR~~≥ [1.6]% Δk/k~~.

TA3.1-77

APPLICABILITY: MODE 2 with k_{eff} < 1.0,
MODES 3, 4, ~~[~~and 5~~]~~.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is within limits~~≥ [1.6]% Δk/k~~.	TA3.1-77 24 hours

~~3.1 REACTIVITY CONTROL SYSTEMS~~

TA3.1-76

~~3.1.2 SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F~~

~~LCO 3.1.2 The SDM shall be ≥ [1.0]% Δk/k.~~

~~APPLICABILITY: MODE 5.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. SDM not within limit.~~	~~A.1 Initiate boration to restore SDM to within limit.~~	~~15 minutes~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~	~~FREQUENCY~~
~~SR 3.1.2.1 Verify SDM is ≥ [1.0]% Δk/k.~~	~~24 hours~~

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2~~3~~ Core Reactivity

TA3.1-76

LCO 3.1.2~~3~~ The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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~~72 hours~~ TA3.1-79
	<u>AND</u>		
	A.2	Establish appropriate operating restrictions and SRs.	7 days~~72 hours~~
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify measured core reactivity is within ± 1% $\Delta k/k$ of predicted values.	PA3.1-81 Prior to entering MODE 1 after each refueling
SR 3.1.2~~3~~.2~~1~~	-------------------NOTES------------------ 1. Only required to be performed after 60 effective full power days (EFPD). 2. 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. -- Verify measured core reactivity is within ± 1% $\Delta k/k$ of predicted values.	PA3.1-81 ~~Once prior to entering MODE 1 after each refueling~~ ~~AND~~ ~~-----NOTE-----~~ ~~Only required after 60 EFPD~~ ~~--------------~~ 31 EFPD ~~thereafter~~

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3~~4~~ Isothermal ~~Moderator~~ Temperature Coefficient (IMTC)

TA3.1-76

CL3.1-82

LCO 3.1.3~~4~~ The IMTC shall be maintained within the limits specified in the COLR. The maximum COLR upper limit shall be:

CL3.1-83

a. ≤ 5 pcm/~~[≤ [ ] Δk/k~~°F for ~~at hot zero~~ power levels ≤ 70% RTP; and

b. ≤ 0 pcm/°F for power levels > 70% RTP~~] [that specified in Figure 3.1.4-1]~~.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper IMTC limit, MODES 1, 2, and 3 for the lower IMTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. IMTC not within upper limit.	A.1	Establish administrative withdrawal limits for control banks to maintain IMTC within limit.	24 hours
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -------NOTE------ Required Action C.1 must be completed whenever Condition C is entered. ----------------- Projected end of cycle (EOC) IMTC not within lower limit.	------------NOTE----------- LCO 3.0.4 is not applicable. -------------------------- C.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation. ~~Be in MODE 4.~~	PA3.1-84 Once prior to reaching the equivalent of an equilibrium RTP all rods out boron concentration of 300 ppm ~~12 hours~~
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 4.	12 hours PA3.1-84

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.4.2 Confirm~~Verify~~ MTC will be~~is~~ within ~~300 ppm Surveillance~~ limits at 70% RTP ~~specified in the COLR~~.	PA3.1-85 ~~NOTE~~ ~~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~~ Once after each refueling ~~cycle~~ prior to THERMAL POWER exceeding 70% RTP

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.4.3 ~~-----NOTES-----~~ ~~1. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.4.3 shall be repeated once per 14 EFPD during the remainder of the fuel cycle.~~ ~~2. SR 3.1.4.3 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.~~ ----- Confirm that ~~Verify~~ MTC will be ~~is~~ within ~~lower~~ limits at EOC.	~~-----NOTE-----~~ ~~Not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm~~ ----- PA3.1-84 Once after each refueling ~~cycle~~ prior to THERMAL POWER exceeding 70% RTP

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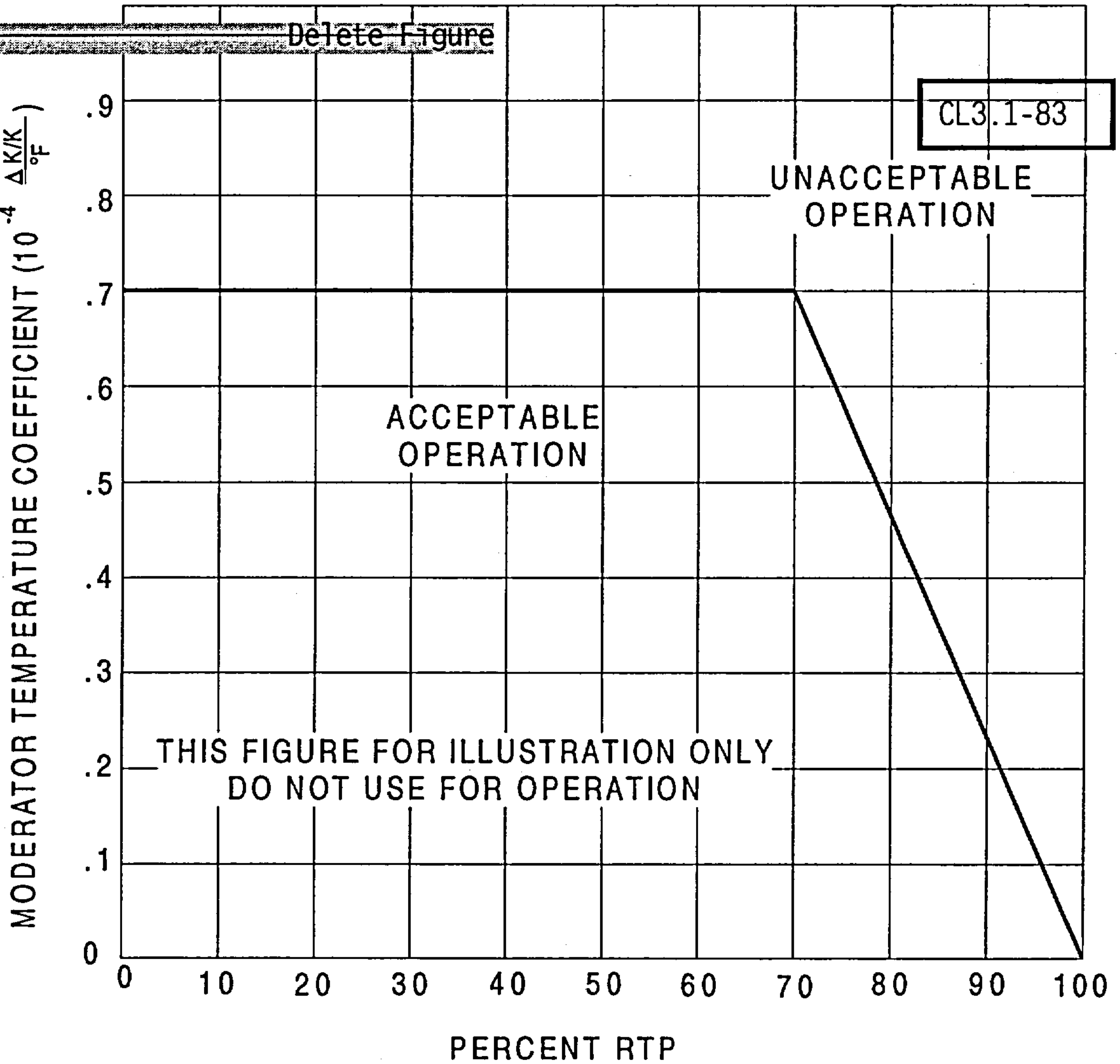

Figure 3.1.4-1 (page 1 of 1)
Moderator Temperature Coefficient vs. Power Level

3.1 REACTIVITY CONTROL SYSTEMS

TA3.1-76

3.1.4~~5~~ Rod Group Alignment Limits

TA3.1-86

CL3.1-87

LCO 3.1.4~~5~~ All shutdown and control rods shall be OPERABLE and~~, with all~~ individual actual~~indicated~~ rod positions shall be within 24~~12~~ steps of their group step counter demand position when the demand position is between 30 and 215 steps, or within 36 steps of their group step counter demand position when the demand position is ≤ 30 steps, or ≥ 215 steps.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more rod(s) inoperable~~untrippable~~.	A.1.1	Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~.	1 hour
	<u>OR</u>		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	A.2	Be in MODE 3.	6 hours

TA3.1-86

TA3.1-77

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limits.	~~B.1 Restore rod to within alignment limits.~~ ~~OR~~ B.~~2.~~1.1 Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~. OR	~~1 hour~~ PA3.1-88 1 hour TA3.1-77 (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
B. (continued)	B.~~2.~~1.2 Initiate boration to restore SDM to within limit.	1 hour	
	AND		
	B.2.1.1 Perform SR 3.2.1.1 and SR 3.2.1.2.	2 hours	CL3.1-89 TA3.1-90
	AND		
	B.2.1.2 Perform SR 3.2.2.1.	2 hours	CL3.1-89
	OR		
	B.2.2 Reduce THERMAL POWER to ≤ 8~~7~~5% RTP.	2 hours	CL3.1-89
	AND		
	B.~~2.~~3 Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~.	Once per 12 hours	TA3.1-77
	AND		
	B.~~2.~~4 ~~Perform SR 3.2.1.1.~~	~~72 hours~~	
	~~AND~~		
	~~B.2.5 Perform SR 3.2.2.1.~~	~~72 hours~~	
	~~AND~~		

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	~~B.2.6~~ Re-evaluate safety analyses and determine the THERMAL POWER for which the ~~confirm~~ results remain valid for duration of operation under these conditions.	30~~5~~ days CL3.1-89
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)

D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~. <u>OR</u> D.1.2 Initiate boration to restore required SDM to within limit. <u>AND</u> D.2 Be in MODE 3.	1 hour TA3.1-77 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.45.1	------------------NOTE------------------ If RPI differs by > 12 steps from the group step counter demand position, enter LCO 3.1.7 to determine RPI OPERABILITY. -- Verify individual rod positions within alignment limit.	PA3.1-91 12 hours ~~AND~~ ~~Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable~~ TA3.1-97
SR 3.1.45.2	Verify rod freedom of movement (trippability) by moving each rod, not fully inserted in the core, ≥ 10 steps in either direction.	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4~~5~~.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 1.8~~[2.2]~~ seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F; and b. Both~~All~~ reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

CL3.1-92

3.1 REACTIVITY CONTROL SYSTEMS

TA3.1-76

3.1.5~~6~~ Shutdown Bank Insertion Limits

LCO 3.1.5~~6~~ Each shutdown bank shall be within insertion limits specified in the COLR.

PA3.1-93

-----------------------------NOTE-----------------------------
This LCO is not applicable while performing SR 3.1.4.2.

APPLICABILITY: MODES 1~~,~~ and 2.
~~MODE 2 with any control bank not fully inserted.~~

TA3.1-94

~~-----------------------------NOTE-----------------------------~~
~~This LCO is not applicable while performing SR 3.1.5.2.~~
~~--~~

PA3.1-93

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown banks to within limits.	2 hours

TA3.1-77

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5~~6~~.1 Verify each shutdown bank is within the limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6~~7~~ Control Bank Insertion Limits TA3.1-76

LCO 3.1.6~~7~~ Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR. PA3.1-93

-----------------------------NOTE-----------------------------
----- This LCO is not applicable while performing SR 3.1.4.2.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

~~-----------------------------NOTE-----------------------------~~
~~This LCO is not applicable while performing SR 3.1.5.2.~~ PA3.1-93
~~---~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1 Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~.	1 hour TA3.1-77
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore control bank(s) to within limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~. <u>OR</u> B.1.2 Initiate boration to restore SDM to within limit. <u>AND</u> B.2 Restore control bank sequence and overlap to within limits.	1 hour 1 hour 2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2 with K_{eff} < 1.0~~3~~.	6 hours

TA3.1-77

TA3.1-95

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6~~7~~.1 Verify estimated critical control bank position is within the limits specified in the COLR.	~~Within 4 hours~~ P~~p~~rior to achieving criticality

PA3.1-96

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		(continued)
SR 3.1.6~~7~~.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours ~~AND~~ ~~Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable~~
SR 3.1.6~~7~~.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours

TA3.1-97

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7~~8~~ Rod Position Indication

TA3.1-76

LCO 3.1.7~~8~~ The ~~[Digital]~~ Rod Position Indication (~~[D]~~RPI) System and ~~the~~ d~~D~~emand p~~P~~osition i~~I~~ndication~~ System~~ shall be OPERABLE.

CL3.1-98

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable rod position indicator ~~per group~~ and each demand position indicator~~ per bank~~.

TA3.1-99

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 position indicators by using movable incore detectors.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

CL3.1-101

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one RPI per group inoperable for one or more groups.	B.1	Monitor and record demand position indication for rods with inoperable RPI.	Once per hour
	<u>AND</u>		
	B.2	Monitor and record reactor coolant system average temperature.	Once per hour
	<u>AND</u>		
	B.3	Verify, using movable incore detectors, position of rods with inoperable RPIs which have been moved in excess of 24 steps in one direction since last determination of their position.	Once per 4 hours
	<u>AND</u>		
	B.4	Restore inoperable RPIs to OPERABLE status such that a maximum of one RPI per group is inoperable.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. 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.~~	~~B.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.~~ ~~OR~~	~~[4] hours~~ ~~(continued)~~
~~B. (continued)~~	~~B.2 Reduce THERMAL POWER to ≤ 50% RTP.~~	~~8 hours~~
C. Indication for o~~O~~ne demand position ~~indicator~~ per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all ~~[D]~~RPIs for the affected bank(s) are OPERABLE. AND C.1.2 Verify RPI of the most withdrawn rod and the least withdrawn rod of the affected bank(s) are ≤ 12 steps apart. OR C.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours Once per 8 hours 8 hours

CL3.1-101

X3.1-108

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7~~8~~.1 Verify each ~~[D]~~RPI agrees within ~~[12]~~ steps of the group demand position between 30 and 215 steps, or within 24 steps of the group demand position when the demand position is ≥ 215 steps or ≤ 30 steps ~~for the [full indicated range] of rod travel~~.	Once prior to criticality after each removal of the reactor head ~~[18 months]~~

CL3.1-101

TA3.1-102

3.1 REACTIVITY CONTROL SYSTEMS

TA3.1-103

3.1.~~10~~8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.~~10~~8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3~~4~~, "Isothermal ~~Moderator~~ Temperature Coefficient (IMTC)"; CL3.1-82
LCO 3.1.4~~5~~, "Rod Group Alignment Limits";
LCO 3.1.5~~6~~, "Shutdown Bank Insertion Limits";
LCO 3.1.6~~7~~, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e may be reduced to "3" required channels, provided: TA3.1-111

a. RCS lowest loop average temperature is ≥ 535 ~~[531]~~°F; ~~and~~ PA3.1-104

b. SDM is within the limits provided in the COLR; and ~~≥ [1.6]% Δk/k.~~ TA3.1-77

c. THERMAL POWER is ≤ 5% RTP. TA3.1-105

APPLICABILITY: ~~MODE 2~~ D~~d~~uring PHYSICS TESTS initiated in MODE 2.

TA3.1-106

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. <u>AND</u> 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
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.~~10~~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~~]~~.	~~Within 12 hours~~ P~~p~~rior to initiation of PHYSICS TESTS TA3.1-107
SR 3.1.~~10~~8.2 Verify the RCS lowest loop average temperature is ≥ 535 ~~[531]~~°F.	30 minutes PA3.1-104
SR 3.1.8.3 Verify THERMAL POWER is ≤ 5% RTP.	30 minutes TA3.1-105
SR 3.1.~~10~~8.~~4~~3 Verify SDM is within the limits provided in the COLR ~~≥ 1.6% Δk/k~~.	24 hours TA3.1-77

TA3.1-103

~~3.1 REACTIVITY CONTROL SYSTEMS~~

~~3.1.9 PHYSICS TESTS Exceptions - MODE 1~~

~~LCO 3.1.9~~ ~~During the performance of PHYSICS TESTS, the requirements of~~

~~LCO 3.1.5, "Rod Group Alignment Limits";~~
~~LCO 3.1.6, "Shutdown Bank Insertion Limits";~~
~~LCO 3.1.7, "Control Bank Insertion Limits";~~
~~LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and~~
~~LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)"~~

~~may be suspended, provided:~~

~~a. THERMAL POWER is maintained ≤ 85% RTP;~~

~~b. Power Range Neutron Flux - High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and~~

~~c. SDM is ≥ [1.6]% Δk/k.~~

~~APPLICABILITY: MODE 1 during PHYSICS TESTS.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. SDM not within limit.~~	~~A.1 Initiate boration to restore SDM to within limit.~~	~~15 minutes~~
	~~AND~~	
	~~A.2 Suspend PHYSICS TESTS exceptions.~~	~~1 hour~~

ACTIONS (continued)

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
		~~(continued)~~
~~B. THERMAL POWER not within limit.~~	~~B.1 Reduce THERMAL POWER to within limit.~~	~~1 hour~~
	~~OR~~	
	~~B.2 Suspend PHYSICS TESTS exceptions.~~	~~1 hour~~
~~C. Power Range Neutron Flux – High trip setpoints > 10% RTP above the PHYSICS TEST power level.~~ ~~OR~~ ~~Power Range Neutron Flux – High trip setpoints > 90% RTP.~~	~~C.1 Restore Power Range Neutron Flux – High trip setpoints to ≤ 10% above the PHYSICS TEST power level, or to ≤ 90% RTP, whichever is lower.~~	~~1 hour~~
	~~OR~~	
	~~C.2 Suspend PHYSICS TESTS exceptions.~~	~~1 hour~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.1.9.1~~	~~Verify THERMAL POWER is ≤ 85% RTP.~~	~~1 hour~~
~~SR 3.1.9.2~~	~~Verify Power Range Neutron Flux~~ - ~~High trip setpoints are ≤ 10% above the PHYSICS TEST power level, and ≤ 90% RTP.~~	~~Within 8 hours prior to initiation of PHYSICS TESTS~~
~~SR 3.1.9.3~~	~~Perform SR 3.2.1.1 and SR 3.2.2.1.~~	~~12 hours~~
~~SR 3.1.9.4~~	~~Verify SDM is ≥ [1.6]% Δk/k.~~	~~24 hours~~

TA3.1-103

~~3.1 REACTIVITY CONTROL SYSTEMS~~

~~3.1.11 SHUTDOWN MARGIN (SDM) Test Exceptions~~

~~LCO 3.1.11~~ ~~The SDM requirements in MODE 2 may be suspended, 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 when measuring control rod worth and SDM.~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One or more control rods not fully inserted.~~ ~~AND~~ ~~Available trip reactivity from OPERABLE control rods less than the highest estimated control rod worth.~~	~~A.1 Initiate boration to restore SDM to within limit.~~	~~15 minutes~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~B. All control rods fully inserted.~~ ~~AND~~ ~~Reactor subcritical by less than the highest estimated control rod worth.~~	~~B.1 Initiate boration to restore SDM to within limits.~~	~~15 minutes~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.1.11.1~~	~~NOTE~~ ~~Only required for control rods not fully inserted.~~ ~~Determine the position of each control rod.~~	~~2 hours~~
~~SR 3.1.11.2~~	~~NOTE~~ ~~Only required for control rods not fully inserted.~~ ~~Trip each control rod from ≥ the 50% withdrawn position, and verify full control rod insertion.~~	~~Within 24 hours prior to reducing SDM outside limits~~

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) – ~~T_{avg} > 200°F~~

TA3.1-76

PA3.1-121

BASES

BACKGROUND

According to AEC GDC Criteria 27 and 28~~26~~ (Ref. 1), two independent~~the~~ reactivity control systems must be provided which are~~redundant and~~ capable of holding the reactor core subcritical from any hot standby or hot operating~~when shut down under cold~~ condition~~s~~. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. CL3.1-122

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 control assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperatures are changed to the nominal hot zero power temperature, 547 °F. PA3.1-123

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 Rod 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 Rod 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

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(continued)

PA3.1-121 TA3.1-76

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.

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~~7~~, "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.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an 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. The primary safety analyses that rely on the SDM limits are the boron dilution and MSLB analyses. CL3.1-124

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;

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

(continued)

PA3.1-121

TA3.1-76

BASES (continued)

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements, at end of cycle (EOC), 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 an ~~the~~ RCS cooldown. 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 the initial 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 as a result of the return to power will not cause offsite doses to exceed the 10CFR100 limits ~~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~~. CL3.1-124

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~In addition to the limiting MSLB transient, the SDM requirement must also protect against:~~ CL3.1-124

~~a. Inadvertent boron dilution;~~

~~b. An uncontrolled rod withdrawal from subcritical or low power condition;~~

(continued)

PA3.1-121

TA3.1-76

~~c. Startup of an inactive reactor coolant pump (RCP); and~~ CL3.1-124

~~d. Rod ejection.~~

~~Each of these events is discussed below.~~

The most limiting accident at beginning of cycle (BOC) is~~In~~ the boron dilution accident~~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, that is, the time available to operators to stop the dilution event. As the unit changes MODES the volume being diluted may change, i.e., if RHR is in service, as well as the critical boron concentration due to the different temperature ranges. Thus different SDMs may be required for the different modes and dilution flow rates. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

~~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.~~ CL3.1-124

~~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.~~

(continued)

PA3.1-121

TA3.1-76

BASES (continued)

~~APPLICABLE SAFETY ANALYSES~~ (continued)

~~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 ~~the NRC Policy Statement~~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. TA3.1-76

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 in the RCS.

The COLR provides the shutdown margin requirements. The MSLB ~~(Ref. 2)~~ and the boron dilution ~~(Ref. 3)~~ accidents (Ref. 2) are the most limiting analyses that establish the SDM requirements in the COLR ~~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. 4)~~. 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. PA3.1-126 CL3.1-127

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3, ~~and~~ 4, and 5 the SDM requirements are applicable to provide sufficient

(continued)

PA3.1-121

TA3.1-76

BASES (continued)

negative reactivity to meet the assumptions of the safety analyses discussed above. ~~[In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F."]~~ In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2 with K_{eff} ≥ 1.0, the SDM requirements specified in the COLR are ~~is~~ ensured by complying with LCO 3.1.5~~6~~, "Shutdown Bank Insertion Limits," and LCO 3.1.6~~7~~, "Control Bank Insertion Limits." TA3.1-76

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, and the probability of a design basis accident (DBA) occurring during this time is very low. It is assumed that CL3.1-128

ACTIONS

A.1 (continued)

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 water storage tank. The~~ operator should borate with the best source available for the plant conditions. PA3.1-129

(continued)

PA3.1-121

TA3.1-76

BASES

~~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 [] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 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 [] gpm and [] ppm represent typical values and are provided for the purpose of offering a specific example.~~ PA3.1-131

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5~~6~~ and LCO 3.1.6~~7~~ 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.

In MODE 2 with K_{eff} < 1.0 and MODES 3, 4, and 5, the SDM is verified by comparing the RCS boron concentration to a Shutdown Boron Concentration requirement curve that was generated by taking into account~~performing a reactivity balance calculation, considering the listed reactivity effects~~: CL3.1-132

a. Required SDM~~RCS boron concentration~~; CL3.1-132

b. Shutdown and c~~C~~ontrol bank position;

(continued)

PA3.1-121

TA3.1-76

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration~~; and~~

~~g. Isothermal temperature coefficient (ITC)~~.

~~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.~~

CL3.1-132

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 comparison ~~calculation~~.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27 and 28, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~10 CFR 50, Appendix A, GDC 26~~.

CL3.1-122

2. UFSAR, Sections~~Chapter~~ 14.4 and 14.5~~[15]~~.

(continued)

PA3.1-121

TA3.1-76

BASES

~~3. FSAR, Chapter [15].~~

~~4. 10 CFR 100.~~

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~23~~ Core Reactivity

TA3.1-76

PA3.1-121

BASES

BACKGROUND

According to AEC GDC Criteria 27~~26~~, ~~GDC~~ 28, ~~and~~ ~~GDC~~ 29 and 30 (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 SHUTDOWN MARGIN (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) ~~– T_{avg} > 200°F~~") in ensuring the reactor can be brought safely to cold, subcritical conditions. CL3.1-122 TA3.1-76

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

(continued)

PA3.1-121 TA3.1-76

BASES

BACKGROUND (continued)

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 or stable (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. PA3.1-134

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 RATED THERMAL POWER (RTP) and normal operating~~moderator~~ temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

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.

(continued)

PA3.1-121 TA3.1-76

BASES

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the uncertainties in the nuclear design methods are within the expected range and that the calculational models used to generate ~~reactivity balance limit ensures plant operation is maintained within the assumptions of~~ the safety analyses are adequate. CL3.1-137

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.

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.

APPLICABLE SAFETY ANALYSES (continued)

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 early in the cycle (≤ 60 effective full power days (EFPD)) ~~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 PA3.1-138

(continued)

PA3.1-121 TA3.1-76

BASES

reactivity exists early in the cycle ~~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 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. PA3.1-138

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 early in the cycle ~~at BOC conditions~~, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle. PA3.1-138

Core reactivity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement~~.

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 ~~n~~Nuclear ~~d~~Design ~~m~~Methods~~ology~~ 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

(continued)

PA3.1-121 TA3.1-76

BASES

LCO (continued)

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, 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. Verification of measured core reactivity (SR 3.1.2.1) ~~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). CL3.1-141

(continued)

PA3.1-121

TA3.1-76

BASES

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~~72 hours~~ 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. TA3.1-79

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.

(continued)

BASES (continued) PA3.1-76 PA3.1-121

ACTIONS

A.1 and A.2 (continued)

The required Completion Time of 7 days~~72 hours~~ is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation. TA3.1-79

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant 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 systems. PA3.1-143

SURVEILLANCE REQUIREMENTS

SR 3.1.~~2~~3.1

Core reactivity must be verified following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The comparison is made when the core conditions such as control rod position, moderator temperature, and samarium concentration are fixed or stable. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at beginning of cycle (BOC). PA3.1-81

SR 3.1.2.2

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 control rod position, BASES (continued) PA3.1-76 PA3.1-121

SURVEILLANCE REQUIREMENTS

SR 3.1.2.2 (continued)

moderator temperature, fuel temperature, fuel depletion, 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 required Frequency of 31 effective full power days (EFPD) 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. The SR is modified by two~~a~~ Notes. Note 1 states that the SR is only required to be performed after 60 EFPD. ~~The~~ Note 2 indicates that the normalization of predicted core reactivity to the measured value may~~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.~~ PA3.1-81

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 27, 28, 29 and 30, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29~~. CL3.1-122

2. UFSAR, Section~~Chapter~~ 14~~[15]~~.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.[3]~~4~~ [Isothermal]~~Moderator~~ Temperature Coefficient ([I]MTC)

CL3.1-82

TA3.1-76

PA3.1-121

BASES

BACKGROUND

According to ~~AEC~~ GDC ~~Criterion~~ 8~~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. CL3.1-122

The moderator temperature coefficient (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 ITC is defined as the reactivity change associated with a unit change in the moderator and fuel temperatures. Essentially, the ITC is the sum of the MTC and fuel temperature coefficient. The ITC is measured directly during low power PHYSICS TEST in order to verify analytical prediction of the MTC. The units of the isothermal temperature coefficient are pcm/°F, where 1 pcm = $1 \times 10^{-5} \Delta k/k$. CL3.1-144

The reactor is designed to operate with a negative [I]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.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements at beginning of cycle PA3.1-146

(continued)

BASES

(BOC). Reactor ~~Both initial and reload~~ cores are designed so that the ~~beginning of cycle (BOC)~~ MTC is less than 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 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~~ limits. CL3.1-146

The limitations on MTC are provided to ensure that the value of MTC~~this coefficient~~ remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

BACKGROUND (continued)

~~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.~~ PA3.1-147

~~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 to the principally to the reduction in RCS boron concentration associated with fuel burnup.~~ PA3.1-148

(continued)

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

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR~~, 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 for the cycle exposure being evaluated to ensure that the accident results are bounding~~ (Ref. 3)~~.

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive (i.e., upper limit). 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 the main steam line break~~, sudden feedwater flow increase and sudden decrease in feedwater temperature~~. CL3.1-151

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,

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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. 2).

~~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.~~ CL3.1-153

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement~~. 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.

LCO

LCO 3.1.34 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 will remain within the bounds of the original accident analyse~~i~~s 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 usually 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.

(continued)

BASES

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance check~~s~~ at BOC ~~and EOC~~ on IMTC provides confirmation that the IMTC is behaving as anticipated and will be within limits at 70% RTP, full power, and EOC so that the acceptance criteria are met. PA3.1-157

CL3.1-154

LCO (continued)

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.

If the LCO limits are not met, the assumptions of the safety analysis may not be met. The core could violate criteria that prohibit a return to criticality, or the DNBR criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity. PA3.1-147

APPLICABILITY

Technical Specifications place both LCO and SR values on IMTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on IMTC 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 rod cluster control~~CONTROL ROD assembly or group~~ withdrawal) will not violate the assumptions of the accident analysis. The lower IMTC limit must be maintained in MODES 2 and 3, in addition

(continued)

BASES

to MODE 1, to ensure that cooldown accidents at EOC will not violate the assumptions of the accident analysis since ITC becomes more negative as the cycle burnup increases and the RCS boron concentration is reduced. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

ITC must be kept within the upper limit specified in LCO 3.1.3 to ensure that assumptions made in the safety analysis remain valid. The upper limit of Condition A is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO. PA3.1-158

If the upper ~~BOC~~ [I]MTC limit is violated at BOC, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits in the future. The [I]MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the [I]MTC measurement and computing the required bank withdrawal limits. PA3.1-159

The control rods are maintained within the administrative withdrawal limits until a subsequent calculation verifies that ITC has been restored with its limit. As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the [I]MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated [I]MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no CL3.1-156

(continued)

BASES

longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

ACTIONS (continued)

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.

C.1

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 it is determined during PHYSICS TESTS that the EOC [MTC value will exceed the most negative ITC limit specified in the COLR, the safety analysis and core design must be re-evaluated prior to reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm to ensure that operation near the EOC remains acceptable. The 300 ppm limit is sufficient to prevent EOC operation at or below the accident analysis MTC assumptions. ~~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.~~ PA3.1-84

~~The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from~~

(continued)

BASES

~~full power conditions in an orderly manner and without challenging plant systems.~~

Condition C has been modified by a NOTE that requires Required Action C.1 to be completed whenever this Condition is entered. This is necessary to ensure that the plant does not operate at conditions where the ITC would be below the most negative limit specified in the COLR. PA3.1-84

Required Action C.1 is modified by a Note which states that LCO 3.0.4 is not applicable. This Note is provided since the requirement to re-evaluate the core design and safety analysis prior to reaching an equivalent RTP ARO boron concentration of 300 ppm is adequate action without restricting entry into MODE 1.

D.1

If the re-evaluation of the safety analysis cannot support the predicted EOC ITC lower limit, or if the Required Actions of Condition C are not completed within the associated Completion Time the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 4 within 12 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 systems. PA3.1-84

SURVEILLANCE REQUIREMENTS

SR 3.1.84.1

(continued)

BASES

This SR requires measurement of the IMTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive IMTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC IMTC value for ARO will be ~~inferred from isothermal temperature coefficient measurements~~ obtained from measurements during the physics tests after refueling. The ARO value can be directly compared to the BOC IMTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks. CL3.1-161

Measurement of the ITC at the beginning of the fuel cycle is adequate to confirm that the ITC remains within its upper limit. PA3.1-148

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.34.2 ~~and SR 3.1.4.3~~

This SR requires measurement of ITC at BOC prior to exceeding 70% RTP after each refueling in order to confirm compliance with the 70% RTP ITC limit. The Frequency of "Once after each refueling prior to THERMAL POWER exceeding 70% RTP" ensures the limit will be met prior to being applicable. ~~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 LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value~~ CL3.1-161

(continued)

BASES

~~to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.~~

~~SR 3.1.4.3 is modified by a Note that includes the following requirements:~~

~~a. 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.~~

~~b. 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 because of the gradual manner in which MTC changes with core burnup.~~

SR 3.1.3.3

This SR requires measurement of ITC at BOC prior to exceeding 70% RTP after each refueling in order to confirm compliance with the most negative ITC LCO. Meeting this limit prior to exceeding 70% RTP ensures that the limit will also be met at EOC. PA3.1-84

The ITC value for EOC is derived from the ITC low power PHYSICS TESTS. The EOC value is calculated using the predicted EOC ITC from the core design report and the difference between the measured and predicted BOC ITC. The predicted EOC value is directly compared to the most negative EOC value established in the COLR to ensure that the predicted EOC negative ITC value is within the safety analysis assumptions.

BASES

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 8, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~10 CFR 50, Appendix A, GDC 11~~. CL3.1-122

2. UFSAR, Sections~~Chapter~~ 14.4 and 14.5~~[15]~~.

3. ~~WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.~~

4. ~~FSAR, Chapter [15].~~

TA3.1-76
CL3.1-162
CL3.1-163

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4~~5~~ Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e.~~g.~~, 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. TA3.1-86

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC Criteria 6, 14, 27, and 28~~10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection"~~ (Ref. 1), and 10 CFR 50.46~~, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"~~ (Ref. 2). CL3.1-122

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. ~~Control r~~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. TA3.1-164

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. TA3.1-164 PA3.1-166

(continued)

BASES

BACKGROUND (continued)

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. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Both~~All~~ units have four control banks and ~~at least~~ two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, 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 indications~~systems~~, which are the bBank dDemand pPosition iIndication ~~System~~ (usually the ~~commonly called~~ group step

(continued)

BASES

counters) and the individual~~Digital~~ Rod Position Indication (~~D~~RPI) System.

The b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication ~~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 b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication ~~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 reliable~~accurate~~ indication of ~~actual control~~ rod position, but at a lower accuracy~~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 six steps~~. TA3.1-164 The RPI System is designed with an accuracy of ±5% (approximately 12 steps) of full rod travel. There are inaccuracies arising from the normal range of coolant temperature variation from hot shutdown to full power which are compensated for by allowing ± 24 steps at the lower and upper ends of rod travel. CL3.1-168 ~~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 DRPI will go on half~~

BACKGROUND ~~(continued)~~

~~accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches).~~ With an TA3.1-164

(continued)

BASES

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. At the lower and upper ends of rod travel with an indicated deviation of 24 steps between the group step counter and RPI, the deviation between actual rod position and the demand position could be 36 steps, or 22.5 inches. CL3.1-168

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment assure ~~are~~ that: CL3.1-137

a. There are ~~be~~ no violations of:

   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

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. 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.

(continued)

BASES

The safety ~~Two types of~~ analysis ~~are performed in~~ regarding ~~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 DNBR ~~departure from nucleate boiling ratio~~ in ~~both of these~~ this cases bounds the situation when a rod is misaligned from its group by 24~~12~~ steps. CL3.1-137

APPLICABLE SAFETY ANALYSES (continued)

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. 3~~5~~). CL3.1-137

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected, that the linear heat rates (LHRs) are not significantly affected, 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.

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 assumed power distribution used in the safety analysis may not be preserved ~~assumptions that are used to determine the rod insertion limits, AFD limits,~~ CL3.1-137

(continued)

BASES

~~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.

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) ~~the NRC Policy Statement~~. TA3.1-86

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 ~~also~~ 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. CL3.1-87

The rod alignment requirements are satisfied when individual actual rod positions are within 24 steps of their group step counter demand position when the demand position is between 30 and 215 steps, or within 36 steps of their group step counter demand position when the demand position is ≤ 30 steps, or ≥ 215 steps.

The requirement to maintain the rod alignment to within plus

(continued)

BASES

or minus 12 steps when the group step counter demand position is between 30 and 215 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.

LCO (continued)

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. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are normally 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 (SDM) ~~$- T_{avg} > 200°F$~~," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling. TA3.1-76

ACTIONS

A.1.1 and A.1.2

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 TA3.1-86

(continued)

BASES

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. PA3.1-171

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

A.2

TA3.1-86

If the inoperable~~untrippable~~ 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

ACTIONS

A.2 (continued)

this status, the unit must be brought to at least MODE 3 within 6 hours.

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.

PA3.1-88

~~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.~~

~~An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the~~

(continued)

BASES

~~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 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "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.~~2.~~1.1 and B.~~2.~~1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

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.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

ACTIONS

B.~~2.~~1.1 and B.~~2.~~1.2 (continued)

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.

(continued)

BASES

B 3.2.1.1, B 3.2.1.2, B.2.2, B.~~2-~~3, ~~B.2.4, B.2.5,~~ and B.~~42.6~~ CL3.1-89

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 $F_{\Delta H}^N$) must be verified within limits or reactor power must be reduced. SDM must periodically be verified within limits ~~and~~ the safety analyses must be re-evaluated to confirm continued operation is permissible, and, if necessary, the power level must be reduced to a level consistent with the safety analysis. Considerations in these analyses include the potential ejected rod worth and associated transient power distribution peaking factors.

~~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.~~

~~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.~~ TA3.1-90

Verifying that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within the required limits (i.e., SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1) 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 ~~7~~2 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 $F_{\Delta H}^N$. CL3.1-89

(continued)

BASES

In lieu of determining hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) within the Completion Time of 2 hours, reducing power to 85% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System. CL3.1-89

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. CL3.1-89

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analyses 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 Ref. 3 that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. TA3.1-164

If the analyses do not support continued operation at RTP, then the power must be reduced to a level consistent with the safety analyses.

A Completion Time of 805 days is sufficient time to obtain the required input data and to perform the analysis and adjust power level. CL3.1-89

C.1

(continued)

BASES

When Required Actions cannot be completed within their 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 3 within 6 hours, which eliminates 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 systems. TA3.1-172

ACTIONS (continued)

D~~C~~.1.1 and D~~C~~.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 for 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 initiate boration~~start the boric acid pumps~~. Boration will continue until the required SDM is restored. CL3.1-173

D~~C~~.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~~ CL3.1-174

(continued)

BASES

~~misalignment,~~ [T]~~t~~he 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 3 within 6 hours.

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.

~~D.1~~

~~When Required Actions cannot be completed within their 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 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~~ TA3.1-172

~~ACTIONS~~ ~~D.1 (continued)~~

~~conditions in an orderly manner and without challenging the plant systems.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.~~4~~5.1

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. ~~If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal.~~ The specified Frequency takes into account TA3.1-97

(continued)

BASES

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 3.1.4.1 is modified by a Note which direct the operators to Specification 3.1.7, "Rod Position Indication," if a rod appears to be misaligned by more than 12 steps. If the rod position indication is determined to be correct in accordance with Specification 3.1.7, then the operator must return to Specification 3.1.4 and enter the appropriate Conditions for rod misalignment. PA3.1-91

SR 3.1.4~~5~~.2 CL3.1-176

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, 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 providing rod alignment limits are not exceeded. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4~~5~~.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4~~5~~.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable ~~and aligned~~, 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) TA3.1-86 PA3.1-91

(continued)

BASES

of the control rod(s) must be made, and appropriate action taken.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4~~5~~.3

CL3.1-177

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 moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions. Actual rod drop time is measured from opening of the RTB which is conservative with respect to beginning of decay of stationary gripper coil voltage.

CL3.1-122

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.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits", Criteria 6, 14, 27, and 28, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~10 CFR 50, Appendix A, GDC 10 and GDC 26~~.

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants".

(continued)

BASES

3. UFSAR, Section~~Chapter~~ 14.4~~[15]~~.

~~4. FSAR, Chapter [15].~~

~~5. FSAR, Chapter [15].~~

~~6. FSAR, Chapter [15].~~

~~7. FSAR, Chapter [15].~~

TA3.1-76

PA3.1-121

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5~~6~~ Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and 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, SHUTDOWN MARGIN (SDM) and initial reactivity insertion rate. PA3.1-178

CL3.1-122

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC Criteria 27, 28, 29, and 32~~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. CL3.1-181

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Some~~Each~~ banks 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 that consists of two groups~~ that~~ are moved in a staggered fashion, but always within one step of each other. Each reactor has~~All plants have~~ four control banks and ~~at least~~ two shutdown banks. See LCO 3.1.4~~5~~, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7~~8~~, "Rod Position Indication," for position indication requirements.

(continued)

BASES

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.

BACKGROUND (continued)

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 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.

TA3.1-76

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

(continued)

BASES

control banks may be partially inserted in the core, as allowed by LCO 3.1.6~~7~~, "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 (SDM) ~~- T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F~~") 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 (547 °F), 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. PA3.1-182

The acceptance criteria for addressing shutdown and control rod bank insertion limits ~~and inoperability or misalignment~~ assure~~is~~ that: PA3.1-183

APPLICABLE SAFETY ANALYSES (continued)

a. There are ~~be~~ no violations of:

   1. specified acceptable fuel design limits, or
   2. RCS pressure boundary integrity; and

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).

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

(continued)

BASES

power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3). PA3.1-93

Operation at the insertion limits assures that the maximum linear heat generation rate or peaking factor will be less than that used in the misaligned rod analysis. Operation at the insertion limit also assures that the maximum ejected RCCA worth will be less than the limiting value used in the ejected RCCA analysis. TA3.1-94

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) ~~the NRC Policy Statement~~.

PA3.1-184

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. TA3.1-76

The shutdown bank insertion limits are defined in the COLR.

The LCO is modified by a Note indicating that a shutdown bank may be below the insertion limit when required for performance of 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.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. ~~The applicability in~~

(continued)

BASES

~~MODE 2 begins prior to initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown.~~ 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 bank~~s~~ insertion limit does not apply because the reactor is not producing fission power. In shutdown MODES the OPERABILITY of the shutdown 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~~are fully inserted in the core and contribute to the SDM~~. Refer to LCO 3.1.1 ~~and LCO 3.1.2~~ for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

~~APPLICABILITY (continued)~~

~~The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.2. 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.~~

PA3.1-93

ACTIONS

A.1.1, A.1.2 and A.2

With~~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

PA3.1-187

(continued)

BASES

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.

Operation beyond the LCO limits is allowed for a short time period in order to take appropriate action because the simultaneous occurrence of either an accident or transient 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 provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time. PA3.1-187

B.1

PA3.1-188

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, ~~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 systems.

PA3.1-191

SURVEILLANCE REQUIREMENTS

SR 3.1.5~~6~~.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,~~

(continued)

BASES

~~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~~ PA3.1-191

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.1.56.1 (continued)~~

~~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. CL3.1-122

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 27, 28, 29, and 32, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28~~.
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
3. UFSAR, Sections~~Chapter~~ 14.4 and 14.5~~[15]~~.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS TA3.1-76

B 3.1.6~~7~~ Control Bank Insertion Limits PA3.1-121

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and 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 SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate. 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. PA3.1-178

The applicable criteria for these reactivity and power distribution design requirements are AEC GDC 27, 28, 29, and 32 ~~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 ($K_{eff} \geq 1.0$) to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. CL3.1-122

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Some ~~Each~~ banks 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 that consists of two groups ~~that~~ are moved in a staggered CL3.1-181

(continued)

BASES

fashion, but always within one step of each other. Each reactor has ~~All plants have~~ four control banks and ~~at least~~ two shutdown banks. See LCO 3.1.4~~5~~, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7~~8~~, "Rod Position Indication," for position indication requirements. CL3.1-181 TA3.1-76

Insertion Limits

The control bank insertion limits are specified in a figure in the COLR. ~~An example is provided for information only in Figure B 3.1.7-1.~~ The control banks are required to be at or above the insertion limit lines. PA3.1-192

~~Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The predetermined~~

~~BACKGROUND (continued)~~ ~~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.~~ PA3.1-193

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. The fully withdrawn position is defined in the COLR. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. CL3.1-194

(continued)

BASES

Overlap and Sequence

CL3.1-196

The insertion limits Figure in the COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. By overlapping control bank movements, the small reactivity addition at the beginning and end of control bank travel will be compensated for; that is, the overlapping sequential movement of control banks makes the reactivity addition more uniform.

Control banks are moved in an overlap pattern, 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 fully withdrawn position, 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 near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. 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.

General

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together,

(continued)

BASES

LCO 3.1.4~~5~~, "Rod Group Alignment Limits," TA3.1-76
LCO 3.1.5~~6~~, "Shutdown Bank Insertion Limits,"
LCO 3.1.6~~7~~, "Control Bank Insertion Limits," 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 assures ~~will prevent~~ fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant will be bounded by the safety analysis results in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other transient~~accident~~ requiring termination by a Reactor Trip System (RTS) trip function. CL3.1-198

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.~~ PA3.1-183
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

(continued)

BASES

amount of negative reactivity on a reactor trip signal. 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 (SDM)") 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 (547 °F), and to maintain the required SDM at rated no load temperature (Ref. 3). The control bank insertion limit also limits the reactivity worth of an ejected control rod.

APPLICABLE SAFETY ANALYSES (continued)

PA3.1-182

The acceptance criteria for addressing shutdown and control bank insertion limits ~~and inoperability or misalignment~~ assure ~~are~~ that:

a. There are ~~be~~ no violations of:

1. specified acceptable fuel design limits, or

2. Reactor Coolant System pressure boundary integrity; and

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).

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

(continued)

BASES

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. 3~~4~~).

Operation at the insertion limits assures that ~~or AFD limits may approach~~ the maximum ~~allowable~~ linear heat generation rate or peaking factor will be less than that used in the misaligned rod analysis ~~with the allowed QPTR present~~. Operation at the insertion limit ~~may~~ also assures that ~~indicate~~ the maximum ejected RCCA worth will be less than ~~could be equal to~~ the limiting value used in the ~~in fuel cycles that have sufficiently high~~ ejected RCCA analysis ~~worths~~. PA3.1-183

~~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).~~

The control bank insertion, sequence and overlap limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement~~, in that they are initial conditions assumed in the safety analysis. PA3.1-196

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 limited ~~maintained~~, and ensuring adequate PA3.1-201

LCO (continued)

negative reactivity insertion is available on a trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to

(continued)

BASES

maintain acceptable power peaking during control bank motion.

The LCO is modified by a Note indicating that a control bank may be below the insertion limit when required for performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would normally violate the LCO. PA3.1-93

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, and SDM~~, and reactivity rate insertion assumptions~~. Applicability in MODE~~S~~ 2 with $K_{eff} < 1.0$ and in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. PA3.1-202 PA3.1-203

~~The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.5.2. 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.~~ PA3.1-93

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

b. Moving rods to be consistent with power.

(continued)

BASES

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 is normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~- T_{avg} > 200°F~~") ~~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. PA3.1-204 TA3.1-76

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

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 an accident or transient ~~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. PA3.1-206

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 to remain in an unacceptable condition for an extended period of time.

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant

(continued)

BASES

must be brought to MODE 2 with K_{eff} < 1.0~~3~~, 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 systems. TA3.1-95

SURVEILLANCE REQUIREMENTS

SR 3.1.6~~7~~.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. Prior to achieving criticality, the estimated critical position calculation appropriate for the time at which criticality is achieved shall be verified for control bank position. CL3.1-207

~~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~~ CL3.1-207

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.1.67.1 (continued)~~

~~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~~7~~.2

~~With an OPERABLE bank insertion limit monitor,~~ Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ~~ensure OPERABILITY of the bank insertion limit monitor and to~~ TA3.1-97

BASES

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detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. ~~If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.~~ TA3.1-97

This verification may be performed manually by an operator or through a computer insertion limit monitoring program. PA3.1-208

SR 3.1.6~~7~~.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6~~7~~.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~~7~~.2. This verification may be performed manually by an operator or through a computer sequence and overlap monitoring program. PA3.1-208

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 27, 28, 29, and 32, issued for comment July 10, 1967, as referenced in USAR Section 1.2~~10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28~~. CL3.1-122

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors".

3. UFSAR, Sections~~Chapter~~ 14.4 and 14.5 ~~[15]~~.

BASES

~~4. FSAR, Chapter [15].~~

~~5. FSAR, Chapter [15].~~

BASES

~~This Figure is deleted~~

PA3.1-192

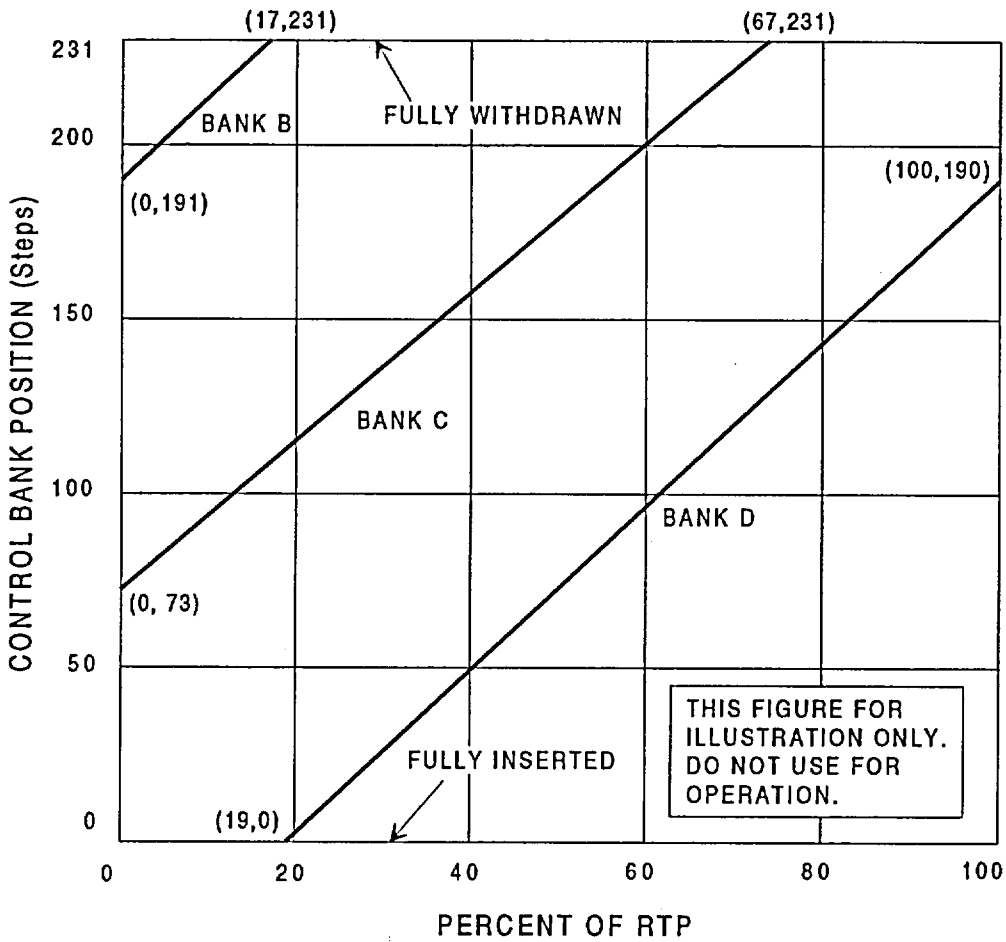

Figure B 3.1.7-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

TA3.1-76

PA3.1-121

CL3.1-162

CL3.1-163

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.78 Rod Position Indication

BASES

BACKGROUND

According to AEC GDC Criteria 12 and 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 3.1.78 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. CL3.1-122

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 SHUTDOWN MARGIN (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

(continued)

BASES

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.

BACKGROUND (continued)

The axial positions of shutdown rods and control rods are determined by two separate and independent systems: the b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication ~~System~~ (commonly called group step counters) and the individual ~~[Digital]~~ Rod Position Indication (~~D~~RPI) System.

The b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication ~~System~~ counts the pulses from the Rod Control System that move 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 b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication ~~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 reliable ~~accurate~~ indication of actual control rod position, but at a lower accuracy ~~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 DRPI will go on half accuracy with an~~ CL3.1-168

(continued)

BASES

~~effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches).~~ The RPI System is designed with an accuracy of ±5% (approximately 12 steps) of full rod travel. There are inaccuracies arising from the normal range of coolant temperature variation from hot shutdown to full power which are compensated for by allowing ± 24 steps at the lower and upper ends of rod travel. 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. At the lower and upper ends of rod travel with an indicated deviation of 24 steps between the group counter and RPI, the deviation between actual rod position and the demand position could be 36 steps, or 22.5 inches. CL3.1-168

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 3.1.56, "Shutdown Bank Insertion Limits," and LCO 3.1.67, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.45, "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.

APPLICABLE SAFETY ANALYSES (continued)

(continued)

BASES

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement~~. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7~~8~~ specifies that the ~~one~~ ~~D~~RPI System and ~~one~~ b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication~~ 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:

a. The ~~D~~RPI System indicates within 12 steps of the group step counter demand position when the demand position is between 30 and 215 steps, or within 24 steps of their group step counter demand position when the demand position is greater than or equal to 215 steps, or less than or equal to 30 steps ~~as required by LCO 3.1.5, "Rod Group Alignment Limits"~~; CL3.1-212

b. ~~For the DRPI System there are no failed coils; and~~

~~c.~~ ~~The~~ Bank d~~D~~emand i~~I~~ndication~~ System~~ has been calibrated either in the fully inserted position or to the ~~D~~RPI System. Demand position indication may be provided by various means such as step counters, Emergency Response Computer System (ERCS), calculations using rod drive cabinet counters or Pulse to Analog counters. CL3.1-212

The 12 step agreement limit between ~~the~~ b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication~~ System~~ and the ~~D~~RPI System when the demand position is between 30 and 215 steps indicates that the b~~B~~ank d~~D~~emand p~~P~~osition i~~I~~ndication CL3.1-213

(continued)

BASES

~~System~~ is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given above ~~in LCO 3.1.5~~, 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.

LCO (continued)

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 3.1.45, LCO 3.1.56, and LCO 3.1.67), 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. 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. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODE 2 with K_{eff} < 1.0 and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

PA3.1-214

(continued)

BASES

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable RPI ~~rod position indicator per group~~ and each demand position indicator ~~per bank~~. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator. TA3.1-99

A.1

CL3.1-216

When one ~~D~~RPI channel per group fails, the position of the rod may~~can~~ still be determined indirectly by use of the moveable incore ~~movable~~ detectors. Based on experience, normal power operation does not require excessive movement of banks. ~~If a bank has been significantly moved, the Required Action of B.1 or B.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. Verification may determine that the RPI is OPERABLE and the rod is misaligned, then the Conditions of 3.1.4, "Rod Group Alignment Limits" must be entered. PA3.1-217

ACTIONS (continued)

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. 3)~~.

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

(continued)

BASES

and allowing for rod position determination by Required Action A.1 above.

B.1, B.2, B.3, and B.4

CL3.1-101

When more than one RPI channel per group fails, additional monitoring shall be performed to assure that the reactor remains in a safe condition. The demand position from the group step counters associated with the rods with inoperable position indicators shall be monitored and recorded on an hourly basis. This ensures a periodic assessment of rod position to determine if rod movement in excess of 24 steps has occurred since the last determination of rod position. If rod movement in excess of 24 steps has occurred since the last determination of rod position, the Required Action of B.3 below is required.

The reactor coolant system average temperature shall be monitored and recorded on an hourly basis. Monitoring and recording of the reactor coolant system average temperature may provide early detection of mispositioned or dropped rods.

When one or more rods have been moved in excess of 24 steps in one direction, since the position was last determined, action is initiated sooner to begin verifying that these rods are still properly positioned relative to their group positions. The four hour allowance for completion of this action allows adequate time to complete the rod position verification using the moveable incore detectors.

The position of rods with inoperable RPIs will also continue to be verified indirectly using the moveable incore detectors every 8 hours in accordance with Required Action A.1. Using the moveable incore detectors provides further assurance that the rods have not moved.

(continued)

BASES

Based on experience, normal power operation does not require excessive movement of banks. Therefore, the actions specified in this condition are adequate for continued full plant operation for up to 24 hours 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 allowed out of service time also provides sufficient time to troubleshoot and restore the RPI system to operation following a component failure in the system, while avoiding the challenges associated with a plant shutdown. CL3.1-101

CL3.1-101

~~B.1 and B.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 are still appropriate but must be initiated promptly under Required Action B.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.~~

C.1.1 and C.1.2 CL3.1-221

Demand position indication is provided by any of the following means: step counters, Emergency Response Computer

(continued)

BASES

System (ERCS), calculations using rod drive cabinet counters and Pulse to Analog counters. With all indication for 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 administrative means that the rod position indicators are OPERABLE and the RPI of the most withdrawn rod and the RPI of the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate. This ensures that the most withdrawn and least withdrawn rod are no more than 24 steps apart (including instrument uncertainty) which bounds the accident analysis assumptions. This verification can be an examination of logs, administrative controls, or other information that shows that all RPIs in the affected bank are OPERABLE. PA3.1-222

ACTIONS (continued)

C.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)~~. 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.

D.1

If the Required Actions 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 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 systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.~~7~~8.1

Verification that the ~~D~~RPI agrees with the demand position within ~~[12]~~ steps (between 30 and 215 steps) or within 24 steps (when ≤ 30 steps or ≥ 215 steps) ensures that the ~~D~~RPI is operating correctly. ~~Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~ PA3.1-213 CL3.1-223

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. ~~The [18 month] Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for unnecessary plant transients if the SR were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at a Frequency of once every [18 months.] Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ TA3.1-102

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits" Criteria 12 and 13, issued for comment July 10, 1967, as referenced in USAR Section 1.2 ~~10 CFR 50, Appendix A, GDC 13~~. CL3.1-122

2. UFSAR, Sections ~~Chapter~~ 14.4 and 14.5 ~~[15]~~.

~~3. FSAR, Chapter [15].~~

B 3.1 REACTIVITY CONTROL SYSTEMS

TA3.1-76

B 3.1.~~10~~8 PHYSICS TESTS Exceptions – MODE 2

TA3.1-103

PA3.1-121

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. ~~Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).~~ PA3.1-224 PA3.1-226

The key objectives of a test program are to ~~(Ref. 3)~~: PA3.1-227

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.

(continued)

BASES

BACKGROUND (continued)

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. 1~~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.

The PHYSICS TESTS required for reload fuel cycles (Ref. 1~~-4~~) in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Critical Boron Concentration – Control Rods Inserted;

c. Control Rod Worth; and

d. Isothermal Temperature Coefficient (ITC).~~; and~~

~~e. Neutron Flux Symmetry.~~ CL3.1-228

~~The first 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~~ Low power physics tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

(continued)

BASES

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 D 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 could violate LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)" ~~should not violate any of the referenced LCOs~~. CL3.1-231

b. The Critical Boron Concentration – Control Rods Inserted Test measures the critical boron concentration at HZP, with the highest worth rod bank ~~a bank having a worth of at least 1% Δk/k when~~ fully inserted into the core. This test is used to give an indication of ~~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 CL3.1-232

(continued)

BASES

BACKGROUND (continued)

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 gives an indication of the Boron Reactivity Coefficient compared to the measured bank worth. ~~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 LCO 3.1.4~~5~~, "Rod Group Alignment Limits"; LCO 3.1.5~~6~~, "Shutdown Bank Insertion Limit"; or LCO 3.1.6~~7~~, "Control Bank Insertion Limits." CL3.1-232

c. The Control Rod 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 Dilution ~~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 Dilution ~~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 PA3.1-233

(continued)

BASES

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. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4~~5~~, LCO 3.1.5~~6~~, or LCO 3.1.6~~7~~.

CL3.1-234

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP using the Slope Method ~~and has two methods of~~

BACKGROUND (continued)

~~performance. The first method, the~~ 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 ITC at BOC, 70% RTP and at EOC is determined from the ITC measured in this test. This test satisfies the requirements of SR 3.1.3.1, SR 3.1.3.2 and SR 3.1.3.3. ~~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."

CL3.1-234

~~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~~

CL3.1-236

(continued)

BASES

~~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 3.1.5, LCO 3.1.6, or LCO 3.1.7. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.~~ CL3.1-236

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 of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5).~~ The above mentioned PHYSICS TESTS~~, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational~~ CL3.1-237

APPLICABLE SAFETY ANALYSES (continued)

~~problems,~~ may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. USAR Appendix J ~~Tables [14.1-1 and 14.1-2]~~ summarizes the initial plant startup zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in Reference 1 ~~ANSI/ANS-19.6.1-1985 (Ref. 4)~~. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, CL3.1-238

(continued)

BASES

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~~ T~~t~~he requirements specified in the following LCOs may be suspended for PHYSICS TESTING:

LCO 3.1.3~~4~~, "Isothermal ~~Moderator~~ Temperature Coefficient (IMTC)~~,~~"; TA3.1-241
LCO 3.1.4~~5~~, "Rod Group Alignment Limits";
LCO 3.1.5~~6~~, "Shutdown Bank Insertion Limits";
LCO 3.1.6~~7~~, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality". ~~are suspended for PHYSICS TESTS.~~

When these LCOs 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 ≥ 535~~1~~°F, and SDM is within the limits provided in the COLR ~~≥ [1.6]% Δk/k~~. TA3.1-77

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 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. ~~PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the~~ PA3.1-243

(continued)

BASES

~~components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.~~

~~Reference 6 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 IMTC and minimum temperature for criticality to be outside their specified limits to conduct PHYSICS TESTS in MODE 2, to verify certain core physics parameters. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from "4" to "3". 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. PA3.1-244 TA3.1-111

LCO (continued)

The requirements of LCO 3.1.3~~4~~, LCO 3.1.4~~5~~, LCO 3.1.5~~6~~, LCO 3.1.6~~7~~, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels during the performance of PHYSICS TESTS provided: TA3.1-111

a. RCS lowest loop average temperature is ≥ ~~[~~535~~1]~~ °F; ~~and~~

b. SDM is within the limits provided in the COLR; and ~~≥ [1.6]% Δk/k.~~ TA3.1-77

(continued)

BASES

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c. THERMAL POWER is ≤ 5% RTP. TA3.1-105

APPLICABILITY

This LCO is applicable ~~in MODE 2~~ when performing low power PHYSICS TESTS. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions. ~~The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions – MODE 1."~~ TA3.1-106 TA3.1-103

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 conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour. PA3.1-246

B.1

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

(continued)

BASES

prevent operation of the reactor outside of its design limits.

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 535~~1~~°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 to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 535~~1~~°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If ~~the~~ Required Action~~s~~ C.1 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 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 systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8~~10~~.1

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 performed on each power range and intermediate range channel ~~within 12 hours~~ prior to initiation of the PHYSICS TESTS. This will ensure that the TA3.1-107

(continued)

BASES

RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. ~~The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.~~

SR 3.1.8~~10~~.2

Verification that the RCS lowest loop T_{avg} is ≥ 535~~1~~°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

SURVEILLANCE REQUIREMENTS

SR 3.1.8~~10~~.2 (continued)

performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant 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. TA3.1-105

SR 3.1.8~~10~~.4~~3~~

Prior to achieving criticality, t~~T~~he SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects: CL3.1-247

(continued)

BASES

a. RCS boron concentration;

b. Control and shutdown bank position; CL3.1-247

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration~~; and~~

~~g. Isothermal temperature coefficient (ITC)~~.

~~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.~~

After achieving criticality, this SR is met by determining the reactivity insertion available from tripping the shutdown and control banks. CL3.1-247

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.~~ PA3.1-224

~~2. 10 CFR 50.59.~~ PA3.1-226

~~3. Regulatory Guide 1.68, Revision 2, August, 1978.~~ PA3.1-227

~~4.~~ ANSI/ANS-19.6.1-1985, Reload Startup Physics Tests for Pressurized Water Reactors~~, December 13, 1985~~. CL3.1-238

BASES

~~REFERENCES (continued)~~	~~5.~~	~~WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.~~
	~~6.~~	~~WCAP-11618, including Addendum 1, April 1989.~~

CL3.1-237

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART F

JUSTIFICATION FOR DIFFERENCES (JFD)

from

NUREG-1431 IMPROVED STANDARD TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

PART F

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

JUSTIFICATION FOR DIFFERENCES FROM IMPROVED STANDARD TECHNICAL SPECIFICATIONS (NUREG-1431) AND BASES

See Part E for specific proposed wording and location of referenced deviations.

Difference Category	Difference Number 3.1-	Justification for Differences
TA	76	This change incorporates TSTF-136.
TA	77	This change incorporates TSTF-9, Rev. 1. The PI ITS wording differs from the TSTF in that "is" is used in lieu of "to be" to make the requirements read better in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6 and 3.1.8 and their Bases.
	78	Not used.
TA	79	This change incorporates TSTF-142.
	80	Not used.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	81	ISTS SR 3.1.3.1 and the associated SR section of the Bases were revised to divide the SR into two SRs (ITS SR 3.1.2.1 and SR 3.1.2.2). The Note in the Frequency column as to when the SR is required to be performed was moved to the Surveillance column as a Note. This change clarifies the Notes that modify the SRs, eliminates the potential for the misapplication of the usage rules relating to the Frequency, and improves the readability and understanding of the SRs. "Once" has not been included in the Frequency for SR 3.1.2.1 since this is understood because there is not a recurring frequency for this SR.
CL	82	PI CTS requirements are all based on isothermal temperature coefficient and therefore the ITS requirements are also based on isothermal temperature coefficient (ITC). This is not a significant change since the ITC and moderator temperature coefficient (MTC) are numerically related to each other. The nomenclature has been changed to ITC throughout this specification and the Bases (except where the term MTC is still applicable).
CL	83	The PI CTS specific ITC limits have been included explicitly in the LCO in lieu of reference to a figure. Thus, NUREG-1431 Figure 3.1.4.1 is not included in the PI ITS. These ITC limits have been approved for use at PI and assure that the plant operates safely at all power levels. The presentation of the LCO statement has been revised for clarity.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	84	PI does not currently have TS requirements to monitor the ITC lower limit during the core operating cycle. The provisions of ISTS LCO 3.1.4 have been replaced by the proposed LCO 3.1.3 Action Statements C and D and SR 3.1.3.3 and their associated Bases which more closely address current plant practices of predicting EOC ITC. This change is consistent with the approved GITS.
PA	85	PI does not currently have TS requirements to verify ITS limits within 300 ppm as specified in ISTS SR 3.1.4.2. The provisions of ISTS 3.1.4.2 have been replaced by the proposed PI ITS SR 3.1.3.2 which more closely address current plant practices of confirming ITC will be within its limits at 70% RTP. This change is consistent with the approved GITS. Approved TSTF-13 has not been incorporated, since it is incompatible with the changes made to ISTS SR 3.1.4.2 to accommodate PI current practices.
TA	86	This change incorporates TSTF-107. The logical connector "AND" format is not used since this is not consistent with the description of Logical Connectors provided in ISTS Section 1.2.
CL	87	CTS only requires action when a control rod misalignment exceeds 24 steps. At the upper and lower limits of rod travel, the rod is misaligned when it deviates by 36 steps.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	88	Required Action B.1 was not included in the PI ITS since, in accordance with the Writer's Guide, 4.1.6.g, "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." The logic of these Required Actions is simplified by not including B.1. This change is also consistent with proposed TSTF-240.
CL	89	PI CTS requires performance of SRs within 2 hours to assure acceptable core power distribution or reduce power to 85% RTP. ITS Specification 3.1.4 Action Statement B and the associated Bases have been revised to incorporate these CTS requirements. Also for consistency with CTS, the power level for which the rod misalignment is acceptable, is re-evaluated and the power is adjusted to this re-evaluated power level within 30 days. CTS 3.10.G.5 considerations for the analyses have been relocated to this section of the Bases.
TA	90	This change incorporates TSTF-314.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	91	A note is included in PI ITS SR 3.1.4.1 which clarifies the required PI approach for dealing with indications of rod misalignment. At PI when a rod appears to be misaligned, the operator will first verify operability of the RPI within the requirements of PI ITS 3.1.7. If the RPI is operable, then the rod will be treated as misaligned under the requirements of this Specification 3.1.4.
CL	92	This change incorporates CTS requirement for rods to drop in 1.8 seconds with both RCPs operating. Since PI is a two loop plant, "both" is used in lieu of "all".
PA	93	The Note in the Applicability section of the ISTS has been moved to the LCO section of the ITS LCOs since the Note applies to exceptions to the entire LCO and not to exceptions to the Applicability. These changes make the Note and its use clearer for the plant operators. The associated changes to the Bases were also made.
TA	94	This change incorporated TSTF-239.
TA	95	This change incorporates TSTF-238.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	96	The PI ITS does not include the requirement to perform this SR "within 4 hours" prior to achieving criticality. In accordance with current plant practices, the estimated critical control bank position is prepared for all possible startup times such that Xenon decay is not a factor. Therefore, the requirement to perform this verification "within 4 hours" is not necessary. Since this is a new SR and is not in the PI CTS, this is a plant specific change.
TA	97	This change incorporates TSTF-110, Revision 2.
CL	98	PI does not have a "Digital" rod position indication system, thus the bracketed term "Digital" is not included in the PI ITS. This change has been made throughout the Specification and associated Bases. Also "DRPI" has been changed to "RPI". Since this is the system used in the plant and specified in the CTS, this a current licensing basis change.
TA	99	This change incorporates TSTF-234. This is an editorial change which deletes the modifying clauses "per group" and "per bank". These terms are unnecessary since these limitations are defined in the applicable Conditions.

Difference Category	Difference Number 3.1-	Justification for Differences
	100	Not used.
CL	101	NUREG-1431 Specification 3.1.8 (ITS 3.1.7) and the Bases are revised to incorporate CTS LAs 139/130. LAs 139/130 incorporated the provisions of TSTF-234 as appropriate for PI and as agreed upon with the NRC Staff.
TA	102	This change incorporates TSTF-89.
TA	103	This change incorporates both TSTF-12, Revision 1 and TSTF-136 which result in the Specification number changing from ISTS 3.1.10 to ITS 3.1.8.
PA	104	The lowest temperature for which PI reactor criticality is considered in safety evaluations is 535°F; thus, this temperature is provided in lieu of the bracketed term.
TA	105	This change incorporates TSTF-14, Revision 1.
TA	106	This change incorporates TSTF-256.
TA	107	This change incorporates TSTF-108.

Difference Category	Difference Number 3.1-	Justification for Differences
X	108	CTS do not include specifications for demand position indication. ITS includes a condition which addresses when all demand position indication is inoperable. The Bases clarify that demand position indication includes the step counters, Pulse to Analog counters, plant process computer system and calculations based on the rod control cabinet counters. This change is made to accommodate PI specific requirements which are not currently in the TS.
	109	Not used.
	110	Not used.
TA	111	This change incorporates TSTF-315. The reference to LCO 3.3.1, Function 18.e has been changed to 16.e to be consistent with the PI ITS.
	112	Not used.
	113	Not used.
	114	Not used.
	115	Not used.
	116	Not used.
	117	Not used.

Difference Category	Difference Number 3.1-	Justification for Differences
	118	Not used.
	119	Not used.
	120	Not used.
PA	121	Included throughout the Bases are reference corrections, renumbering and relettering of paragraphs and minor wording changes which have been made to accommodate changes to the Specifications and Prairie Island (PI) unique needs. These changes are not identified by change numbers.
CL	122	Reference to the General Design Criteria (GDC) contained in 10CFR50 Appendix A is replaced by reference to the Atomic Energy Commission (AEC) proposed GDC which is the PI licensing basis. PI was licensed to the proposed AEC GDC which pre-dated the 10CFR50 App A GDC. Some text changes have been made in some locations to conform to the actual requirements of the AEC GDC.
PA	123	Verbiage from the definition of SDM, ". . . and the fuel and moderator temperatures are changed to the nominal hot zero power temperature, 547°F," is included to make this paragraph clearer for the operators. This change is consistent with the approved GITS.

Difference Category	Difference Number 3.1-	Justification for Differences
CL	124	The Applicable Safety Analyses discussion has been modified to agree with the assumptions and results associated with the PI specific analyses.
	125	Not used.
PA	126	To be consistent with LCO 3.1.1 as modified by approved traveler TSTF-9, clarification is provided that the SDM requirements are specified in the COLR.
CL	127	The phrase, "the DNBR limit and to exceed," is not included since this is not accurate for PI. The PI MSLB analysis allows the fuel to exceed DNB limits and fail fuel.
CL	128	Since 15 minutes is a short time, the phrase, "and the probability of a DBA occurring during this time is very low." was included. This statement is true, consistent with other ITS Bases and provides further basis for allowing some time for operator action.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	129	Under these conditions the operator should borate with the best source available. The discussion of the BAST having high concentration boron may not be true for PI in the future. PI does not have a borated water storage tank. Thus the phrase, ". . .boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the borated water storage tank." have not been included.
	130	Not used.
PA	131	The example of boration rates is not included. It is sufficient that the operators borate with their best source. This example does not provide any further operator guidance or illusory information and therefore is not included.
CL	132	The Bases discussion for SR 3.1.1.1 was revised to describe the methods and considerations by which PI determines that the SDM limits are met.
	133	Not used.
PA	134	The clause, "or stable" was included since some of these parameters are not really fixed, but are stable for some time period.

Difference Category	Difference Number 3.1-	Justification for Differences
	135	Not used.
	136	Not used.
CL	137	The Applicable Safety Analyses discussion has been modified to reflect the analyses methods used at PI and the manner in which these analyses are used.
PA	138	The NUREG-1431 terminology, "beginning of cycle" has been replaced with "early in the cycle" to be consistent with the analyses.
	139	Not used.
	140	Not used.
CL	141	The clause, "An SDM demonstration" has been replaced with, "Verification of measured core reactivity (SR 3.1.2.1)" since PI will not perform a demonstration but will instead verify core reactivity.
	142	Not used.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	143	The discussion of actions in MODE 3 have not been included since this Specification is not applicable in MODE 3. When MODE 3 is entered, the appropriate specification will govern the required actions as necessary.
CL	144	Discussion from the PI CTS is included to provide background on the relationship between ITC and MTC since both terms are used in this Bases.
	145	Not used.
PA	146	The fourth paragraph of this Bases Background has been modified to make it accurate and consistent with the PI use of ITC.
PA	147	The next to last paragraph of the Bases Background discussion has been revised and relocated to the Bases LCO. This paragraph is better situated in the Bases LCO since it discusses the LCO limits and will help define the operability requirements.
PA	148	The last paragraph of the Bases Background is not included in the PI ITS since this discussion relates to the SRs. The essence of the paragraph is included in the Bases discussion for SR 3.1.3.1 as applicable to PI.

Difference Category	Difference Number 3.1-	Justification for Differences
	149	Not used.
	150	Not used.
CL	151	B 3.1.3 Bases Applicable Safety Analyses discussion of core overheating accidents was modified to be accurate for PI. The "loss of main feedwater flow" accident was deleted and the applicable accidents list was clarified.
	152	Not used.
CL	153	The next to last paragraph of B 3.1.3 Bases Applicable Safety Analyses discussion was not included since the SR is not included in the Specification and PI will not be making this measurement.
CL	154	The clause, "and will be within limits at 70% RTP, full power, and EOC" to make it clear when we intend to make our checks. This change supports the Specifications changes.
	155	Not used.
CL	156	This statement is a CTS requirement that is relocated to the ITS Bases.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	157	B 3.1.3 Bases Applicability discussion was modified to make it clearer as to how ITC changes as burnup increases and when the accidents are evaluated.
PA	158	A new paragraph was included in B 3.1.3 Bases Action A.1 to provide further clarification of the role of the ITC limits.
PA	159	Clarification is provided on which ITC limit applies, when it will be violated and the purpose of administrative withdrawal limits.
	160	Not used.
CL	161	The SR Bases discussion was revised to reflect the use of ITC and incorporate changes to reflect how the SRs will be used at PI.
CL	162	PI does not have a Bank Demand Position Indication System, as such, and does not have any CTS for bank demand position indication. Thus the name has been changed to lower case and the term "System" has been deleted.
CL	163	PI does not have a "Digital" RPI System, thus "digital" and "D" are not included with "RPI".

Difference Category	Difference Number 3.1-	Justification for Differences
TA	164	These changes incorporate TSTF-331. Some changes were not incorporated since there are unique PI design features which need to be presented to support the LCO statement.
	165	Not used.
PA	166	The clause, "and OPERABILITY" was included for clarity by making this statement consistent with the Specification LCO statement.
	167	Not used.
CL	168	The discussion of the RPI system has been revised to accurately describe the system as installed at PI.
	169	Not used.
	170	Not used.
PA	171	The term "emergency" is not included since the Specification does not require use of emergency boration and the operators may use any appropriate boration method available.
TA	172	This change incorporates TSTF-15.

Difference Category	Difference Number 3.1-	Justification for Differences
CL	173	The clause, "start the boric acid pumps" has been replaced with "initiate boration" since the operators may take actions other than starting boric acid pumps to satisfy performance of the Required Actions.
CL	174	The portion of the sentence, "Since automatic bank sequencing would continue to cause misalignment" has not been included since this may not always be true.
	175	Not used.
CL	176	A clause which reads, "providing rod alignment limits are not exceeded." is included to reinforce with the operators that there are limits on the control movement requirement associated with satisfaction of this SR.
CL	177	Since the terminology for this SR and Bases discussion differs from the common PI terminology, a clarifying sentence has been included to assure that the current PI practices are allowed.
PA	178	For completeness and clarity, a sentence was added which defines what insertion limits do for the plant.
	179	Not used.

Difference Category	Difference Number 3.1-	Justification for Differences
	180	Not used.
CL	181	The NUREG-1431 Bases have been modified to accommodate the PI design in which some RCCA banks have 2 groups and some have only one group.
PA	182	The phrase, "and inoperability or misalignment" has not been included since these conditions are not within the subject of this specification. They are addressed in other specifications.
PA	183	NUREG-1431 Specifications 3.1.6, "Shutdown Bank Insertion Limits," and 3.1.7, "Control Bank Insertion Limits," have very similar purposes and design features. Therefore, many of the discussions in either Bases for these specifications apply to both specifications. Thus, two paragraphs from NUREG-1431, B 3.1.7 are included here to make the discussion complete.
PA	184	For clarity, more discussion was included in the Bases Applicability on the role of the shutdown banks in the SHUTDOWN MODES.
	185	Not used.
	186	Not used.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	187	The discussion about 2 hours to restore the shutdown banks within insertion limits is not included. While it may be true that the SDM may be significantly reduced, this not a basis for allowing 2 hours to restore shutdown banks within their insertion limits. Thus this discussion is not included. A new sentence is included later in this Actions discussion which provides the basis for short term operation beyond the LCO limits.
PA	188	Since there are more actions than just restoration of the shutdown banks to within their insertion limits, this Action Statement was modified for clarity.
	189	Not used.
	190	Not used.
PA	191	Unnecessary verbiage has been removed which improves the accuracy of the discussion since the SR does not specify verification of insertion limits prior to an approach to criticality.
PA	192	NUREG-1431 Figure B 3.1.7 is not included and all references to this figure were not included in the PI ITS. Since the actual limits are in the COLR, use of a figure could cause operator confusion.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	193	The detailed discussion of the steps at which the banks begin to move is not included since this level of detail is unnecessary in the Bases.
CL	194	To make the insertion limits background complete, discussion of the normal power operation control bank position, location of the definition for fully withdrawn and use of boration is included.
	195	Not used.
CL	196	The LCO statement for Specification 3.1.6, "Control Bank Insertion Limits," requires "sequence and overlap" limits to be met. Since the Bases provide very little background on the purpose of these limits, discussion has been added to the PI ITS. Also, where appropriate, "sequence and overlap" have been added to other discussions.
	197	Not used.
CL	198	The last paragraph of the B 3.1.6 Bases Background has been modified to make the discussion accurate on what the LCO limits will do for PI with respect to the safety analyses.
	199	Not used.
	200	Not used.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	201	The Bases LCO discussion of ejected rod worth has been modified since it is not "maintained", it is "limited".
PA	202	Since the control bank insertion limits do not preserve the "reactivity rate insertion assumptions", this clause is not included in the PI ITS.
PA	203	NUREG-1431 does not address "MODE 2 with Keff <1.0", thus this has been included.
PA	204	This sentence is awkward and unclear with the verb dangling at the end; thus, it has been reworded.
	205	Not used.
PA	206	The discussion of accident and transients was generalized since this detailed list is unnecessary in the Bases and may be inaccurate in the future.
CL	207	The NUREG-1431 discussion of estimated critical position calculations has been replaced by a description of how PI actually handles these calculations.
PA	208	Clarification is provided on acceptable means for performing this required verification.

Difference Category	Difference Number 3.1-	Justification for Differences
	209	Not used.
	210	Not used.
	211	Not used.
CL	212	The RPI requirements for OPERABILITY have been rewritten to incorporate the current plant requirements for this system. (These requirements have been determined through discussions with the NRC and were documented in NCR 19970613.)
CL	213	At PI, the bank demand position indication accuracy of 12 step agreement is only valid over the range of 30 to 215 steps; thus, for consistency with CTS and completeness, this range has been defined. Outside this range, the accuracy is 24 steps which has been stated when appropriate.
PA	214	Since the ISTS briefly discusses SDM, the SDM specifications have been referenced for a more complete discussion of SDM.
	215	Not used.
CL	216	The statement regarding use of incore detectors has been revised to be consistent with the CTS and Bases discussion of the capabilities of these detectors.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	217	Clarification is provided that if the actions determine that the rod is misaligned, then the appropriate specification for misaligned rods must be entered.
	218	Not used.
	219	Not used.
	220	Not used.
CL	221	PI does not have a single defined TS required demand position indication system. AT PI demand position can be determined by a number of methods. The ITS Specification 3.1.7 requirement to have demand position indication OPERABLE is a new requirement and can be met by any of the various methods which are listed in this Bases.
PA	222	Clarification is provided on how far the most and least withdrawn rods can be apart and acceptable means for verification. These clarifications are essential to provide the operators with sufficient guidance.
CL	223	Discussion of the actual shutdown rod positions is not included since this is not true at PI.

Difference Category	Difference Number 3.1-	Justification for Differences
PA	224	Reference 1 was not included since this reference is not used again and a full description of the reference is included in the text.
	225	Not used.
PA	226	Since 10CFR50 does not specify notification requirements, this sentence has not been included and reference 2 is not included.
PA	227	Reference 3 is not included since PI is not committed to Regulatory Guide 1.68.
CL	228	The list of Physics Tests has been customized to be consistent with current requirements for PI.
	229	Not used.
	230	Not used.
CL	231	This discussion was clarified to specify that Bank D is nearly fully withdrawn and that the Critical Boron Concentration Test could violate ITC specifications.
CL	232	The discussion of Critical Boron worth testing was clarified to be consistent with PI test requirements and terminology.

Difference Category	Difference Number 3.1-	Justification for Differences
CL	233	The test "Boron Exchange Method" was changed to "Boron Dilution Method" to be consistent with plant procedures.
CL	234	PI uses a single test for determining ITC, the Slope Method; thus, this discussion has been revised accordingly.
	235	Not used.
CL	236	PI does not use the Flux Symmetry Test, thus this discussion is not included in the PI ITS.
CL	237	PI does not use the Westinghouse Reload Safety Evaluation Methodology and thus reference to this methodology is not included. The Base Background discussion addresses all of the physics testing which is performed at PI; thus, the clause referring to "other tests that may be required . . ." is not included.
CL	238	The basis for PI initial plant testing is USAR Appendix J and therefore this paragraph was modified accordingly. ANSI/ANS - 19.6.1 was made Reference 1 and details of this reference are not included in the text since it is adequately described in the Bases References Section.

Difference Category	Difference Number 3.1-	Justification for Differences
	239	Not used.
	240	Not used.
PA	241	For consistency, the titles for all of the specifications were listed rather than just the first one.
	242	Not used.
TA	243	This change incorporates TSTF-154, Revision 2.
PA	244	The purpose of this specification was elaborated to make it clear that it is for testing in MODE 2.
	245	Not used.
PA	246	Clarification was provided that the 1 hour in the specification applies.
CL	247	The discussion of the Bases for this SR was modified to be consistent with the reactivity effects considered at PI and the point in time at which the activities are performed. Since "Isothermal temperature coefficient (ITC)" has been removed from the list of items considered, approved TSTF-249, which elaborated on ITC, has not been incorporated.

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

PART G

NO SIGNIFICANT HAZARDS DETERMINATION
(NSHD)

and

ENVIRONMENTAL ASSESSMENT

for

CHANGES TO PRAIRIE ISLAND
CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Part G

PACKAGE 3.1

REACTIVITY CONTROL SYSTEMS

NO SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes to the Operating License have been evaluated to determine whether they constitute a significant hazards consideration as required by 10CFR Part 50, Section 50.91 using the standards provided in Section 50.92.

For ease of review, the changes are evaluated in groupings according to the type of change involved. A single generic evaluation may suffice for some of the changes while others may require specific evaluation in which case the appropriate reference change numbers are provided.

A - Administrative (GENERIC NSHD)
(A3.1-01, A3.1-02, A3.1-06, A3.1-08, A3.1-12, A3.1-34, A3.1-36, A3.1-48, A3.1-52, A3.1-54, A3.1-57)

Most administrative changes have not been marked-up in the Current Technical Specifications, and may not be specifically referenced to a discussion of change. This No Significant Hazards Determination (NSHD) may be referenced in a discussion of change by the prefix "A" if the change is not obviously an administrative change and requires an explanation.

These proposed changes are editorial in nature. They involve reformatting, renaming, renumbering, or rewording of existing Technical Specifications to provide consistency with NUREG-1431 or conformance with the Writer's Guide, or change of current plant terminology to conform to NUREG-1431. Some administrative changes involve relocation of requirements within the Technical Specifications without affecting their technical content. Clarifications within the new Prairie Island Improved Technical Specifications which do not impose new requirements on plant operation are also considered administrative.

1. <u>The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The proposed conversion of Prairie Island Current Technical Specifications to conform to NUREG-1431 involves reformatting, rewording, changes in terminology and relocating requirements. These changes are simply editorial, or do not involve technical changes and thus they do not impact any initiators of previously analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.</u>

   These proposed administrative changes do not involve physical modification of the plant, no new or different type of equipment will be installed or removed associated with these administrative changes, nor will there be changes in parameters governing normal plant operation. The proposed administrative changes do not impose new or different requirements on plant operation. Therefore, these administrative changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   These proposed administrative changes do not impact any safety analysis assumptions. Therefore, these changes do not involve a reduction in the plant margin of safety.

M - More restrictive (GENERIC NSHD)
(M3.1-04, M3.1-09, M3.1-11, M3.1-17, M3.1-18, M3.1-19, M3.1-22, M3.1-24, M3.1-27, M3.1-29, M3.1-31, M3.1-32, M3.1-38, M3.1-42, M3.1-44, M3.1-47, M3.1-53, M3.1-62, M3.1-66)

This proposed Technical Specifications revision involves modifying the Current Technical Specifications to impose more stringent requirements upon plant operations to achieve consistency with the guidance of NUREG-1431, correct discrepancies or remove ambiguities from the specifications. These more restrictive Technical Specifications have been evaluated against the plant design, safety analyses, and other Technical Specifications requirements to ensure the plant will continue to operate safely with these more stringent specifications.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes provide more stringent requirements for operation of the plant. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event.

   These more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed changes do not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed, nor do they change the methods governing normal plant operation.

   These more stringent requirements do impose different operating restrictions. However, these operating restrictions are consistent with the boundaries established by the assumptions made in the plant safety analyses and licensing bases. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

M - More restrictive (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The imposition of more stringent requirements on plant operation either has no impact on the plant margin of safety or increases the margin of safety. Each change in this category is by definition providing additional restrictions to enhance plant safety by:

a) increasing the analytical or safety limit;
b) increasing the scope of the specifications to include additional plant equipment;
c) adding requirements to current specifications;
d) increasing the applicability of the specification;
e) providing additional actions;
f) decreasing restoration times;
g) imposing new surveillances; or
h) decreasing surveillance intervals.

These changes maintain requirements within the plant safety analyses and licensing bases. Therefore, these changes do not involve a significant reduction in a margin of safety.

R - Relocation (GENERIC NSHD)
(None in this Package)

This License Amendment Request (LAR) proposes to relocate requirements contained in the Current Technical Specifications out of the Technical Specifications into licensee controlled programs. These requirements are relocated because they 1) do not meet the Technical Specifications selection criteria defined in 10 CFR 50.36; or 2) are mandated by current Nuclear Regulatory Commission (NRC) regulations and are therefore unnecessary in the Technical Specifications.

In the NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (dated 7/16/93), the NRC stated:

> . . . since 1969, there has been a trend towards including in Technical Specifications not only those requirements derived from the analyses and evaluations included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power reactors... This has contributed to the volume of Technical Specifications and to the several-fold increase, since 1969, in the number of license amendment applications to effect changes to the Technical Specifications. It has diverted both staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety.

Thus, relocation of unnecessary requirements from the Current Technical Specifications should result in an overall improvement in plant safety through more focused attention to the requirements that are most important to plant safety.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   These proposed changes relocate requirements for structures, systems, components or variables which did not meet the criteria for inclusion in the improved Technical Specifications, or which duplicate regulatory requirements. 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.

   These relocated operability requirements will continue to be maintained pursuant to 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Controls section of these proposed improved Technical Specifications.

R – Relocation (continued)

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes do not impose any different requirements and adequate control of existing requirements will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

These proposed changes will not reduce the margin of safety because they do not impact any safety analysis assumptions. In addition, the relocated requirements for the affected structure, system, component or variables are the same as the current Technical Specifications. Since future changes to these requirements will be evaluated per the requirements of 10 CFR 50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), or the Administrative Control section of the Improved Technical Specifications, proper controls are in place to maintain the plant margin of safety. Therefore, these changes do not involve a significant reduction in the margin of safety.

LR - Less restrictive, Relocated details (GENERIC NSHD)
(LR3.1-03, LR3.1-07, LR3.1-37, LR3.1-43, LR3.1-51, LR3.1-59, LR3.1-65)

Some information in the Prairie Island Current Technical Specifications that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and relocated to the proposed Bases, Updated Safety Analysis Report or licensee controlled procedures. The relocation of this descriptive information to the Bases of the Improved Technical Specifications, Updated Safety Analysis Report or licensee controlled procedures is acceptable because these documents will be controlled by the Improved Technical Specifications required programs, procedures or 10CFR50.59. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes relocate detailed, descriptive requirements from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures. These documents containing the relocated requirements will be maintained under the provisions of 10CFR50.59, a program or procedure based on 10CFR50.59 evaluation of changes, or NRC approved methodologies. Since these documents to which the Technical Specifications requirements have been relocated are evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no increase in the probability or consequences of an accident previously evaluate will be allowed without prior NRC approval. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   These proposed changes do not necessitate physical alteration of the plant, that is, no new or different type of equipment will be installed, or change parameters governing normal plant operation. The proposed changes will not impose any different requirements and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

LR - Less restrictive, Relocated details (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to the Bases, Updated Safety Analysis Report or licensee controlled procedures are the same as the existing Technical Specifications. Since future changes to these requirements will be evaluated under 10CFR50.59 or its guidance, or in accordance with NRC approved methodologies, no reduction in a margin of safety will be allowed without prior NRC approval. Therefore, these changes do not involve a significant reduction in a margin of safety.

L - Less restrictive, Specific

Each CTS change which is designated as Less (L prefix) restrictive on plant operations is provided with a specific NSHD.

Specific NSHD for Change L3.1-21

CTS does not include action statements which provide remedial actions if rod insertion limits are not met. Thus the plant would enter CTS 3.0.C (ITS 3.0.3) which would require the plant to be in MODE 3 in 6 hours and MODE 5 in 36 hours. The new ITS Action Statement will require the plant to be in MODE 3 in 6 hours. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change allows the plant to stay in MODE 3 when rod insertion limits are not met in MODES 1 and 2 rather than continuing shutdown to MODE 5. This change does not involve an increase in the probability or consequences of an accident previously evaluated because there are not any accidents evaluated in MODES 3, 4 and 5 which consider rod insertion.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a significant reduction in margin of safety. The effect of this change is to allow the plant to remain in MODE 3 when rod insertion limit Action Statements have not been met. When the plant is in MODE 3 all rods are fully inserted and the plant is maintained in a safe condition by SDM TS

Specific NSHD for Change L3.1-21 (continued)

requirements. Since rod insertion is not considered in MODES 3, 4 and 5 in any accident analyses, this change does not involve a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.1-23

CTS require control banks to meet insertion limits when the reactor is approaching criticality. This change will require insertion limits once criticality is achieved. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Control banks insertion limits are required when the reactor is at power to assure that the assumed power distribution, ejected rod worth, SDM and reactivity insertion rate assumptions are met. Prior to achieving criticality the reactor is not at power. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   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 parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change eliminates TS required control bank insertion limits when the reactor is approaching criticality. As a practical matter, when criticality is achieved, the insertion limits are met through use of the estimated critical concentration of boron. Therefore, at some time prior to reaching criticality, the insertion limits must be met to assure that they are met when criticality is achieved. Since CTS do not define when "approach to criticality" occurs, essentially the plant is required to meet control bank insertion limits at some time during the approach to criticality although it is no longer a TS requirement. Control banks insertion limits are required when the reactor is at power to assure that the assumed power distribution assumptions are met. Prior to criticality, the reactor is not at power and does not have a power distribution which requires control. Thus, this change does not involve a significant reduction in margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.1-26

CTS does not include action statements which provide remedial actions if control bank insertion limits are not met. Thus the plant would enter CTS 3.0.C (ITS 3.0.3) which would require the plant to be in MODE 3 in 6 hours and MODE 5 in 36 hours. The new ITS Action Statements removes the plant from the MODE of Applicability by requiring the plant to be in MODE 2 with $K_{eff} < 1.0$ in 6 hours. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change allows the plant to stay in MODE 2 with $K_{eff} < 1.0$ when rod insertion limits are not met in MODES 1 and 2 with $K_{eff} \geq 1.0$ rather than continuing to shutdown to MODE 5. This change does not involve an increase in the probability or consequences of an accident previously evaluated because there are not any accidents evaluated in MODE 2 with $K_{eff} < 1.0$ and in MODES 3, 4 and 5 which consider rod insertion.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident.

<u>Specific NSHD for Change L3.1-26</u> (continued)

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   The proposed change makes the PI ITS consistent with the guidance of NUREG-1431 and does not involve a significant reduction in margin of safety. The intent of this specification is to remove the plant from the MODE of Applicability. The effect of this change is to allow the plant to remain in MODE 2 with $K_{eff} < 1.0$ when control bank insertion limit Action Statements have not been met. When the plant is in MODE 2 with $K_{eff} < 1.0$, control bank insertion limits are not a safety concern.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.1-28

The proposed change allows Physics Testing exceptions from the TS requirements for ITC, rod group alignment limits and RCS minimum temperature for criticality. These exceptions are not allowed by CTS. To assure that the plant is maintained in a safe condition, new Physics Testing limitations on the RCS lowest loop average temperature, SDM and thermal power are provided. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change provides new TS exceptions for Physics Testing. These exceptions are acceptable as long as the fuel design criteria are not violated. When the TS requirements for ITC, rod group alignment limits and RCS minimum temperature for criticality 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 ≥ 535 °F and SDM is within the limits provided in the COLR. Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>Specific NSHD for Change L3.1-28</u> (continued)

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

   This change allows some TS limitations to be suspended for Physics Testing. Fuel design criteria are preserved through new limitations imposed on the reactor during Physics Testing. Therefore, in consideration of the offsetting changes, the proposed change does not result in a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.1-33

The proposed change requires power to be reduced to 85% of rated power in lieu of reducing the high neutron flux trip setpoint to 85%. This change is acceptable since it provides the benefit of reducing power without the risk of a transient from reducing the trip setpoint. This change is consistent with the guidance of NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change requires reducing power to 85% in lieu of reducing the high neutron flux trip setpoint to 85%. This change does not affect the probability that rod misalignment will occur; therefore, this change does not affect the probability of a previously evaluated accident. With this change, a transient with a rod misalignment which would have resulted in a high flux trip under the CTS requirements may not trip until a higher power level is reached. However, this would not involve a significant increase in the consequences of an accident previously evaluated because the initial power level has been reduced to 85% of rating. Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Specific NSHD for Change L3.1-33 (continued)

3. The proposed amendment will not involve a significant reduction in the margin of safety.

This change requires the plant power level to be reduced to 85% of rating in lieu of reducing the high neutron trip setpoints. With this change, a transient with a rod misalignment which would have resulted in a high flux trip under the CTS requirements may not trip until a higher power level is reached. However, the initial power is reduced and the possibility of transients associated with adjusting the high neutron flux trip setpoint is avoided. Therefore, in consideration of the offsetting changes, the proposed change does not result in a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.1-63

CTS requires operability of the rod position indication system in MODES 1, 2, 3, 4 and 5. This change will require the rod position indication system to be operable in MODES 1 and 2. This change is acceptable because in MODES 3, 4 and 5, the control rods are fully inserted, the reactor is shutdown and reactivity control is maintained by SDM requirements. Thus indication of rod position is not required in these modes. This change is consistent with NUREG-1431.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The RPI system is an instrumentation system which is not an accident initiator and does not affect the consequences of an accident. Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   This proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. The proposed change only changes the modes for which the RPI must be operable. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   This change allows the RPI system to be inoperable in Modes 3, 4, and 5. In these modes the reactor is shutdown, the rods are fully inserted and reactor safety is assured by boration to the meet the SDM TS requirements. Thus, indication of rod position through the RPIs is not required. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431.

Specific NSHD for Change L3.1-64

CTS requires verification of the RPI system prior to each startup following shutdown in excess of two days if not done in the previous 30 days. This change will require this verification to be performed prior to criticality after each removal of the reactor head. This change is consistent with NUREG-1431 as modified by TSTF-89.

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The RPI system is an instrumentation system which is not an accident initiator and does not affect the consequences of an accident. Therefore, this change does not involve a significant increase in the probability or consequences of a previously analyzed accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   This proposed change does not involve a physical alteration of the plant, that is, no new or different type of equipment will be installed. The proposed change only changes the conditions for which the RPI functional verification must be performed. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

   The proposed change modifies the criteria for performing RPI function verifications. Activities associated with removal of the reactor head are more likely to affect performance of the RPI system than the act of shutting down and remaining shutdown for more than two days. Thus the criteria for performing this SR are directly tied to those activities which affect system performance. Therefore, this change does not involve a significant reduction in a margin of safety.

Therefore it is concluded this proposed change does not involve a significant hazards consideration. This change is consistent with the guidance of NUREG-1431 as modified by approved TSTF-89.

ENVIRONMENTAL ASSESSMENT

The Nuclear Management Company has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.