Indian Point
3

Improved

Technical Specifications

Conversion Submittal

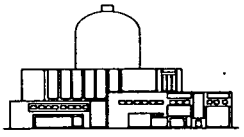
New York Power
Authority

New York Power Authority

Improved Technical Specifications Conversion Submittal

VOLUME 1

SECTION I:
SPLIT REPORT
and
SECTION II:
RELOCATED ITEMS



INDIAN POINT 3

SECTION I:

APPLICATION OF THE

NRC FINAL POLICY STATEMENT

SELECTION CRITERIA TO THE

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

TECHNICAL SPECIFICATIONS

CONTENTS

1. INTRODUCTION

The purpose of this document is to confirm that the results of the Westinghouse Owners Group application of the Technical Specification selection criteria are applicable to (IP3). The New York Power Authority (NYPA) has reviewed WCAP-11618, "Methodically Engineered, Restructured and Improved, Technical Specifications, Merits Program - Phase II Task 5, Criteria Application" (Reference 1) including Addendum 1, NRC Staff Review of NSSS Vendor Owners Group Application of the Commission's Interim Policy Statement to Standard Technical Specifications, Newton/Murley letter dated May 9, 1988 and as revised in NUREG-1431, Revision 1 "Standard Technical Specifications, Westinghouse Plants," (Reference 2) and applied the criteria to each of the current Indian Point Nuclear Generating Unit No. 3 Technical Specifications. Additionally, in accordance with the NRC guidance in the Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (Reference 3), this confirmation of the application of selection criteria to IP3 included confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations provided in the Reference 1.

2. SELECTION CRITERIA

NYPA used the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 (Reference 3) to develop the results contained in the attached matrix. Probabilistic Risk Assessment (PRA) insights as used in WCAP-11618 were used, confirmed by NYPA, and are discussed in the next section of this report. The selection criteria and discussion provided in Reference 3 are as follows:

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

<u>Discussion of Criterion 1</u>: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident. This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however,

be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

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

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing design basis accident and transient analyses and that the plant will be operated to preclude un-analyzed transients and accidents. Analyses consist of postulated events, analyzed in the FSAR, for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 14 of the FSAR and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the design basis accident or transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room. These could also include other features or characteristics that are specifically assumed in Design basis accident or transient analyses if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors).

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

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

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

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

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

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

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

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

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

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as part of the Commission's ongoing program of improving technical specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements."

3. PROBABILISTIC RISK ASSESSMENT INSIGHTS

Introduction and Objectives

Reference 3 includes an NRC expectation that NYPA utilize the available literature on risk insights to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Relocated specifications have been compared to a variety of Probabilistic Risk Assessment (PRA) material with two purposes: 1) to identify if a component or variable is addressed by PRA, and 2) if addressed, to judge if the component or variable is risk-important. In addition, in some cases risk was judged independent of any specific PRA material. The intent of the review was to provide a supplemental screen to the deterministic criteria. Therefore, those TS that remain in the Improved Technical Specifications (ITS) based on Criteria 1, 2 or 3 were not reviewed for risk significance.

The IP3 risk assessment was performed using WCAP-11618 (Reference 1), Appendix A, PRA Technical Specification Review. This document provides a generic risk assessment of items relocated from the Westinghouse Technical Specifications which is based on "Indian Point Probabilistic Safety Study," Power Authority State of New York, Consolidated Edison Company of New York, Inc., 1982, and similar studies for Zion Nuclear Plant and Millstone Unit 3.

NYPA has reviewed these generic risk assessments and verified that it is applicable to IP3. Additionally, the Indian Point 3 Nuclear Power Plant Individual Plant Examination, June, 1994, was reviewed to verify that the insights provided by this evaluation are consistent with the results of the generic risk assessments.

For relocated Specifications for which WCAP-11618 (Reference 1), Appendix A, does not have a generic risk assessment, a plant specific risk assessment was performed and the results documented in Appendix B.

Assumptions and Approach

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

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

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

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

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

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

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

d. The criteria used to determine if a sequence was risk dominant was the following:

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

   For off-site health effects, any sequence whose frequency of

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

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

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

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the IP3 Technical Specifications. The results are presented in following attachments to this report: Attachment I, CTS to ITS Disposition and Relocation Matrix, and Attachment II, ITS to CTS Disposition Matrix. Detailed discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Section II, Relocated

Requirements - Descriptions and Justifications for the relocation of selected IP3 current Technical Specifications (CTS), No Significant Hazards Considerations (10 CFR 50.92) evaluations for those Specifications relocated are provided Sections IV.

5. REFERENCES

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

2. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 1, April 1995.

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

APPLICATION OF THE

NRC FINAL POLICY STATEMENT

SELECTION CRITERIA TO THE

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

TECHNICAL SPECIFICATIONS

ATTACHMENT 1:

CTS to ITS Disposition and Relocation Matrix

INDIAN POINT 3
Conversion to Improved Technical Specifications
CTS to ITS Disposition and Relocation Matrix

Current Technical Specification (CTS)	INDIAN POINT 3: Current Technical Specification Disposition and Relocation Matrix

License 2.C.1

License Condition: Maximum Power Level 3025 mW thermal (100% of rated power)

Status:	Discussion
Retained	Maximum Power Level will be maintained as both a License Condition and as a Technical Specification Safety Limit.

ITS Requirements that maintain requirements in CTS License 2.C.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
1.0	1.0	USE AND APPLICATION	N/A	None

License 2.C.2

License Condition: Technical Specifications

Status:	Discussion
Retained	Retained as a License Condition

ITS Requirements that maintain requirements in CTS License 2.C.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.C.3

License Condition: Less Than Four Loop Operation

Status:	Discussion
Retained	License Condition 2.C.3 allows 3 loop operation below the P-7 setpoint (approximately 10% RTP). ITS LCO 3.4.4 requires four RCS loops OPERABLE and in operation when in Modes 1 and 2. License Condition 2.C.3 conflicts with ITS LCO 3.4.4 and must be delete

ITS Requirements that maintain requirements in CTS License 2.C.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.4	3.4.4	RCS Loops - MODES 1 and 2	2	None

License 2.C.4

License Condition: Pressurizer Weld Inspection prior to power operation following second refueling shutdown

Status:	Discussion
DELETED	License Condition 2.C.4 has been satisfied and is no longer applicable. License Condition 2.C.4 should be deleted.

ITS Requirements that maintain requirements in CTS License 2.C.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
CTS DELETED	CTS DELETED	CTS REQUIREMENT DELETED	N/A	None

License 2.D

License Condition: Deleted by CTS Amendment 46

Status:	Discussion
N/A	None

ITS Requirements that maintain requirements in CTS License 2.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.E

License Condition: Deleted by CTS Amendment 37

Status:	Discussion
N/A	None

ITS Requirements that maintain requirements in CTS License 2.E

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.F

License Condition: Section 401, Federal Water Pollution Control Act Amendments of 1972

Status:	Discussion
Retained	Retained as a License Condition

ITS Requirements that maintain requirements in CTS License 2.F

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.G

License Condition: Physical Security Plan

Status:	Discussion
Retained	Retained as a License Condition

ITS Requirements that maintain requirements in CTS License 2.G

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.H

License Condition: Fire Protection Program

Status:	Discussion
Retained	Retained as a License Condition

ITS Requirements that maintain requirements in CTS License 2.H

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.I

License Condition: Secondary Water Chemistry Monitoring Program

Status:	Discussion
Retained	Secondary Water Chemistry Monitoring Program is retained as Technical Specifications 5.5.9. License Condition 2.I should be deleted to eliminate repetition.

ITS Requirements that maintain requirements in CTS License 2.I

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.10	5.5.9	Secondary Water Chemistry Program	N/A	None

License 2.J

License Condition: Inspect all four steam generators by March 31, 1982.

Status:	Discussion
DELETED	License Condition 2.J has been satisfied and is no longer applicable. License Condition 2.J should be deleted. Steam Generator tube inspection requirements will be governed by Technical Specification 5.5.8.

ITS Requirements that maintain requirements in CTS License 2.J

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
CTS DELETED	CTS DELETED	CTS REQUIREMENT DELETED	N/A	None

License 2.K

License Condition: Deleted by Amendment 49

Status:	Discussion
N/A	None

ITS Requirements that maintain requirements in CTS License 2.K

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.L

License Condition: Program to Reduce Leakage from Systems Outside Containment

Status: Retained

Discussion

Program to Reduce Leakage from Systems Outside Containment is retained as Technical Specifications 5.5.2. License Condition 2.L should be deleted to eliminate repetition.

ITS Requirements that maintain requirements in CTS License 2.L

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.2	5.5.2	Primary Coolant Sources Outside Containment	N/A	None

License 2.M

License Condition: Program to ensure capability to determine airborne iodine concentration in vital areas in accident conditions

Status: Retained

Discussion

Program to ensure capability to determine airborne iodine concentration in vital areas in accident conditions is retained as Technical Specifications 5.5.3. License Condition 2.M should be deleted to eliminate repetition.

ITS Requirements that maintain requirements in CTS License 2.M

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.3	5.5.3	Post Accident Sampling	N/A	None

License 2.N

License Condition: Deleted by Amendment 49

Status: N/A

Discussion

None

ITS Requirements that maintain requirements in CTS License 2.N

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

License 2.O

License Condition: Schedule for Completion of Balance of Plant Modifications to NRC by January 1, 1984.

Status: DELETED

Discussion

License Condition 2.O has been satisfied and is no longer applicable. License Condition 2.O should be deleted.

ITS Requirements that maintain requirements in CTS License 2.O

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
CTS DELETED	CTS DELETED	CTS REQUIREMENT DELETED	N/A	None

1.0

Definitions

Status: Retained

Discussion

Definitions are provided to improve understanding and ensure consistent application. Application of the TS selection criteria to these definitions is not appropriate.

ITS Requirements that maintain requirements in CTS 1.0

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
1.0	1.0	USE AND APPLICATION	N/A	None

1.10

Containment Integrity Definition

Status: Retained

Discussion

CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. Requirements ar maintained as Limiting Conditions of Operation listed below.

ITS Requirements that maintain requirements in CTS 1.10

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.1	3.6.1	Containment	3	None
3.6.2	3.6.2	Containment Air Locks	3	None
3.6.3	3.6.3	Containment Isolation Valves	3	None

2.0 Safety Limits and Limiting Safety System Settings

Status: N/A

Discussion

10 CFR 50.36 requires that Safety Limits are included in the Technical Specifications.

ITS Requirements that maintain requirements in CTS 2.0

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
2.0	2.0	SAFETY LIMITS (SLs)	N/A	None

2.1 Safety Limits, Reactor Core (Reactor Power, Pressure and Temperature)

Status: Retained

Discussion

Application of TS selection criteria to Safety Limits is not appropriate. These safety limits are retained in the ITS.

ITS Requirements that maintain requirements in CTS 2.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
2.1.1	2.1.1	Safety Limits	N/A	None

2.2 Safety Limit, Reactor Coolant System Pressure

Status: Retained

Discussion

Application of TS selection criteria to Safety Limits is not appropriate. These safety limits are retained in the ITS.

ITS Requirements that maintain requirements in CTS 2.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
2.1.2	2.1.2	Safety Limits	N/A	None

2.3 Limiting Safety System Settings, Protective Instrumentation

Status: Retained

Discussion

The limiting safety settings for protective instrumentation function to actuate the Reactor Protection System (RPS) to mitigate the consequences of Design Basis Accidents (DBAs) and/or transients.

ITS Requirements that maintain requirements in CTS 2.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.1	3.3.1	Reactor Protection System (RPS) Instrumentation	3	None

3.0 Limiting Conditions for Operation

Status: Retained

Discussion

This Specification provides guidance applicable to one or more Limiting Conditions for Operation (LCOs). The requirements of CTS Section 3 are expanded in ITS.

ITS Requirements that maintain requirements in CTS 3.0

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.0	3.0	LCO APPLICABILITY and SR APPLICABILITY	N/A	None

3.1.A Operational Components

Status: RETAINED

Discussion

None

ITS Requirements that maintain requirements in CTS 3.1.A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4	3.4	REACTOR COOLANT SYSTEM (RCS)	N/A	None

3.1.A.1 Coolant Pumps

Status: Retained

Discussion

Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.

ITS Requirements that maintain requirements in CTS 3.1.A.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.4	**3.4.4**	RCS Loops - MODES 1 and 2	2	None
3.4.5	**3.4.5**	RCS Loops - MODE 3	3	None
3.4.6	**3.4.6**	RCS Loops - MODE 4	4	None
3.4.7	**3.4.7**	RCS Loops - MODE 5, Loops Filled	4	None
3.4.8	**3.4.8**	RCS Loops - MODE 5, Loops Not Filled	4	None
3.4.12	**3.4.12**	Low Temperature Overpressure Protection (LTOP)	2	LTOP requirements are enforced during RCP starts below LTOP enable temperature.

3.1.A.2 Safety Valves (Pressurizer)

Status: Retained

Discussion

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS.

ITS Requirements that maintain requirements in CTS 3.1.A.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.10	**3.4.10**	Pressurizer Safety Valves	3	None

3.1.A.3 Pressurizer Heaters

Status: Retained

Discussion

A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remov core decay heat by either forced or natural circulation of reactor coolant.

ITS Requirements that maintain requirements in CTS 3.1.A.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.9	3.4.9	Pressurizer	2	None

3.1.A.4 Power Operated Relief Valves

Status: Retained

Discussion

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal and alternate pressurizer spray are not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator action are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are incorporated into the TS in response to GL 90-06.

ITS Requirements that maintain requirements in CTS 3.1.A.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.11	3.4.11	Pressurizer Power Operated Relief Valves (PORVs)	3	None

3.1.A.5 Power Operated Relief Block Valves

Status: Retained

Discussion

The block valves are used to isolate the PORVs in case of the associated PORV has excessive leakage or is stuck open.

ITS Requirements that maintain requirements in CTS 3.1.A.5

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.11	3.4.11	Pressurizer Power Operated Relief Valves (PORVs)	3	None

3.1.A.6 Deleted by Amendment 170

Status: N/A

Discussion N/A

ITS Requirements that maintain requirements in CTS 3.1.A.6

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

3.1.A.7 Reactor Vessel Head Vents

Status: Relocated

Discussion See CTS Relocated Item R.1.

ITS Requirements that maintain requirements in CTS 3.1.A.7

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.1	R.1	REACTOR VESSEL HEAD VENTS	None	None

3.1.A.8 Overpressure Protection System (OPS)

Status: Retained

Discussion LTOP is established to limit RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. In MODES 1, 2, and 3, with RCS cold leg temperature exceeding 411 F, the pressurizer safety valves will prevent RCS pressure from exceeding Appendix G limits. At 319 F and below, overpressure prevention must be ensured by two OPERABLE PORVs in conjunction with the Overpressure Protection System or to a depressurized RCS and a sufficient sized RCS vent.

ITS Requirements that maintain requirements in CTS 3.1.A.8

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.12	3.4.12	Low Temperature Overpressure Protection (LTOP)	2	None

3.1.B Heatup and Cooldown

Status: Retained/Relocated

Discussion

This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. These limits are needed to avoid encountering pressure, temperature, or temperature rate of change condition that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary.

ITS Requirements that maintain requirements in CTS 3.1.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.3	3.4.3	RCS Pressure and Temperature (P/T) Limits	2	None
R.2	R.2	STEAM GENERATOR SECONDARY SIDE MINIMUM TEMPERATURE FOR PRESSURIZATION	None	WCAP-11618 determined that steam generator P/T limits were not significant risk contributors to core damage frequency and offsite releases.
R.3	R.3	PRESSRIZER HEATUP AND COOLDOWN	None	WCAP-11618 determined that the pressurizer temperature limits were not significant risk contributors to core damage frequency and offsite releases.

3.1.C Minimum Conditions for Criticality

Status: N/A

Discussion

Establishes operating restrictions such that operation is bounded by the accident analysis.

ITS Requirements that maintain requirements in CTS 3.1.C

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1	3.1	REACTIVITY CONTROL SYSTEMS	N/A	None
3.4	3.4	REACTOR COOLANT SYSTEM (RCS)	N/A	None

3.1.C.1 Moderator Temperature Coefficient

Status: Retained

Discussion

The FSAR contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding.

ITS Requirements that maintain requirements in CTS 3.1.C.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.4	3.1.3	Moderator Temperature Coefficient (MTC)	2	ITS limits both he most positive value and most negative value of the MTC.

3.1.C.2 Deleted

Status: N/A

Discussion

None

ITS Requirements that maintain requirements in CTS 3.1.C.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

3.1.C.3 Minimum Temperature for Criticality

Status: Retained

Discussion

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

ITS Requirements that maintain requirements in CTS 3.1.C.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.10	3.1.8	PHYSICS TESTS Exceptions MODE 2	1, 2, 3	None
3.4.2	3.4.2	RCS Minimum Temperature for Criticality	2	None

3.1.C.4 Pressurizer Water Level

Status: Retained

Discussion

All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer.

ITS Requirements that maintain requirements in CTS 3.1.C.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.9	3.4.9	Pressurizer	2	CTS requires Pressurizer at the normal level; ITS requires only that there is a bubble in the pressurizer.

3.1.D Primary Coolant Activity

Status: Retained

Discussion

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.

ITS Requirements that maintain requirements in CTS 3.1.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.16	3.4.16	RCS Specific Activity	2	None

3.1.E Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

Status: Relocated

Discussion

See CTS Relocated Item R.4.

ITS Requirements that maintain requirements in CTS 3.1.E

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.4	R.4	MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION	None	None

3.1.F Leakage of Reactor Coolant

Status: Retained

Discussion

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, operational LEAKAG is related to the safety analyses for LOCA because the amount of leakage can affect the probability of such an event.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR).

ITS Requirements that maintain requirements in CTS 3.1.F

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.13	3.4.13	RCS Operational LEAKAGE	2	None
3.4.14	3.4.14	RCS Pressure Isolation Valve (PIV) Leakage	2	ITS retains the CTS allowance that leakage into closed systems is not counted as either identified or unidentified leakage.

3.1.G Secondary Coolant Activity

Status: Retained

Discussion

The accident analysis of the main steam line break (MSLB),as discussed in the FSAR assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the EAB (i.e., site boundary) limits for whole body and thyroid dose rates.

ITS Requirements that maintain requirements in CTS 3.1.G

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.18	3.7.17	Secondary Specific Activity	2	None

3.1.H RCS Pressure, Temperature, and Flow DNB Limits

Status: Retained

Discussion

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses. The safet analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed include loss of coolant flow events and dropped or stuck rod events.

ITS Requirements that maintain requirements in CTS 3.1.H

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.1	3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	2	None

3.2 Chemical and Volume Control System

Status: Relocated

Discussion

See CTS Relocated Item R.5.

ITS Requirements that maintain requirements in CTS 3.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.5	R.5	CHEMICAL AND VOLUME CONTROL SYSTEM	None	None

3.3.A Safety Injection and Residual Heat Removal Systems

Status: Retained

Discussion

ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients.

The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.

ITS Requirements that maintain requirements in CTS 3.3.A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.3	3.3.3	Post Accident Monitoring (PAM) Instrumentation	3, 4	The IP3 design does not include automatic switchover fro SI mode to recirc mode based on RWST level. This functi is performed manually based on RWST level alarm and containment sump level.
3.4.6	3.4.6	RCS Loops - MODE 4	4	None
3.4.7	3.4.7	RCS Loops - MODE 5, Loops Filled	4	None
3.4.8	3.4.8	RCS Loops - MODE 5, Loops Not Filled	4	None
3.4.12	3.4.12	Low Temperature Overpressure Protection (LTOP)	2	LTOP requirements place restrictions on bothECCS and RHR lineups.
3.5.1	3.5.1	Accumulators	3	None
3.5.2	3.5.2	ECCS - Operating	3	None
3.5.3	3.5.3	ECCS - Shutdown	3	None
3.5.4	3.5.4	Refueling Water Storage Tank (RWST)	3	None

3.3.B Containment Cooling and Iodine Removal Systems

Status: Retained

Discussion

The Containment Spray System and Containment Fan Cooler System provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal reduces the release of radioactivity to the environment following a DBA.

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a DBA.

ITS Requirements that maintain requirements in CTS 3.3.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.6A	**3.6.6**	Containment Spray System and Containment Fan Cooler System	3	None
3.6.7	**3.6.7**	Spray Additive System	3	None

3.3.C Isolation Valve Seal Water System (IVSWS)

Status: Retained

Discussion

The IVSW System is a seal system as described in 10 CFR 50, Appendix J. The IVSW System improves the effectiveness of certain containment isolation valves by providing a water seal to valve leakage paths.

ITS Requirements that maintain requirements in CTS 3.3.C

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
IP3 UNIQUE 1x	**3.6.9**	Isolation Valve Seal Water (IVSW) System	3	None

3.3.D Weld Channel and Penetration Pressurization System (WC & PPS)

Status: Relocated

Discussion

See CTS Relocated Item R.6.

ITS Requirements that maintain requirements in CTS 3.3.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.6	**R.6**	Weld Channel and Penetration Pressurization System (WC & PPS)	None	None

3.3.E Component Cooling System

Status: Retained

Discussion

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient.

ITS Requirements that maintain requirements in CTS 3.3.E

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.7	3.7.8	Component Cooling Water (CCW) System	3	None

3.3.F Service Water System/Ultimate Heat Sink

Status: Retained

Discussion

The Service Water System (SWS) and UHS functions in conjunction with the CCS to remove post LOCA heat loads from the containment sump during the recirculation phase. The SWS in conjunction with CCS also functions to cool the unit from RHR entry conditions to cold shutdown during normal and post accident conditions.

ITS Requirements that maintain requirements in CTS 3.3.F

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.8	3.7.9	Service Water System (SWS)	3	None
3.7.9	3.7.10	Ultimate Heat Sink (UHS)	3	None

3.3.G Containment Hydrogen Monitoring Systems

Status: Retained

Discussion

Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

ITS Requirements that maintain requirements in CTS 3.3.G

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.3	3.3.3	Post Accident Monitoring (PAM) Instrumentation	3, 4	None

3.3.H Control Room Ventilation System (including toxic gas monitoring)

Status: Retained/Relocated

Discussion

Control room ventilation provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The CRACS is designed so that the functional capability of the Control Room is maintained during a Design Basis Accident. Functional capability means that the ambient temperature for safety equipment located in this room will not exceed 108.2 F. Control Room safety equipment is specified to a temperature of 120 F and the 108.2 F limit for Control Room temperature is sufficient to account for the temperature rise in the enclosed cabinets.

ITS Requirements that maintain requirements in CTS 3.3.H

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.7	**3.3.7**	Control Room Emergency Ventilation (CRVS) Actuation Instrumentation	3	None
3.7.10	**3.7.11**	Control Room Ventilation System (CRVS)	3	None
3.7.11	**3.7.12**	Control Room Air Conditioning System (CRACS)	3	None
R.18	**R.18**	Toxic Gas Monitoring	None	Toxic gas monitoring is an alarm only function and is relocated consistent with Generic Letter 95-10.

3.3.I Electric Hydrogen Recombiner System

Status: Retained

Discussion

Hydrogen control ensures that pressure and temperature assumed in the analyses are not exceeded during a post LOCA hydrogen burn

ITS Requirements that maintain requirements in CTS 3.3.I

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.8	**3.6.8**	Hydrogen Recombiners	3	None

3.4 Steam and Power Conversion System

Status:	Discussion
N/A	None

ITS Requirements that maintain requirements in CTS 3.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7	3.7	PLANT SYSTEMS	N/A	None

3.4.A.1 Main Steam Safety Valves

Status:	Discussion
Retained	The purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary by providing a heat sink for the removal of energy from the RCS.

ITS Requirements that maintain requirements in CTS 3.4.A.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.1	3.7.1	Main Steam Safety Valves (MSSVs)	3	None

3.4.A.2 Auxiliary Feedwater Pumps

Status:	Discussion
Retained	The Auxiliary Feedwater System functions to remove decay heat during design basis events thus mitigating consequences of events which could result in over pressurization of the RCS pressure boundary.

ITS Requirements that maintain requirements in CTS 3.4.A.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.5	3.7.5	Auxiliary Feedwater (AFW) System	3	None

3.4.A.3 Condensate Storage Tank

Status: Retained

Discussion

The Condensate Storage Tank or city water function to provide cooling water to remove decay heat and cool down the unit for design basis events.

ITS Requirements that maintain requirements in CTS 3.4.A.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.6	3.7.6	Condensate Storage Tank (CST)	2, 3	None

3.4.A.4 Piping and Valves For Main Steam Safety Valves, Auxiliary Feedwater and Condensate Storage Tank

Status: Retained

Discussion

See CTS 3.4.A.1, CTS 3.4.A.2 and CTS 3.4.A.3.

ITS Requirements that maintain requirements in CTS 3.4.A.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

3.4.A.5 Main Steam Stop Valves

Status: Retained

Discussion

The combination of MSIVs and MSCVs precludes the blowdown of more than one steam generator, assuming a single active componen failure (e.g., the failure of one MSIV to close on demand). For a break upstream of an MSIV, either the MSIVs in the other three steam lines or the MSCV in the steam line with the faulted SG must close to prevent the blowdown of more than one SG. For a break downstream of an MSIV, the MSCVs are not required to function.

ITS Requirements that maintain requirements in CTS 3.4.A.5

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.2	3.7.2	Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)	3	None

3.4.A.6 Steam Generators (Two SGs for decay heat removal)

Status: Retained

Discussion

This requirement is deleted because it is redundant to ITS LCO 3.4.4, which requires four RCS loops Operable and in operation in Modes 1 and 2, and ITS LCO 3.4.5, which requires two RCS loops Operable in Mode 3. Although the requirements established in ITS LCO 3.4.4 and ITS LCO 3.4.5 are not intended to ensure minimum redundant decay heat removal capability, these Technical Specifications and associated Required Actions provide adequate assurance that the requirements of CTS 3.4.A.6 are satisfied at all times in Modes 1, 2 and 3.

ITS Requirements that maintain requirements in CTS 3.4.A.6

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.4	3.4.4	RCS Loops - MODES 1 and 2	2	None
3.4.5	3.4.5	RCS Loops - MODE 3	3	None

3.4.A.7 City Water

Status: Retained

Discussion

City Water is the backup to the Condensate Storage Tank (CST) as a water supply for the Auxiliary Feedwater System.

ITS Requirements that maintain requirements in CTS 3.4.A.7

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
IP3 ONLY	3.7.7	City Water	3	None

3.4.B Actions for CTS 3.4.A, Main Steam Safety Valves, Condensate Storage Tank, Main Steam Stops and City Water

Status:	Discussion
Retained	See CTS 3.4.A

ITS Requirements that maintain requirements in CTS 3.4.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.1	3.7.1	Main Steam Safety Valves (MSSVs)	3	None
3.7.2	3.7.2	Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)	3	None
3.7.6	3.7.6	Condensate Storage Tank (CST)	2, 3	None
IP3 ONLY	3.7.7	City Water	3	None

3.4.C Actions for CTS 3.4.A.2, Auxiliary Feedwater System

Status:	Discussion
Retained	See CTS 3.4.A.2.

ITS Requirements that maintain requirements in CTS 3.4.C

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.5	3.7.5	Auxiliary Feedwater (AFW) System	3	None

3.4.D Turbine Generator Electrical Output

Status:	Discussion
Relocated	See CTS Relocated Item R.7.

ITS Requirements that maintain requirements in CTS 3.4.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.7	R.7	STEAM AND POWER CONVERSION SYSTEM (Turbine Generator)	None	None

3.4.E Auxiliary Feedwater System Lineup and Associated Requried Actions

Status: Retained

Discussion

The Auxiliary Feedwater System functions to remove decay heat during design basis events thus mitigating consequences of events which could result in over pressurization of the RCS pressure boundary

ITS Requirements that maintain requirements in CTS 3.4.E

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.5	3.7.5	Auxiliary Feedwater (AFW) System	3	None

T 3.5-1 Engineered Safety Features Initiation Instrument Allowable Values

Status: Retained

Discussion

The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.

ITS Requirements that maintain requirements in CTS T 3.5-1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.3.5	3.3.5	Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	3	None

T 3.5-2 Reactor Trip Instrumentation Limiting Operating Conditions

Status: Retained

Discussion

The Reactor Trip System Instrumentation functions to maintain safety limits during operation and to mitigate the consequences of design basis events.

ITS Requirements that maintain requirements in CTS T 3.5-2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.1	3.3.1	Reactor Protection System (RPS) Instrumentation	3	None

T 3.5-3 Instrumentation Operating Condition for Engineered Safety Features

Status: Retained

Discussion

The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.

ITS Requirements that maintain requirements in CTS T 3.5-3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.3.5	3.3.5	Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	3	None
3.3.6	3.3.6	Containment Purge System and Pressure Relief Line Isolation Instrumentation	3	None
3.4.12	3.4.12	Low Temperature Overpressure Protection (LTOP)	2	Actuation instrumentation for the Power Oerater Relief Vavles in LTOP Mode are retained.
3.7.3	3.7.3	Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MBFRVs) and MBFRV Low Flow Bypass Valves	3	CTS requires that ESFAS signal isolate Feedwater but no CTS LCO governs the actuated devices.

T 3.5-4 Instrument Operating Conditions for Isolation Functions

Status: Retained/Relocated

Discussion

The Isolation Instrumentation functions to provide isolation of containment atmosphere and process systems that penetrate containment from the environment to limit the release of radioactivity following design basis accidents.

ITS Requirements that maintain requirements in CTS T 3.5-4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.3.3	3.3.3	Post Accident Monitoring (PAM) Instrumentation	3, 4	Main Steam Line Radiation Monitor is RG 1.97 Instrument.
3.3.6	3.3.6	Containment Purge System and Pressure Relief Line Isolation Instrumentation	3	None
R.8	R.8	AREA RADIATIN MOITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; Plant Wide Range Vent Monitor	None	Relocated to the ODCM

T 3.5-5 Table of Indicators and/or Recorders Available to the Operator

Status: Retained

Discussion

RG 1.97 Type A variables are retained because they provide information required for control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety function for DBAs.

RG 1.97 Category I variables are retained because of the following: needed to determine if systems important to safety are performing intended functions; provide information to determine the likelihood of a gross breach of the barriers to radioactivity release; and provide information regarding release of radioactive materials to allow early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.

ITS Requirements that maintain requirements in CTS T 3.5-5

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.3	3.3.3	Post Accident Monitoring (PAM) Instrumentation	3, 4	Variable that are neither Type A or Category I are relocated to the FSAR.

3.6 Containment System

Status: N/A

Discussion None

ITS Requirements that maintain requirements in CTS 3.6

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6	3.6	CONTAINMENT SYSTEMS	N/A	None

3.6.A Containment Integrity

Status: Retained

Discussion The containment functions to limit radioactive material released to the environment following design basis accidents.

ITS Requirements that maintain requirements in CTS 3.6.A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.1	3.6.1	Containment	3	None
3.6.2	3.6.2	Containment Air Locks	3	None
3.6.3	3.6.3	Containment Isolation Valves	3	None
3.9.1	3.9.1	Boron Concentration	2	Boron concentration requirements during refueling.
3.9.4	3.9.3	Containment Penetrations	3	None

3.6.B Internal Pressure

Status: Retained

Discussion The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a los of coolant accident (LOCA) or steam line break (SLB).

ITS Requirements that maintain requirements in CTS 3.6.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.4A	3.6.4	Containment Pressure	2	None

3.6.C Containment Temperature

Status: Retained

Discussion

The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

ITS Requirements that maintain requirements in CTS 3.6.C

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.5A	3.6.5	Containment Air Temperature	2	None

3.6.D Containment Vent [Pressure Relief] and Purge System

Status: Retained

Discussion

Containment Purge System supply and exhaust isolation are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 36 inch purge valves must be maintained sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Containment pressure relief line isolation valve opening is limited by mechanical stops so that opening angle is limited to an angle at which analysis indicates the valve will operate against containment accident pressures. Additionally, pressure relief isolation valve opening must be limited to the time necessary for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

ITS Requirements that maintain requirements in CTS 3.6.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.3	3.6.3	Containment Isolation Valves	3	None

3.7.A AC and DC Electrical Sources-Operating

Status: Retained

Discussion

The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.

ITS Requirements that maintain requirements in CTS 3.7.A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.8.1	3.8.1	AC Sources - Operating	3	None
3.8.3	3.8.3	Diesel Fuel Oil and Starting Air	3	None
3.8.4	3.8.4	DC Sources - Operating	3	None
3.8.6	3.8.6	Battery Cell Parameters	3	None
3.8.7	3.8.7	Inverters - Operating	3	None
3.8.9	3.8.9	Distribution Systems - Operating	3	None

3.7.B Actions for CTS 3.7.A

Status: Retained

Discussion

See CTS 3.7.A

ITS Requirements that maintain requirements in CTS 3.7.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

3.7.C Actions for Inoperable AC and DC Sources and Distribution

Status: N/A

Discussion See CTS 3.7.A.

ITS Requirements that maintain requirements in CTS 3.7.C

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

3.7.D Emergency System Wide Blackout

Status: Relocated

Discussion Relaxation of requirements for offsite power during an emergency system wide blackout.

ITS Requirements that maintain requirements in CTS 3.7.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
DETAIL REMOVED	Detail Removed	CTS Requirement retained in ITS but some details in CTS moved to Licensee controlled document. See Discussions of Change for ITS LCOs listed above.	N/A	Will be maintained in the FSAR

3.7.E A.C. Lighting Circuit Inside Primary Containment

Status: Relocated

Discussion See CTS Relocated Item R.9

ITS Requirements that maintain requirements in CTS 3.7.E

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.9	R.9	AUXILIARY ELECTRICAL SYSTEMS (A.C. Circuit Inside Containment)	None	None

3.7.F AC and DC Electrical Sources-Shutdown

Status: Retained

Discussion

The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of events during refueling and other shutdown conditions.

ITS Requirements that maintain requirements in CTS 3.7.F

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.8.2	3.8.2	AC Sources - Shutdown	3	None
3.8.5	3.8.5	DC Sources - Shutdown	3	None
3.8.8	3.8.8	Inverters - Shutdown	3	None
3.8.10	3.8.10	Distribution Systems - Shutdown	3	None

3.7.G Actions for Inoperable Offsite Circuits and Emergency Diesel Generators

Status: Retained

Discussion

This requirement only applies if a safeguards power train cannot be powered from an offsite source or the emergency diesel. This requirement ensures that an event coincident with a single failure of the associated DG or offsite source will not result in a complete loss of redundant required features. Required safety features are designed with a redundant safety feature that is powered from a different safeguards power train. Therefore, if a required safety feature is supported by an inoperable offsite circuit or inoperable diesel, then the failure of the DG or offsite source associated with that required safety feature will not result in the loss of a safety function because the safety function will be accomplished by the redundant safety feature that is powered from a different safeguards power train.

ITS Requirements that maintain requirements in CTS 3.7.G

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.8.1	3.8.1	AC Sources - Operating	3	Requirement Maintained by ITS LCO 3.8.1, Requried Actions A.3 and C.1.

3.8 Refueling, Fuel Handling and Storage

Status: **Discussion**

Retained

ITS Requirements that maintain requirements in CTS 3.8

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.9	3.9	REFUELING OPERATIONS	N/A	None

3.8.A Refueling Operations (Reactor Vessel & Containment)

Status: RETAINED

Discussion

The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.

ITS Requirements that maintain requirements in CTS 3.8.A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
1.0	**1.0**	USE AND APPLICATION	N/A	RCS temperature limit of 140 F relocated to FSAR.
3.3.6	**3.3.6**	Containment Purge System and Pressure Relief Line Isolation Instrumentation	3	None
3.3.8	**3.3.8**	Fuel Storage Building Emergency Ventilation System (FSBEVS) Instrumentation	3	None
3.7.13	**3.7.13**	Fuel Storage Building Emergency Ventilation System (FSBEVS)	3	None
3.9.3	**3.9.2**	Nuclear Instrumentation	3	Requirements for SRMs during refueling operations.
3.9.4	**3.9.3**	Containment Penetrations	3	Requirements for containment during refueling operations.
3.9.5	**3.9.4**	Residual Heat Removal (RHR) and Coolant Circulation - High Water Level	4	None
3.9.6	**3.9.5**	Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level	4	None
3.9.7	**3.9.6**	Refueling Cavity Water Level	2	None
R.8	**R.8**	AREA RADIATIN MOITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; Plant Wide Range Vent Monitor	None	None
R.10	**R.10**	Refueling, Fuel Handling and Storage (Communications)	None	None
R.11	**R.11**	Refueling, Fuel Handling and Storage (Decay Time)	None	None
R.12	**R.12**	Refueling (Manipulator Cranes and Spent Fuel Cask)	None	None

3.8.B Actions for CTS 3.8.A, Refueling Operations (Reactor Vessel & Containment)

Status:	Discussion
Retained	See CTS 3.8.A

ITS Requirements that maintain requirements in CTS 3.8.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)

Status:	Discussion
Retained	None

ITS Requirements that maintain requirements in CTS 3.8.C

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.13	3.7.13	Fuel Storage Building Emergency Ventilation System (FSBEVS)	3	None
3.7.15	3.7.14	Spent Fuel Pit Water Level	2, 3	None
3.7.16	3.7.15	Spent Fuel Pit Boron Concentration	2	None
3.7.17	3.7.16	Spent Fuel Assembly Storage	2	None
R.8	R.8	AREA RADIATIN MOITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; Plant Wide Range Vent Monitor	None	None
R.12	R.12	Refueling (Manipulator Cranes and Spent Fuel Cask)	None	None

3.8.D Boron Concentration during Refueling

Status: Retained

Discussion

The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains 0.95 % k/k during refueling operations.

ITS Requirements that maintain requirements in CTS 3.8.D

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.9.1	3.9.1	Boron Concentration	2	Boron concentration requirements retained; boron concentration limits relocated to the COLR.

3.9 Radioactive Materials Management

Status: Relocated

Discussion

See CTS Relocated Item R.14

ITS Requirements that maintain requirements in CTS 3.9

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.14	R.14	RADIOACTIVE MATERIALS MANAGEMENT	None	None

3.10.1 Shutdown Reactivity

Status: Retained

Discussion

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis 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. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis. The SDM requirement for refueling operations function to ensure the reactivity condition of the core is consistent with the applicable safety analysis and is conservative for MODE 6.

ITS Requirements that maintain requirements in CTS 3.10.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.1	3.1.1	SHUTDOWN MARGIN (SDM)	2	None
3.1.10	3.1.8	PHYSICS TESTS Exceptions MODE 2	1, 2, 3	None

3.10.2 Power Distribution Limits

Status: Retained

Discussion

The purpose of the limits on the values of Heat Flux Hot Channel Factor (FQ(Z)) is to limit the local (i.e., pellet) peak power density. Limits on FQ(Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid.

Limits on Nuclear Enthalpy Rise Hot Channel Factor preclude core power distributions that exceed for transients that may be DNB limited.

ITS Requirements that maintain requirements in CTS 3.10.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.2.1B	3.2.1	Heat Flux Hot Channel Factor (FQ(Z))	2	None
3.2.2	3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	2	None
3.2.3A	3.2.3	AXIAL FLUX DIFFERENCE (AFD)	2	None
3.2.4	3.2.4	QUADRANT POWER TILT RATIO (QPTR)	2	None

3.10.3 Quadrant Power Tilt Limits

Status: Retained

Discussion

The core power distribution limits function to preclude core power distributions that violate fuel design criteria.

ITS Requirements that maintain requirements in CTS 3.10.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.2.4	3.2.4	QUADRANT POWER TILT RATIO (QPTR)	2	None

3.10.4 Rod Insertion Limits

Status: Retained

Discussion

The core power distribution limits function to preclude core power distributions that violate fuel design criteria.

ITS Requirements that maintain requirements in CTS 3.10.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	None
3.1.6	3.1.5	Shutdown Bank Insertion Limits	2	None
3.1.7	3.1.6	Control Bank Insertion Limits	2	None
3.1.10	3.1.8	PHYSICS TESTS Exceptions MODE 2	1, 2, 3	None

3.10.5 Rod Misalignment Limitations

Status: Retained

Discussion

Control rod misalignment and insertion limits are included in the safety analysis. These limits ensure there be no violations of specified acceptable fuel design limits or Reactor Coolant System (RCS) pressure boundary integrity. Additionally, these limits ensure rhe core remains subcritical after accident transients.

ITS Requirements that maintain requirements in CTS 3.10.5

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	None

3.10.6 Inoperable Rod Position Indication Channels

Status: Retained

Discussion

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 if 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. The rod positions must also be known in order to verify the alignment limits are preserved.

ITS Requirements that maintain requirements in CTS 3.10.6

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.8	3.1.7	Rod Position Indication	2	None

3.10.7 Inoperable Rod Limitations

Status: Retained

Discussion

Control rod misalignment and insertion limits are included in the safety analysis. These limits ensure there be no violations of specified acceptable fuel design limits or Reactor Coolant System (RCS) pressure boundary integrity. Additionally, these limits ensure rhe core remains subcritical after accident transients.

ITS Requirements that maintain requirements in CTS 3.10.7

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	None
3.1.10	3.1.8	PHYSICS TESTS Exceptions MODE 2	1, 2, 3	None

3.10.8 Rod Drop Time

Status: Retained

Discussion

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.

ITS Requirements that maintain requirements in CTS 3.10.8

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	SR 3.1.4.3

3.10.9 Rod Position Monitor

Status: RETAINED

Discussion

ITS SR 3.1.4.1 maintains the requirement that rod position be verified every 12 hours regardless of the status of the deviation monitor. The ITS SR 3.1.4.1 Frequency of 12 hours for the verification of rod position recognizes that rod position information is continuously available to the operator in the control room, so that deviations can immediately be detected. Additionally, the requirement for accelerate verification if the rod position monitor is inoperable will be maintained in the FSAR and plant procedures.

ITS Requirements that maintain requirements in CTS 3.10.9

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	None

3.10.10 Reactivity Balance

Status: Retained

Discussion

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis. Accident analyses are, 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.

ITS Requirements that maintain requirements in CTS 3.10.10

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.3	3.1.2	Core Reactivity	2	None

3.11 Movable Incore Instrumentation

Status: Relocated

Discussion

See Relocated Item R.15

ITS Requirements that maintain requirements in CTS 3.11

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.1	3.3.1	Reactor Protection System (RPS) Instrumentation	3	Notes to ITS SR 3.3.1.3 and SR 3.3.1.6 maintain CTS allowance for calibration of axial offset detection.
R.15	R.15	MOVABLE INCORE INSTRUMENTATION	None	None

3.12 River Level

Status: Relocated

Discussion See Relocated Item R.16

ITS Requirements that maintain requirements in CTS 3.12

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.16	R.16	RIVER LEVEL (Flooding Protection)	None	None

3.13 Safety-Related Shock Suppressors (Snubbers)

Status: Relocated

Discussion See Relocated Item R.17

ITS Requirements that maintain requirements in CTS 3.13

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.17	R.17	SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)	None	None

3.14 Fire Protection (Deleted by Amendment 157)

Status: Relocated

Discussion Relocated by CTS Amendment 157.

ITS Requirements that maintain requirements in CTS 3.14

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

4.1 Operational Safety Review

Status: N/A

Discussion

General information related to performance of Surveillance Testing.

ITS Requirements that maintain requirements in CTS 4.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.0	3.0	LCO APPLICABILITY and SR APPLICABILITY	N/A	None

T 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

Status: Retained/Relocated

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS T 4.1-1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	None
3.1.8	3.1.7	Rod Position Indication	2	None
3.3.1	3.3.1	Reactor Protection System (RPS) Instrumentation	3	None
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.3.3	3.3.3	Post Accident Monitoring (PAM) Instrumentation	3, 4	None
3.3.5	3.3.5	Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	3	None
3.3.6	3.3.6	Containment Purge System and Pressure Relief Line Isolation Instrumentation	3	None
3.3.7	3.3.7	Control Room Emergency Ventilation (CRVS) Actuation Instrumentation	3	None
3.3.8	3.3.8	Fuel Storage Building Emergency Ventilation System (FSBEVS) Instrumentation	3	None
3.4.1	3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	2	None
3.4.2	3.4.2	RCS Minimum Temperature for Criticality	2	None
3.4.12	3.4.12	Low Temperature Overpressure Protection (LTOP)	2	None
3.4.15	3.4.15	RCS Leakage Detection Instrumentation	1	None
3.5.1	3.5.1	Accumulators	3	None
3.6.5A	3.6.5	Containment Air Temperature	2	Requirements for Containment Temperature monitoring instruments are relocated.

3.7.9	3.7.10	Ultimate Heat Sink (UHS)	3	Requirements for River Water Temperature monitoring instruments are relocated.
3.7.13	3.7.13	Fuel Storage Building Emergency Ventilation System (FSBEVS)	3	None
3.9.3	3.9.2	Nuclear Instrumentation	3	Mode 6 requirements
3.9.4	3.9.3	Containment Penetrations	3	None
R.8	R.8	AREA RADIATIN MOITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; Plant Wide Range Vent Monitor	None	None

T 4.1-2

Frequencies for Sampling Tests

Status:	Discussion
Retained/Relocated	CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS T 4.1-2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.16	3.4.16	RCS Specific Activity	2	None
3.5.1	3.5.1	Accumulators	3	None
3.5.4	3.5.4	Refueling Water Storage Tank (RWST)	3	None
3.6.7	3.6.7	Spray Additive System	3	None
3.7.7	3.7.8	Component Cooling Water (CCW) System	3	Water chemistry requriements are rlocated.
3.7.16	3.7.16	Spent Fuel Pit Boron Concentration	2	None
3.7.18	3.7.17	Secondary Specific Activity	2	None
R.4	R.4	MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION	None	None
R.5	R.5	CHEMICAL AND VOLUME CONTROL SYSTEM	None	None

T 4.1-3 Frequencies for Equipment Tests

Status: Retained/relocated

Discussion CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS T 4.1-3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.1.5	3.1.4	Rod Group Alignment Limits	2	None
3.4.10	3.4.10	Pressurizer Safety Valves	3	None
3.4.11	3.4.11	Pressurizer Power Operated Relief Valves (PORVs)	3	None
3.4.13	3.4.13	RCS Operational LEAKAGE	2	None
3.4.14	3.4.14	RCS Pressure Isolation Valve (PIV) Leakage	2	None
3.6.3	3.6.3	Containment Isolation Valves	3	None
3.7.1	3.7.1	Main Steam Safety Valves (MSSVs)	3	None
3.7.8	3.7.9	Service Water System (SWS)	3	None
3.8.3	3.8.3	Diesel Fuel Oil and Starting Air	3	None
R.1	R.1	REACTOR VESSEL HEAD VENTS	None	None
R.5	R.5	CHEMICAL AND VOLUME CONTROL SYSTEM	None	None
R.7	R.7	STEAM AND POWER CONVERSION SYSTEM (Turbine Generator)	None	None
R.12	R.12	Refueling (Manipulator Cranes and Spent Fuel Cask)	None	None

4.2 Inservice Inspection

Status:	Discussion
Retained	CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.8	5.5.7	Inservice Testing Program	N/A	None

4.3 Reactor Coolant System Integrity Testing

Status:	Discussion
Retained/Relocated	CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.1	3.4.1	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	2	None
3.4.3	3.4.3	RCS Pressure and Temperature (P/T) Limits	2	None
R.19	R.19	REACTOR COOLANT SYSTEM INTEGRITY TESTING	None	None

4.4 Containment Tests

Status: Retained/Relocated

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.1	3.6.1	Containment	3	None
3.6.2	3.6.2	Containment Air Locks	3	None
3.6.3	3.6.3	Containment Isolation Valves	3	None
IP3 UNIQUE 1x	3.6.9	Isolation Valve Seal Water (IVSW) System	3	None
5.5.2	5.5.2	Primary Coolant Sources Outside Containment	N/A	None
5.5.16	5.5.15	Containment Leakage Rate Testing Program	N/A	None
R.6	R.6	Weld Channel and Penetration Pressurization System (WC & PPS)	None	None
R.13	R.13	Service Water Isolation Valve Leakage (0.36 GPM Leakage Limit)	None	None

4.5 Tests for Engineered Safety Features and Air Filtration Systems

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

4.5.A.1 Safety Injection System

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.1	3.3.1	Reactor Protection System (RPS) Instrumentation	3	None
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.5.2	3.5.2	ECCS - Operating	3	None
3.5.3	3.5.3	ECCS - Shutdown	3	None
3.8.1	3.8.1	AC Sources - Operating	3	Time delay relays

4.5.A.2 Containment Spray System

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.6.6A	3.6.6	Containment Spray System and Containment Fan Cooler System	3	None
3.6.7	3.6.7	Spray Additive System	3	None

4.5.A.3 Containment Hydrogen Monitoring Systems

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.3	**3.3.3**	Post Accident Monitoring (PAM) Instrumentation	3, 4	None

4.5.A.4 Containment Air Filtration System

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.4

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.6A	**3.6.6**	Containment Spray System and Containment Fan Cooler System	3	None
5.5.11	**5.5.10**	Ventilation Filter Testing Program (VFTP)	N/A	None

4.5.A.5 Control Room Air Filtration System

Status: Retained/Relocated

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.5

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.10	**3.7.11**	Control Room Ventilation System (CRVS)	3	None
5.5.11	**5.5.10**	Ventilation Filter Testing Program (VFTP)	N/A	None
R.18	**R.18**	Toxic Gas Monitoring	None	None

4.5.A.6 Fuel Storage Building Emergency Ventilation System

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.6

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.13	**3.7.13**	Fuel Storage Building Emergency Ventilation System (FSBEVS)	3	None
5.5.11	**5.5.10**	Ventilation Filter Testing Program (VFTP)	N/A	None

4.5.A.7 Electric Hydrogen Recombiner Systems

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.A.7

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.6.8	**3.6.8**	Hydrogen Recombiners	3	None

4.5.B Component Tests

Status: Retained

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.5.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.4.14	**3.4.14**	RCS Pressure Isolation Valve (PIV) Leakage	2	None
3.5.1	**3.5.1**	Accumulators	3	None
3.5.2	**3.5.2**	ECCS - Operating	3	None
3.5.3	**3.5.3**	ECCS - Shutdown	3	None
3.6.6A	**3.6.6**	Containment Spray System and Containment Fan Cooler System	3	None
3.6.7	**3.6.7**	Spray Additive System	3	None

4.6.A Diesel Generators

Status: RETAINED

Discussion

CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.6.A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.8.1	**3.8.1**	AC Sources - Operating	3	None
3.8.2	**3.8.2**	AC Sources - Shutdown	3	None

4.6.B Station Batteries

Status: RETAINED

Discussion
CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.6.B

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.8.4	3.8.4	DC Sources - Operating	3	None
3.8.5	3.8.5	DC Sources - Shutdown	3	None
3.8.6	3.8.6	Battery Cell Parameters	3	None

4.7 Main Steam Stop Valves

Status: RETAINED

Discussion
CTS Surveillance Testing Requirements are maintained in the ITS.

ITS Requirements that maintain requirements in CTS 4.7

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	3	None
3.7.2	3.7.2	Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)	3	None

4.8 Auxiliary Feedwater System

Status: RETAINED

Discussion
None

ITS Requirements that maintain requirements in CTS 4.8

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.5	3.7.5	Auxiliary Feedwater (AFW) System	3	None
IP3 ONLY	3.7.7	City Water	3	None

4.9 Steam Generator Tube Inservice Surveillance

Status: RETAINED

Discussion None

ITS Requirements that maintain requirements in CTS 4.9

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.9	5.5.8	Steam Generator (SG) Tube Surveillance Program	N/A	None

4.10 Seismic Instrumentation

Status: Relocated

Discussion See Relocated Item R.20

ITS Requirements that maintain requirements in CTS 4.10

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.20	R.20	SEISMIC INSTRUMENTATION	None	None

4.11 Safety-Related Shock Suppressors (Snubbers)

Status: Relocated

Discussion See Relocated Item R.17

ITS Requirements that maintain requirements in CTS 4.11

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
R.17	R.17	SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)	None	None

4.12 Fire Protection and Detection Systems

Status: N/A

Discussion Relocated by CTS Amendment 157.

ITS Requirements that maintain requirements in CTS 4.12

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

4.13 Containment Vent (Pressure Relief) and Purge System

Status: Retained

Discussion None

ITS Requirements that maintain requirements in CTS 4.13

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.6	3.3.6	Containment Purge System and Pressure Relief Line Isolation Instrumentation	3	None
3.6.3	3.6.3	Containment Isolation Valves	3	None
3.9.4	3.9.3	Containment Penetrations	3	None
5.5.11	5.5.10	Ventilation Filter Testing Program (VFTP)	N/A	None

5.0 Design Features

Status: Retained

Discussion Application of TS Selection Criteria is not appropriate. However Design Features are included in ITS as required by 10 CFR 50.36. CT Section 5.0 review identified an LCO like requirement. See 5.4.2 and 5.4.3.

ITS Requirements that maintain requirements in CTS 5.0

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.7.16	3.7.15	Spent Fuel Pit Boron Concentration	2	None
4.0	4.0	DESIGN FEATURES	N/A	None

6.0 Administrative Controls

Status: Retained

Discussion

Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.3

ITS Requirements that maintain requirements in CTS 6.0

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
2.0	**2.0**	SAFETY LIMITS (SLs)	N/A	None
5.1	**5.1**	Responsibility	N/A	None
5.2	**5.2**	Organization	N/A	None
5.4	**5.4**	Procedures	N/A	None
5.5.1	**5.5.1**	Offsite Dose Calculation Manual (ODCM)	N/A	None
5.5.3	**5.5.3**	Post Accident Sampling	N/A	None
5.5.4	**5.5.4**	Radioactive Effluent Controls Program	N/A	None
5.5.16	**5.5.15**	Containment Leakage Rate Testing Program	N/A	None
5.6	**5.6**	Reporting Requirements	N/A	None
5.7	**5.7**	High Radiation Area	N/A	None
R.13	**R.13**	Service Water Isolation Valve Leakage (0.36 GPM Leakage Limit)	None	None
R.14	**R.14**	RADIOACTIVE MATERIALS MANAGEMENT	None	None
R.18	**R.18**	Toxic Gas Monitoring	None	None
R.20	**R.20**	SEISMIC INSTRUMENTATION	None	None

ETS 1.1/2.1

Explosive Gas Monitoring (TSCR 98-018)

Status:	Discussion
RETAINED	None

ITS Requirements that maintain requirements in CTS ETS 1.1/2.1

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.12	5.5.11	Explosive Gas and Storage Tank Radioactivity Monitoring Program	N/A	None

ETS 1.2/2.2

Radioactive Liquid Effluent Holdup Tanks (TSCR 98-018)

Status:	Discussion
RETAINED	None

ITS Requirements that maintain requirements in CTS ETS 1.2/2.2

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.12	5.5.11	Explosive Gas and Storage Tank Radioactivity Monitoring Program	N/A	None

ETS 1.3/2.3

Explosive Gas Mixture (TSCR 98-018)

Status:	Discussion
RETAINED	None

ITS Requirements that maintain requirements in CTS ETS 1.3/2.3

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.5.12	5.5.11	Explosive Gas and Storage Tank Radioactivity Monitoring Program	N/A	None

ETS 4.0

Administrative Controls (TSCR 98-018)

Status:	Discussion
RETAINED	None

ITS Requirements that maintain requirements in CTS ETS 4.0

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
5.1	5.1	Responsibility	N/A	None
5.4	5.4	Procedures	N/A	None
5.5.1	5.5.1	Offsite Dose Calculation Manual (ODCM)	N/A	None
5.5.4	5.5.4	Radioactive Effluent Controls Program	N/A	None
5.6	5.6	Reporting Requirements	N/A	None

NEW ITS

New Requirement imposed by ITS. No CTS Requirement.

Status:	Discussion
N/A	None

ITS Requirements that maintain requirements in CTS NEW ITS

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
3.3.4	3.3.4	Remote Shutdown Capability	4	FSAR requirements for remote shutdown capability are incorporated into the ITS.
3.7.4	3.7.4	Atmospheric Dump Valves (ADVs)	3	Assumed to be avalable for the mitigation of a Steam Generator Tube Rupture.
3.8.5	3.8.5	DC Sources - Shutdown	3	None
5.5.5	5.5.5	Component Cyclic or Transient Limit	N/A	FSAR requirements incorporated into the ITS.
5.5.7	5.5.6	Reactor Coolant Pump Flywheel Inspection Program	N/A	FSAR requirements incorporated into the ITS.
5.5.13	5.5.12	Diesel Fuel Oil Testing Program	N/A	Neither CTS nor FSAR establishe any requirements for testing of diesel fuel oil.
5.5.14	5.5.13	Technical Specifications (TS) Bases Control Program	N/A	None
5.5.15	5.5.14	Safety Function Determination Program (SFDP)	N/A	None

N/A

Not Applicable

Status:	Discussion
N/A	None

ITS Requirements that maintain requirements in CTS N/A

NR1431 No.	ITS No.	ITS NAME	10 CFR 50.36 Criterion	ITS NOTES
N/A	N/A	N/A	N/A	None

APPLICATION OF THE

NRC FINAL POLICY STATEMENT

SELECTION CRITERIA TO THE

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

TECHNICAL SPECIFICATIONS

ATTACHMENT 2:

ITS to CTS Disposition Matrix

INDIAN POINT 3
Conversion to Improved Technical Speifications
ITS to CTS Disposition Matrix

ITS No.	NR1431 No.	INDIAN POINT 3: Current Technical Specification Disposition and Relocation Matrix

1.0 USE AND APPLICATION

10 CFR 50.36 CRITERION: N/

1.0 USE AND APPLICATION

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
1.0 Definitions	Definitions are provided to improve understanding and ensure consistent application. Application of the TS selection criteria to these definitions is not appropriate.	None
License 2.C.1 License Condition: Maximum Power Level 3025 mW thermal (100% of rated power)	Maximum Power Level will be maintained as both a License Condition and as a Technical Specification Safety Limit.	None
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	RCS temperature limit of 140 F relocated to FS

2.0 SAFETY LIMITS (SLS)

10 CFR 50.36 CRITERION: N/

2.0 SAFETY LIMITS (SLS)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
2.0 Safety Limits and Limiting Safety System Settings	10 CFR 50.36 requires that Safety Limits are included in the Technical Specifications.	None
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

2.1.1 SAFETY LIMITS

10 CFR 50.36 CRITERION: N/

2.1.1 SAFETY LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
2.1 Safety Limits, Reactor Core (Reactor Power, Pressure and Temperature)	Application of TS selection criteria to Safety Limits is not appropriate. These safety limits are retained in the ITS.	None

2.1.2 SAFETY LIMITS

10 CFR 50.36 CRITERION: N/

2.1.2 SAFETY LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
2.2 Safety Limit, Reactor Coolant System Pressure	Application of TS selection criteria to Safety Limits is not appropriate. These safety limits are retained in the ITS.	None

2.2 SAFETY LIMIT VIOLATIONS

10 CFR 50.36 CRITERION: N/

2.2

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.7 Safety Limit Violation	None	None

3.0 LCO APPLICABILITY AND SR APPLICABILITY

10 CFR 50.36 CRITERION: N/

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.1 Operational Safety Review	General information related to performance of Surveillance Testing.	None
3.0 Limiting Conditions for Operation	This Specification provides guidance applicable to one or more Limiting Conditions for Operation (LCOs). The requirements of CTS Section 3 are expanded in ITS.	None

3.1 REACTIVITY CONTROL SYSTEMS

10 CFR 50.36 CRITERION: N/

3.1 REACTIVITY CONTROL SYSTEMS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.C Minimum Conditions for Criticality	Establishes operating restrictions such that operation is bounded by the accident analysis.	None

3.1.1 SHUTDOWN MARGIN (SDM)

10 CFR 50.36 CRITERION: 2

3.1.1 SHUTDOWN MARGIN (SDM) TAVG > 200 F

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.1 Shutdown Reactivity	The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis 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. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis. The SDM requirement for refueling operations function to ensure the reactivity condition of the core is consistent with the applicable safety analysis and is conservative for MODE 6.	None

3.1.2 CORE REACTIVITY

10 CFR 50.36 CRITERION: 2

3.1.3 CORE REACTIVITY

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.10 Reactivity Balance	Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis. Accident analyses are, 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.	None

3.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

10 CFR 50.36 CRITERION: 2

3.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.C.1 Moderator Temperature Coefficient	The FSAR contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding.	ITS limits both he most positive value and most negative value of the MTC.

3.1.4 ROD GROUP ALIGNMENT LIMITS

10 CFR 50.36 CRITERION: 2

3.1.5 ROD GROUP ALIGNMENT LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.4 Rod Insertion Limits	The core power distribution limits function to preclude core power distributions that violate fuel design criteria.	None
3.10.5 Rod Misalignment Limitations	Control rod misalignment and insertion limits are included in the safety analysis. These limits ensure there be no violations of specified acceptable fuel design limits or Reactor Coolant System (RCS) pressure boundary integrity. Additionally, these limits ensure rhe core remains subcritical after accident transients.	None
3.10.7 Inoperable Rod Limitations	Control rod misalignment and insertion limits are included in the safety analysis. These limits ensure there be no violations of specified acceptable fuel design limits or Reactor Coolant System (RCS) pressure boundary integrity. Additionally, these limits ensure rhe core remains subcritical after accident transients.	None
3.10.8 Rod Drop Time	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.	SR 3.1.4.3
3.10.9 Rod Position Monitor	ITS SR 3.1.4.1 maintains the requirement that rod position be verified every 12 hours regardless of the status of the deviation monitor. The ITS SR 3.1.4.1 Frequency of 12 hours for the verification of rod position recognizes that rod position information is continuously available to the operator in the control room, so that deviations can immediately be detected. Additionally, the requirement for accelerated verification if the rod position monitor is inoperable will be maintained in the FSAR and plant procedures.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.1.5 SHUTDOWN BANK INSERTION LIMITS

10 CFR 50.36 CRITERION: 2

3.1.6 SHUTDOWN BANK INSERTION LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.4 Rod Insertion Limits	The core power distribution limits function to preclude core power distributions that violate fuel design criteria.	None

3.1.6 CONTROL BANK INSERTION LIMITS

10 CFR 50.36 CRITERION: 2

3.1.7 CONTROL BANK INSERTION LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.4 Rod Insertion Limits	The core power distribution limits function to preclude core power distributions that violate fuel design criteria.	None

3.1.7 ROD POSITION INDICATION

10 CFR 50.36 CRITERION: 2

3.1.8 ROD POSITION INDICATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.6 Inoperable Rod Position Indication Channels	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 if 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. The rod positions must also be known in order to verify the alignment limits are preserved.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.1.8 PHYSICS TESTS EXCEPTIONS MODE 2

10 CFR 50.36 CRITERION: 1, 2,

3.1.10 PHYSICS TESTS EXCEPTIONS MODE 2

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.4 Rod Insertion Limits	The core power distribution limits function to preclude core power distributions that violate fuel design criteria.	None
3.10.1 Shutdown Reactivity	The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis 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. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis. The SDM requirement for refueling operations function to ensure the reactivity condition of the core is consistent with the applicable safety analysis and is conservative for MODE 6.	None
3.1.C.3 Minimum Temperature for Criticality	Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.	None
3.10.7 Inoperable Rod Limitations	Control rod misalignment and insertion limits are included in the safety analysis. These limits ensure there be no violations of specified acceptable fuel design limits or Reactor Coolant System (RCS) pressure boundary integrity. Additionally, these limits ensure rhe core remains subcritical after accident transients.	None

3.2.1 HEAT FLUX HOT CHANNEL FACTOR (FQ(Z))

10 CFR 50.36 CRITERION: 2

3.2.1B HEAT FLUX HOT CHANNEL FACTOR (FQ(Z)) (FQ METHODOLOGY)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.2 Power Distribution Limits	The purpose of the limits on the values of Heat Flux Hot Channel Factor (FQ(Z)) is to limit the local (i.e., pellet) peak power density. Limits on FQ(Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Limits on Nuclear Enthalpy Rise Hot Channel Factor preclude core power distributions that exceed for transients that may be DNB limited.	None

3.2.2 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

10 CFR 50.36 CRITERION: 2

3.2.2 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (FN H)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.2 Power Distribution Limits	The purpose of the limits on the values of Heat Flux Hot Channel Factor (FQ(Z)) is to limit the local (i.e., pellet) peak power density. Limits on FQ(Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Limits on Nuclear Enthalpy Rise Hot Channel Factor preclude core power distributions that exceed for transients that may be DNB limited.	None

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

10 CFR 50.36 CRITERION: 2

3.2.3A AXIAL FLUX DIFFERENCE (AFD) (CONSTANT AXIAL OFFSET CONTROL (CAOC) METHODOLOGY)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.2 Power Distribution Limits	The purpose of the limits on the values of Heat Flux Hot Channel Factor (FQ(Z)) is to limit the local (i.e., pellet) peak power density. Limits on FQ(Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Limits on Nuclear Enthalpy Rise Hot Channel Factor preclude core power distributions that exceed for transients that may be DNB limited.	None

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

10 CFR 50.36 CRITERION: 2

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.10.3 Quadrant Power Tilt Limits	The core power distribution limits function to preclude core power distributions that violate fuel design criteria.	None
3.10.2 Power Distribution Limits	The purpose of the limits on the values of Heat Flux Hot Channel Factor (FQ(Z)) is to limit the local (i.e., pellet) peak power density. Limits on FQ(Z) ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Limits on Nuclear Enthalpy Rise Hot Channel Factor preclude core power distributions that exceed for transients that may be DNB limited.	None

3.3.1 REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.3.1 REACTOR TRIP SYSTEM (RTS) INSTRUMENTATION

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
4.5.A.1 Safety Injection System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.11 Movable Incore Instrumentation	See Relocated Item R.15	Notes to ITS SR 3.3.1.3 and SR 3.3.1.6 maintai CTS allowance for calibration of axial offset detection.
T 3.5-2 Reactor Trip Instrumentation Limiting Operating Conditions	The Reactor Trip System Instrumentation functions to maintain safety limits during operation and to mitigate the consequences of design basis events.	None
2.3 Limiting Safety System Settings, Protective Instrumentation	The limiting safety settings for protective instrumentation function to actuate the Reactor Protection System (RPS) to mitigate the consequences of Design Basis Accidents (DBAs) and/or transients.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
4.5.A.2 Containment Spray System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 3.5-1 Engineered Safety Features Initiation Instrument Allowable Values	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	None
T 3.5-3 Instrumentation Operating Condition for Engineered Safety Features	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	None
4.7 Main Steam Stop Valves	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 3.5-4 Instrument Operating Conditions for Isolation Functions	The Isolation Instrumentation functions to provide isolation of containment atmosphere and process systems that penetrate containment from the environment to limit the release of radioactivity following design basis accidents.	None
4.5.A.1 Safety Injection System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.3.3 POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

10 CFR 50.36 CRITERION: 3,

3.3.3 POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.A.3 Containment Hydrogen Monitoring Systems	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 3.5-4 Instrument Operating Conditions for Isolation Functions	The Isolation Instrumentation functions to provide isolation of containment atmosphere and process systems that penetrate containment from the environment to limit the release of radioactivity following design basis accidents.	Main Steam Line Radiation Monitor is RG 1.97 Instrument.
3.3.G Containment Hydrogen Monitoring Systems	Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.	None
T 3.5-5 Table of Indicators and/or Recorders Available to the Operator	RG 1.97 Type A variables are retained because they provide information required for control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs. RG 1.97 Category I variables are retained because of the following: needed to determine if systems important to safety are performing intended functions; provide information to determine the likelihood of a gross breach of the barriers to radioactivity release; and provide information regarding release of radioactive materials to allow early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.	Variable that are neither Type A or Category I a relocated to the FSAR.
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	The IP3 design does not include automatic switchover from SI mode to recirc mode based RWST level. This function is performed manual based on RWST level alarm and containment sump level.

3.3.4 REMOTE SHUTDOWN CAPABILITY

10 CFR 50.36 CRITERION: 4

3.3.4 REMOTE SHUTDOWN SYSTEM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	FSAR requirements for remote shutdown capability are incorporated into the ITS.

3.3.5 LOSS OF POWER (LOP) DIESEL GENERATOR (DG) START INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.3.5 LOSS OF POWER (LOP) DIESEL GENERATOR (DG) START INSTRUMENTATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 3.5-3 Instrumentation Operating Condition for Engineered Safety Features	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	None
T 3.5-1 Engineered Safety Features Initiation Instrument Allowable Values	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.3.6 CONTAINMENT PURGE SYSTEM AND PRESSURE RELIEF LINE ISOLATION INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.3.6 CONTAINMENT PURGE AND EXHAUST ISOLATION INSTRUMENTATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.13 Containment Vent (Pressure Relief) and Purge System	None	None
T 3.5-4 Instrument Operating Conditions for Isolation Functions	The Isolation Instrumentation functions to provide isolation of containment atmosphere and process systems that penetrate containment from the environment to limit the release of radioactivity following design basis accidents.	None
T 3.5-3 Instrumentation Operating Condition for Engineered Safety Features	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	None
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.3.7 CONTROL ROOM EMERGENCY VENTILATION (CRVS) ACTUATION INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.3.7 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) ACTUATION INSTRUMENTATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.H Control Room Ventilation System (including toxic gas monitoring)	Control room ventilation provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CRACS is designed so that the functional capability of the Control Room is maintained during a Design Basis Accident. Functional capability means that the ambient temperature for safety equipment located in this room will not exceed 108.2 F. Control Room safety equipment is specified to a temperature of 120 F and the 108.2 F limit for Control Room temperature is sufficient to account for the temperature rise in the enclosed cabinets.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.3.8 FUEL STORAGE BUILDING EMERGENCY VENTILATION SYSTEM (FSBEVS) INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.3.8 FUEL BUILDING AIR CLEANUP SYSTEM (FBACS) ACTUATION INSTRUMENTATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4 REACTOR COOLANT SYSTEM (RCS)

10 CFR 50.36 CRITERION: N/

3.4 REACTOR COOLANT SYSTEM (RCS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.C Minimum Conditions for Criticality	Establishes operating restrictions such that operation is bounded by the accident analysis.	None
3.1.A Operational Components	None	None

3.4.1 RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

10 CFR 50.36 CRITERION: 2

3.4.1 RCS PRESSURE, TEMPERATURE, AND FLOW DEPARTURE FROM NUCLEATE BOILING (DNB) LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.H RCS Pressure, Temperature, and Flow DNB Limits	The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses. The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed include loss of coolant flow events and dropped or stuck rod events.	None
4.3 Reactor Coolant System Integrity Testing	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.2 RCS MINIMUM TEMPERATURE FOR CRITICALITY

10 CFR 50.36 CRITERION: 2

3.4.2 RCS MINIMUM TEMPERATURE FOR CRITICALITY

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.C.3 Minimum Temperature for Criticality	Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.3 RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

10 CFR 50.36 CRITERION: 2

3.4.3 RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.B Heatup and Cooldown	This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. These limits are needed to avoid encountering pressure, temperature, or temperature rate of change condition that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary.	None
4.3 Reactor Coolant System Integrity Testing	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.4 RCS LOOPS - MODES 1 AND 2

10 CFR 50.36 CRITERION: 2

3.4.4 RCS LOOPS MODES 1 AND 2

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
License 2.C.3 License Condition: Less Than Four Loop Operation	License Condition 2.C.3 allows 3 loop operation below the P-7 setpoint (approximately 10% RTP). ITS LCO 3.4.4 requires four RCS loops OPERABLE and in operation when in Modes 1 and 2. License Condition 2.C.3 conflicts with ITS LCO 3.4.4 and must be deleted.	None
3.4.A.6 Steam Generators (Two SGs for decay heat removal)	This requirement is deleted because it is redundant to ITS LCO 3.4.4, which requires four RCS loops Operable and in operation in Modes 1 and 2, and ITS LCO 3.4.5, which requires two RCS loops Operable in Mode 3. Although the requirements established in ITS LCO 3.4.4 and ITS LCO 3.4.5 are not intended to ensure minimum redundant decay heat removal capability, these Technical Specifications and associated Required Actions provide adequate assurance that the requirements of CTS 3.4.A.6 are satisfied at all times in Modes 1, 2 and 3.	None
3.1.A.1 Coolant Pumps	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.	None

3.4.5 RCS LOOPS - MODE 3

10 CFR 50.36 CRITERION: 3

3.4.5 RCS LOOPS MODE 3

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.A.1 Coolant Pumps	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.	None
3.4.A.6 Steam Generators (Two SGs for decay heat removal)	This requirement is deleted because it is redundant to ITS LCO 3.4.4, which requires four RCS loops Operable and in operation in Modes 1 and 2, and ITS LCO 3.4.5, which requires two RCS loops Operable in Mode 3. Although the requirements established in ITS LCO 3.4.4 and ITS LCO 3.4.5 are not intended to ensure minimum redundant decay heat removal capability, these Technical Specifications and associated Required Actions provide adequate assurance that the requirements of CTS 3.4.A.6 are satisfied at all times in Modes 1, 2 and 3.	None

3.4.6 RCS LOOPS - MODE 4

10 CFR 50.36 CRITERION: 4

3.4.6 RCS LOOPS MODE 4

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.A.1 Coolant Pumps	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.	None
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None

3.4.7 RCS LOOPS - MODE 5, LOOPS FILLED

10 CFR 50.36 CRITERION: 4

3.4.7 RCS LOOPS MODE 5, LOOPS FILLED

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.A.1 Coolant Pumps	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.	None
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None

3.4.8 RCS LOOPS - MODE 5, LOOPS NOT FILLED

10 CFR 50.36 CRITERION: 4

3.4.8 RCS LOOPS MODE 5, LOOPS NOT FILLED

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None
3.1.A.1 Coolant Pumps	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.	None

3.4.9 PRESSURIZER

10 CFR 50.36 CRITERION: 2

3.4.9 PRESSURIZER

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.C.4 Pressurizer Water Level	All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer.	CTS requires Pressurizer at the normal level; I requires only that there is a bubble in the pressurizer.
3.1.A.3 Pressurizer Heaters	A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant.	None

3.4.10 PRESSURIZER SAFETY VALVES

10 CFR 50.36 CRITERION: 3

3.4.10 PRESSURIZER SAFETY VALVES

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.A.2 Safety Valves (Pressurizer)	The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS.	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.11 PRESSURIZER POWER OPERATED RELIEF VALVES (PORVS)

10 CFR 50.36 CRITERION: 3

3.4.11 PRESSURIZER POWER OPERATED RELIEF VALVES (PORVS)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.1.A.5 Power Operated Relief Block Valves	The block valves are used to isolate the PORVs in case of the associated PORV has excessive leakage or is stuck open.	None
3.1.A.4 Power Operated Relief Valves	Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal and alternate pressurizer spray are not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are incorporated into the TS in response to GL 90-06.	None

3.4.12 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

10 CFR 50.36 CRITERION: 2

3.4.12 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.A.1 Coolant Pumps	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses.	LTOP requirements are enforced during RCP starts below LTOP enable temperature.
3.1.A.8 Overpressure Protection System (OPS)	LTOP is established to limit RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. In MODES 1, 2, and 3, with RCS cold leg temperature exceeding 411 F, the pressurizer safety valves will prevent RCS pressure from exceeding Appendix G limits. At 319 F and below, overpressure prevention must be ensured by two OPERABLE PORVs in conjunction with the Overpressure Protection System or to a depressurized RCS and a sufficient sized RCS vent.	None
T 3.5-3 Instrumentation Operating Condition for Engineered Safety Features	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	Actuation instrumentation for the Power Oerate Relief Vavles in LTOP Mode are retained.
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	LTOP requirements place restrictions on bothECCS and RHR lineups.
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.13 RCS OPERATIONAL LEAKAGE

10 CFR 50.36 CRITERION: 2

3.4.13 RCS OPERATIONAL LEAKAGE

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.1.F Leakage of Reactor Coolant	Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, operational LEAKAGE is related to the safety analyses for LOCA because the amount of leakage can affect the probability of such an event. Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR).	None

3.4.14 RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

10 CFR 50.36 CRITERION: 2

3.4.14 RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.F Leakage of Reactor Coolant	Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, operational LEAKAGE is related to the safety analyses for LOCA because the amount of leakage can affect the probability of such an event. Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR).	ITS retains the CTS allowance that leakage into closed systems is not counted as either identifi or unidentified leakage.
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.B Component Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.15 RCS LEAKAGE DETECTION INSTRUMENTATION

10 CFR 50.36 CRITERION: 1

3.4.15 RCS LEAKAGE DETECTION INSTRUMENTATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.4.16 RCS SPECIFIC ACTIVITY

10 CFR 50.36 CRITERION: 2

3.4.16 RCS SPECIFIC ACTIVITY

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.1.D Primary Coolant Activity	The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.	None

3.5.1 ACCUMULATORS

10 CFR 50.36 CRITERION: 3

3.5.1 ACCUMULATORS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.B Component Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None

3.5.2 ECCS - OPERATING

10 CFR 50.36 CRITERION: 3

3.5.2 ECCS OPERATING

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None
4.5.B Component Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.A.1 Safety Injection System	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.5.3 ECCS - SHUTDOWN

10 CFR 50.36 CRITERION: 3

3.5.3 ECCS SHUTDOWN

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
4.5.A.1 Safety Injection System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None
4.5.B Component Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.5.4 REFUELING WATER STORAGE TANK (RWST)

10 CFR 50.36 CRITERION: 3

3.5.4 REFUELING WATER STORAGE TANK (RWST)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.A Safety Injection and Residual Heat Removal Systems	ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The primary function of the RHR system is the transfer of decay heat to either the SG secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the RHR system is to carry and ensure mixing of the soluble neutron poison, boric acid.	None

3.6 CONTAINMENT SYSTEMS

10 CFR 50.36 CRITERION: N/

3.6 CONTAINMENT SYSTEMS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.6 Containment System	None	None

3.6.1 CONTAINMENT

10 CFR 50.36 CRITERION: 3

3.6.1 CONTAINMENT (ATMOSPHERIC, SUBATMOSPHERIC, ICE CONDENSER, AND DUAL)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.6.A Containment Integrity	The containment functions to limit radioactive material released to the environment following design basis accidents.	None
1.10 Containment Integrity Definition	CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. Requirements are maintained as Limiting Conditions of Operation listed below.	None
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.6.2 CONTAINMENT AIR LOCKS

10 CFR 50.36 CRITERION: 3

3.6.2 CONTAINMENT AIR LOCKS (ATMOSPHERIC, SUBATMOSPHERIC, ICE CONDENSER, AND DUAL)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
1.10 Containment Integrity Definition	CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. Requirements are maintained as Limiting Conditions of Operation listed below.	None
3.6.A Containment Integrity	The containment functions to limit radioactive material released to the environment following design basis accidents.	None

3.6.3 CONTAINMENT ISOLATION VALVES

10 CFR 50.36 CRITERION: 3

3.6.3 CONTAINMENT ISOLATION VALVES (ATMOSPHERIC, SUBATMOSPHERIC, ICE CONDENSER, AND DUAL)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
1.10 Containment Integrity Definition	CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. Requirements are maintained as Limiting Conditions of Operation listed below.	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.13 Containment Vent (Pressure Relief) and Purge System	None	None
3.6.D Containment Vent [Pressure Relief] and Purge System	Containment Purge System supply and exhaust isolation are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 36 inch purge valves must be maintained sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. Containment pressure relief line isolation valve opening is limited by mechanical stops so that opening angle is limited to an angle at which analysis indicates the valve will operate against containment accident pressures. Additionally, pressure relief isolation valve opening must be limited to the time necessary for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	None
3.6.A Containment Integrity	The containment functions to limit radioactive material released to the environment following design basis accidents.	None

3.6.4 CONTAINMENT PRESSURE

10 CFR 50.36 CRITERION: 2

3.6.4A CONTAINMENT PRESSURE (ATMOSPHERIC, DUAL, AND ICE CONDENSER)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.6.B Internal Pressure	The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).	None

3.6.5 CONTAINMENT AIR TEMPERATURE

10 CFR 50.36 CRITERION: 2

3.6.5A CONTAINMENT AIR TEMPERATURE (ATMOSPHERIC AND DUAL)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	Requirements for Containment Temperature monitoring instruments are relocated.
3.6.C Containment Temperature	The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).	None

3.6.6 CONTAINMENT SPRAY SYSTEM AND CONTAINMENT FAN COOLER SYSTEM

10 CFR 50.36 CRITERION: 3

3.6.6A CONTAINMENT SPRAY AND COOLING SYSTEMS (ATMOSPHERIC AND DUAL) (CREDIT TAKEN FOR CONTAINMENT SPRAY)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.5.A.2 Containment Spray System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.B Component Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.B Containment Cooling and Iodine Removal Systems	The Containment Spray System and Containment Fan Cooler System provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal reduces the release of radioactivity to the environment following a DBA. The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a DBA.	None
4.5.A.4 Containment Air Filtration System	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.6.7 SPRAY ADDITIVE SYSTEM

10 CFR 50.36 CRITERION: 3

3.6.7 SPRAY ADDITIVE SYSTEM (ATMOSPHERIC, SUBATMOSPHERIC, ICE CONDENSER, AND DUAL)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
4.5.A.2 Containment Spray System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.B Containment Cooling and Iodine Removal Systems	The Containment Spray System and Containment Fan Cooler System provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal reduces the release of radioactivity to the environment following a DBA. The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a DBA.	None
4.5.B Component Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.6.8 HYDROGEN RECOMBINERS

10 CFR 50.36 CRITERION: 3

3.6.8 HYDROGEN RECOMBINERS (ATMOSPHERIC, SUBATMOSPHERIC, ICE CONDENSER, AND DUAL)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.I Electric Hydrogen Recombiner System	Hydrogen control ensures that pressure and temperature assumed in the analyses are not exceeded during a post LOCA hydrogen burn.	None
4.5.A.7 Electric Hydrogen Recombiner Systems	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.6.9 ISOLATION VALVE SEAL WATER (IVSW) SYSTEM

10 CFR 50.36 CRITERION: 3

IP3 UNIQUE 1x — IP3 UNIQUE

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.C Isolation Valve Seal Water System (IVSWS)	The IVSW System is a seal system as described in 10 CFR 50, Appendix J. The IVSW System improves the effectiveness of certain containment isolation valves by providing a water seal to valve leakage paths.	None
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.7 PLANT SYSTEMS

10 CFR 50.36 CRITERION: N/

3.7 — PLANT SYSTEMS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.4 Steam and Power Conversion System	None	None

3.7.1 MAIN STEAM SAFETY VALVES (MSSVS)

10 CFR 50.36 CRITERION: 3

3.7.1 — MAIN STEAM SAFETY VALVES (MSSVS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.4.A.1 Main Steam Safety Valves	The purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary by providing a heat sink for the removal of energy from the RCS.	None
3.4.B Actions for CTS 3.4.A, Main Steam Safety Valves, Condensate Storage Tank, Main Steam Stops and City Water	See CTS 3.4.A	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.7.2 MAIN STEAM ISOLATION VALVES (MSIVS) AND MAIN STEAM CHECK VALVES (MSCVS)

10 CFR 50.36 CRITERION: 3

3.7.2 MAIN STEAM ISOLATION VALVES (MSIVS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.7 Main Steam Stop Valves	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.4.A.5 Main Steam Stop Valves	The combination of MSIVs and MSCVs precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). For a break upstream of an MSIV, either the MSIVs in the other three steam lines or the MSCV in the steam line with the faulted SG must close to prevent the blowdown of more than one SG. For a break downstream of an MSIV, the MSCVs are not required to function.	None
3.4.B Actions for CTS 3.4.A, Main Steam Safety Valves, Condensate Storage Tank, Main Steam Stops and City Water	See CTS 3.4.A	None

3.7.3 MAIN BOILER FEEDPUMP DISCHARGE VALVES (MBFPDVS), MAIN FEEDWATER REGULATION VALVES (MBFRVS) AND MBFRV LOW FLOW BYPASS VALVES

10 CFR 50.36 CRITERION: 3

3.7.3 MAIN FEEDWATER ISOLATION VALVES (MFIVS) AND MAIN FEEDWATER REGULATION VALVES (MFRVS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 3.5-3 Instrumentation Operating Condition for Engineered Safety Features	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.	CTS requires that ESFAS signal isolate Feedw but no CTS LCO governs the actuated devices.

3.7.4 ATMOSPHERIC DUMP VALVES (ADVS)

10 CFR 50.36 CRITERION: 3

3.7.4 ATMOSPHERIC DUMP VALVES (ADVS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	Assumed to be avalable for the mitigation of a Steam Generator Tube Rupture.

3.7.5 AUXILIARY FEEDWATER (AFW) SYSTEM

10 CFR 50.36 CRITERION: 3

3.7.5 AUXILIARY FEEDWATER (AFW) SYSTEM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.4.C Actions for CTS 3.4.A.2, Auxiliary Feedwater System	See CTS 3.4.A.2.	None
3.4.E Auxiliary Feedwater System Lineup and Associated Requried Actions	The Auxiliary Feedwater System functions to remove decay heat during design basis events thus mitigating consequences of events which could result in over pressurization of the RCS pressure boundary	None
4.8 Auxiliary Feedwater System	None	None
3.4.A.2 Auxiliary Feedwater Pumps	The Auxiliary Feedwater System functions to remove decay heat during design basis events thus mitigating consequences of events which could result in over pressurization of the RCS pressure boundary.	None

3.7.6 CONDENSATE STORAGE TANK (CST)

10 CFR 50.36 CRITERION: 2,

3.7.6 CONDENSATE STORAGE TANK (CST)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.4.A.3 Condensate Storage Tank	The Condensate Storage Tank or city water function to provide cooling water to remove decay heat and cool down the unit for design basis events.	None
3.4.B Actions for CTS 3.4.A, Main Steam Safety Valves, Condensate Storage Tank, Main Steam Stops and City Water	See CTS 3.4.A	None

3.7.7 CITY WATER

10 CFR 50.36 CRITERION: 3

IP3 ONLY IP3 UNIQUE

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.4.A.7 City Water	City Water is the backup to the Condensate Storage Tank (CST) as a water supply for the Auxiliary Feedwater System.	None
4.8 Auxiliary Feedwater System	None	None
3.4.B Actions for CTS 3.4.A, Main Steam Safety Valves, Condensate Storage Tank, Main Steam Stops and City Water	See CTS 3.4.A	None

3.7.8 COMPONENT COOLING WATER (CCW) SYSTEM

10 CFR 50.36 CRITERION: 3

3.7.7 COMPONENT COOLING WATER (CCW) SYSTEM

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.E Component Cooling System	The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient.	None
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	Water chemistry requriements are rlocated.

3.7.9 SERVICE WATER SYSTEM (SWS)

10 CFR 50.36 CRITERION: 3

3.7.8 SERVICE WATER SYSTEM (SWS)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.F Service Water System/Ultimate Heat Sink	The Service Water System (SWS) and UHS functions in conjunction with the CCS to remove post LOCA heat loads from the containment sump during the recirculation phase. The SWS in conjunction with CCS also functions to cool the unit from RHR entry conditions to cold shutdown during normal and post accident conditions.	None

3.7.10 ULTIMATE HEAT SINK (UHS)

10 CFR 50.36 CRITERION: 3

3.7.9 ULTIMATE HEAT SINK (UHS)

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	Requirements for River Water Temperature monitoring instruments are relocated.
3.3.F Service Water System/Ultimate Heat Sink	The Service Water System (SWS) and UHS functions in conjunction with the CCS to remove post LOCA heat loads from the containment sump during the recirculation phase. The SWS in conjunction with CCS also functions to cool the unit from RHR entry conditions to cold shutdown during normal and post accident conditions.	None

3.7.11 CONTROL ROOM VENTILATION SYSTEM (CRVS)

10 CFR 50.36 CRITERION: 3

3.7.10 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.H Control Room Ventilation System (including toxic gas monitoring)	Control room ventilation provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CRACS is designed so that the functional capability of the Control Room is maintained during a Design Basis Accident. Functional capability means that the ambient temperature for safety equipment located in this room will not exceed 108.2 F. Control Room safety equipment is specified to a temperature of 120 F and the 108.2 F limit for Control Room temperature is sufficient to account for the temperature rise in the enclosed cabinets.	None
4.5.A.5 Control Room Air Filtration System	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.7.12 CONTROL ROOM AIR CONDITIONING SYSTEM (CRACS)

10 CFR 50.36 CRITERION: 3

3.7.11 CONTROL ROOM EMERGENCY AIR TEMPERATURE CONTROL SYSTEM (CREATCS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.H Control Room Ventilation System (including toxic gas monitoring)	Control room ventilation provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CRACS is designed so that the functional capability of the Control Room is maintained during a Design Basis Accident. Functional capability means that the ambient temperature for safety equipment located in this room will not exceed 108.2 F. Control Room safety equipment is specified to a temperature of 120 F and the 108.2 F limit for Control Room temperature is sufficient to account for the temperature rise in the enclosed cabinets.	None

3.7.13 FUEL STORAGE BUILDING EMERGENCY VENTILATION SYSTEM (FSBEVS)

10 CFR 50.36 CRITERION: 3

3.7.13 FUEL BUILDING AIR CLEANUP SYSTEM (FBACS)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None
3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)	None	None
4.5.A.6 Fuel Storage Building Emergency Ventilation System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.7.14 SPENT FUEL PIT WATER LEVEL

10 CFR 50.36 CRITERION: 2,

3.7.15 FUEL STORAGE POOL WATER LEVEL

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)	None	None

3.7.15 SPENT FUEL PIT BORON CONCENTRATION

10 CFR 50.36 CRITERION: 2

3.7.16 FUEL STORAGE POOL BORON CONCENTRATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)	None	None
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
5.0 Design Features	Application of TS Selection Criteria is not appropriate. However Design Features are included in ITS as required by 10 CFR 50.36. CTS Section 5.0 review identified an LCO like requirement. See 5.4.2 and 5.4.3.	None

3.7.16 SPENT FUEL ASSEMBLY STORAGE

10 CFR 50.36 CRITERION: 2

3.7.17 SPENT FUEL ASSEMBLY STORAGE

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)	None	None

3.7.17 SECONDARY SPECIFIC ACTIVITY

10 CFR 50.36 CRITERION: 2

3.7.18 SECONDARY SPECIFIC ACTIVITY

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.G Secondary Coolant Activity	The accident analysis of the main steam line break (MSLB),as discussed in the FSAR assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the EAB (i.e., site boundary) limits for whole body and thyroid dose rates.	None
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.8.1 AC SOURCES - OPERATING

10 CFR 50.36 CRITERION: 3

3.8.1 AC SOURCES - OPERATING

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.A AC and DC Electrical Sources-Operating	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.	None
4.5.A.1 Safety Injection System	CTS Surveillance Testing Requirements are maintained in the ITS.	Time delay relays
4.6.A Diesel Generators	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.7.G Actions for Inoperable Offsite Circuits and Emergency Diesel Generators	This requirement only applies if a safeguards power train cannot be powered from an offsite source or the emergency diesel. This requirement ensures that an event coincident with a single failure of the associated DG or offsite source will not result in a complete loss of redundant required features. Required safety features are designed with a redundant safety feature that is powered from a different safeguards power train. Therefore, if a required safety feature is supported by an inoperable offsite circuit or inoperable diesel, then the failure of the DG or offsite source associated with that required safety feature will not result in the loss of a safety function because the safety function will be accomplished by the redundant safety feature that is powered from a different safeguards power train.	Requirement Maintained by ITS LCO 3.8.1, Required Actions A.3 and C.1.

3.8.2 AC SOURCES - SHUTDOWN

10 CFR 50.36 CRITERION: 3

3.8.2 AC SOURCES SHUTDOWN

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.F AC and DC Electrical Sources-Shutdown	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of events during refueling and other shutdown conditions.	None
4.6.A Diesel Generators	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.8.3 DIESEL FUEL OIL AND STARTING AIR

10 CFR 50.36 CRITERION: 3

3.8.3 DIESEL FUEL OIL, LUBE OIL, AND STARTING AIR

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.7.A AC and DC Electrical Sources-Operating	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.	None

3.8.4 DC SOURCES - OPERATING

10 CFR 50.36 CRITERION: 3

3.8.4 DC SOURCES OPERATING

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.6.B Station Batteries	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.7.A AC and DC Electrical Sources-Operating	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.	None

3.8.5 DC SOURCES - SHUTDOWN

10 CFR 50.36 CRITERION: 3

3.8.5 DC SOURCES SHUTDOWN

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.F AC and DC Electrical Sources-Shutdown	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of events during refueling and other shutdown conditions.	None
4.6.B Station Batteries	CTS Surveillance Testing Requirements are maintained in the ITS.	None
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	None

3.8.6 BATTERY CELL PARAMETERS

10 CFR 50.36 CRITERION: 3

3.8.6 BATTERY CELL PARAMETERS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.A AC and DC Electrical Sources-Operating	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.	None
4.6.B Station Batteries	CTS Surveillance Testing Requirements are maintained in the ITS.	None

3.8.7 INVERTERS - OPERATING

10 CFR 50.36 CRITERION: 3

3.8.7 INVERTERS OPERATING

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.A AC and DC Electrical Sources-Operating	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.	None

3.8.8 INVERTERS - SHUTDOWN

10 CFR 50.36 CRITERION: 3

3.8.8 INVERTERS SHUTDOWN

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.F AC and DC Electrical Sources-Shutdown	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of events during refueling and other shutdown conditions.	None

3.8.9 DISTRIBUTION SYSTEMS - OPERATING

10 CFR 50.36 CRITERION: 3

3.8.9 DISTRIBUTION SYSTEMS OPERATING

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.A AC and DC Electrical Sources-Operating	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.	None

3.8.10 DISTRIBUTION SYSTEMS - SHUTDOWN

10 CFR 50.36 CRITERION: 3

3.8.10 DISTRIBUTION SYSTEMS SHUTDOWN

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.F AC and DC Electrical Sources-Shutdown	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of events during refueling and other shutdown conditions.	None

3.9 REFUELING OPERATIONS

10 CFR 50.36 CRITERION: N/

3.9 REFUELING OPERATIONS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8 Refueling, Fuel Handling and Storage		None

3.9.1 BORON CONCENTRATION

10 CFR 50.36 CRITERION: 2

3.9.1 BORON CONCENTRATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.D Boron Concentration during Refueling	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains 0.95 % k/k during refueling operations.	Boron concentration requirements retained; bor concentration limits relocated to the COLR.
3.6.A Containment Integrity	The containment functions to limit radioactive material released to the environment following design basis accidents.	Boron concentration requirements during refuel

3.9.2 NUCLEAR INSTRUMENTATION

10 CFR 50.36 CRITERION: 3

3.9.3 NUCLEAR INSTRUMENTATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	Requirements for SRMs during refueling operations.
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	Mode 6 requirements

3.9.3 CONTAINMENT PENETRATIONS

10 CFR 50.36 CRITERION: 3

3.9.4 CONTAINMENT PENETRATIONS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.6.A Containment Integrity	The containment functions to limit radioactive material released to the environment following design basis accidents.	None
4.13 Containment Vent (Pressure Relief) and Purge System	None	None
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	Requirements for containment during refueling operations.

3.9.4 RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - HIGH WATER LEVEL

10 CFR 50.36 CRITERION: 4

3.9.5 RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION HIGH WATER LEVEL

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None

3.9.5 RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION - LOW WATER LEVEL

10 CFR 50.36 CRITERION: 4

3.9.6 RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION LOW WATER LEVEL

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None

3.9.6 REFUELING CAVITY WATER LEVEL

10 CFR 50.36 CRITERION: 2

3.9.7 REFUELING CAVITY WATER LEVEL

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None

4.0 DESIGN FEATURES

10 CFR 50.36 CRITERION: N/

4.0 DESIGN FEATURES

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
5.0 Design Features	Application of TS Selection Criteria is not appropriate. However Design Features are included in ITS as required by 10 CFR 50.36. CTS Section 5.0 review identified an LCO like requirement. See 5.4.2 and 5.4.3.	None

5.1 RESPONSIBILITY

10 CFR 50.36 CRITERION: N/

5.1 RESPONSIBILITY

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
ETS 4.0 Administrative Controls (TSCR 98-018)	None	None
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

5.2 ORGANIZATION

10 CFR 50.36 CRITERION: N/

5.2 ORGANIZATION

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

5.3 UNIT STAFF QUALIFICATIONS

10 CFR 50.36 CRITERION: N/

5.3 UNIT STAFF QUALIFICATIONS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.3 Plant Staff Qualifications	None	None

5.4 PROCEDURES

10 CFR 50.36 CRITERION: N/

5.4 PROCEDURES

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
ETS 4.0 Administrative Controls (TSCR 98-018)	None	None

5.5.1 OFFSITE DOSE CALCULATION MANUAL (ODCM)

10 CFR 50.36 CRITERION: N/

5.5.1 OFFSITE DOSE CALCULATION MANUAL (ODCM)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
ETS 4.0 Administrative Controls (TSCR 98-018)	None	None

5.5.2 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

10 CFR 50.36 CRITERION: N/

5.5.2 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
License 2.L License Condition: Program to Reduce Leakage from Systems Outside Containment	Program to Reduce Leakage from Systems Outside Containment is retained as Technical Specifications 5.5.2. License Condition 2.L should be deleted to eliminate repetition.	None
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

5.5.3 POST ACCIDENT SAMPLING

10 CFR 50.36 CRITERION: N/

5.5.3 POST ACCIDENT SAMPLING

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
License 2.M License Condition: Program to ensure capability to determine airborne iodine concentration in vital areas in accident conditions	Program to ensure capability to determine airborne iodine concentration in vital areas in accident conditions is retained as Technical Specifications 5.5.3. License Condition 2.M should be deleted to eliminate repetition.	None

5.5.4 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.4 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
ETS 4.0 Administrative Controls (TSCR 98-018)	None	None
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

5.5.5 COMPONENT CYCLIC OR TRANSIENT LIMIT

10 CFR 50.36 CRITERION: N/

5.5.5 COMPONENT CYCLIC OR TRANSIENT LIMIT

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	FSAR requirements incorporated into the ITS.

5.5.6 REACTOR COOLANT PUMP FLYWHEEL INSPECTION PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.7 REACTOR COOLANT PUMP FLYWHEEL INSPECTION PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	FSAR requirements incorporated into the ITS.

5.5.7 INSERVICE TESTING PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.8 INSERVICE TESTING PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.2 Inservice Inspection	CTS Surveillance Testing Requirements are maintained in the ITS.	None

5.5.8 STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.9 STEAM GENERATOR (SG) TUBE SURVEILLANCE PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.9 Steam Generator Tube Inservice Surveillance	None	None

5.5.9 SECONDARY WATER CHEMISTRY PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.10 SECONDARY WATER CHEMISTRY PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
License 2.I License Condition: Secondary Water Chemistry Monitoring Program	Secondary Water Chemistry Monitoring Program is retained as Technical Specifications 5.5.9. License Condition 2.I should be deleted to eliminate repetition.	None

5.5.10 VENTILATION FILTER TESTING PROGRAM (VFTP)

10 CFR 50.36 CRITERION: N/

5.5.11 VENTILATION FILTER TESTING PROGRAM (VFTP)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.5.A.4 Containment Air Filtration System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.A.6 Fuel Storage Building Emergency Ventilation System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.5.A.5 Control Room Air Filtration System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
4.13 Containment Vent (Pressure Relief) and Purge System	None	None

5.5.11 EXPLOSIVE GAS AND STORAGE TANK RADIOACTIVITY MONITORING PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.12 EXPLOSIVE GAS AND STORAGE TANK RADIOACTIVITY MONITORING PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
ETS 1.1/2.1 Explosive Gas Monitoring (TSCR 98-018)	None	None
ETS 1.3/2.3 Explosive Gas Mixture (TSCR 98-018)	None	None
ETS 1.2/2.2 Radioactive Liquid Effluent Holdup Tanks (TSCR 98-018)	None	None

5.5.12 DIESEL FUEL OIL TESTING PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.13 DIESEL FUEL OIL TESTING PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	Neither CTS nor FSAR establishe any requirements for testing of diesel fuel oil.

5.5.13 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.14 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	None

5.5.14 SAFETY FUNCTION DETERMINATION PROGRAM (SFDP)

10 CFR 50.36 CRITERION: N/

5.5.15 SAFETY FUNCTION DETERMINATION PROGRAM (SFDP)

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
NEW ITS New Requirement imposed by ITS. No CTS Requirement.	None	None

5.5.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

10 CFR 50.36 CRITERION: N/

5.5.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

5.6 REPORTING REQUIREMENTS

10 CFR 50.36 CRITERION: N/

5.6 REPORTING REQUIREMENTS

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
ETS 4.0 Administrative Controls (TSCR 98-018)	None	None
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

5.7 HIGH RADIATION AREA

10 CFR 50.36 CRITERION: N/

5.7 HIGH RADIATION AREA

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

CTS DELETED

CTS DELETED

CTS REQUIREMENT DELETED 10 CFR 50.36 CRITERION: N/

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
License 2.C.4 License Condition: Pressurizer Weld Inspection prior to power operation following second refueling shutdown	License Condition 2.C.4 has been satisfied and is no longer applicable. License Condition 2.C.4 should be deleted.	None
License 2.O License Condition: Schedule for Completion of Balance of Plant Modifications to NRC by January 1, 1984.	License Condition 2.O has been satisfied and is no longer applicable. License Condition 2.O should be deleted.	None
License 2.J License Condition: Inspect all four steam generators by March 31, 1982.	License Condition 2.J has been satisfied and is no longer applicable. License Condition 2.J should be deleted. Steam Generator tube inspection requirements will be governed by Technical Specification 5.5.8.	None

Detail Removed

DETAIL REMOVED

CTS REQUIREMENT RETAINED IN ITS BUT SOME DETAILS IN CTS MOVED TO LICENSEE CONTROLLED DOCUMENT. SEE DISCUSSIONS OF CHANGE FOR ITS LCOS LISTED ABOVE. 10 CFR 50.36 CRITERION: N/

CTS Requirements reflected in this ITS	CTS NOTES	ITS NOTES
6.6 Reportable Event Action	None	None
3.7.D Emergency System Wide Blackout	Relaxation of requirements for offsite power during an emergency system wide blackout.	Will be maintained in the FSAR

N/A **N/A** 10 CFR 50.36 CRITERION: N/

N/A

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
License 2.N License Condition: Deleted by Amendment 49	None	None
3.14 Fire Protection (Deleted by Amendment 157)	Relocated by CTS Amendment 157.	None
4.5 Tests for Engineered Safety Features and Air Filtration Systems	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.1.C.2 Deleted	None	None
License 2.H License Condition: Fire Protection Program	Retained as a License Condition	None
3.7.B Actions for CTS 3.7.A	See CTS 3.7.A	None
3.7.C Actions for Inoperable AC and DC Sources and Distribution	See CTS 3.7.A.	None
N/A Not Applicable	None	None
License 2.K License Condition: Deleted by Amendment 49	None	None
License 2.D License Condition: Deleted by CTS Amendment 46	None	None
3.1.A.6 Deleted by Amendment 170	N/A	None
3.8.B Actions for CTS 3.8.A, Refueling Operations (Reactor Vessel & Containment)	See CTS 3.8.A	None
License 2.G License Condition: Physical Security Plan	Retained as a License Condition	None
License 2.F License Condition: Section 401, Federal Water Pollution Control Act Amendments of 1972	Retained as a License Condition	None
License 2.E License Condition: Deleted by CTS Amendment 37	None	None
License 2.C.2 License Condition: Technical Specifications	Retained as a License Condition	None
4.12 Fire Protection and Detection Systems	Relocated by CTS Amendment 157.	None
3.4.A.4 Piping and Valves For Main Steam Safety Valves, Auxiliary Feedwater and Condensate Storage Tank	See CTS 3.4.A.1, CTS 3.4.A.2 and CTS 3.4.A.3.	None

R.1 REACTOR VESSEL HEAD VENTS

10 CFR 50.36 CRITERION: No

R.1

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.A.7 Reactor Vessel Head Vents	See CTS Relocated Item R.1.	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

R.2 STEAM GENERATOR SECONDARY SIDE MINIMUM TEMPERATURE FOR PRESSURIZATION

10 CFR 50.36 CRITERION: No

R.2

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.B Heatup and Cooldown	This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. These limits are needed to avoid encountering pressure, temperature, or temperature rate of change condition that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary.	WCAP-11618 determined that steam generator P/T limits were not significant risk contributors t core damage frequency and offsite releases.

R.3 PRESSRIZER HEATUP AND COOLDOWN

10 CFR 50.36 CRITERION: No

R.3

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.1.B Heatup and Cooldown	This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. These limits are needed to avoid encountering pressure, temperature, or temperature rate of change condition that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary.	WCAP-11618 determined that the pressurizer temperature limits were not significant risk contributors to core damage frequency and offs releases.

R.4 MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

10 CFR 50.36 CRITERION: No

R.4

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.1.E Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	See CTS Relocated Item R.4.	None

R.5 CHEMICAL AND VOLUME CONTROL SYSTEM

10 CFR 50.36 CRITERION: No

R.5

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
T 4.1-2 Frequencies for Sampling Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.2 Chemical and Volume Control System	See CTS Relocated Item R.5.	None

R.6 WELD CHANNEL AND PENETRATION PRESSURIZATION SYSTEM (WC & PPS)

10 CFR 50.36 CRITERION: No

R.6

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.3.D Weld Channel and Penetration Pressurization System (WC & PPS)	See CTS Relocated Item R.6.	None
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

R.7 STEAM AND POWER CONVERSION SYSTEM (TURBINE GENERATOR)

10 CFR 50.36 CRITERION: No

R.7

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.4.D Turbine Generator Electrical Output	See CTS Relocated Item R.7.	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

R.8 AREA RADIATIN MOITORING AND PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; PLANT WIDE RANGE VENT MONITOR

10 CFR 50.36 CRITERION: No

R.8

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None
T 3.5-4 Instrument Operating Conditions for Isolation Functions	The Isolation Instrumentation functions to provide isolation of containment atmosphere and process systems that penetrate containment from the environment to limit the release of radioactivity following design basis accidents.	Relocated to the ODCM
T 4.1-1 Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)	None	None

R.9 AUXILIARY ELECTRICAL SYSTEMS (A.C. CIRCUIT INSIDE CONTAINMENT)

10 CFR 50.36 CRITERION: No

R.9

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.7.E A.C. Lighting Circuit Inside Primary Containment	See CTS Relocated Item R.9	None

R.10 REFUELING, FUEL HANDLING AND STORAGE (COMMUNICATIONS)

10 CFR 50.36 CRITERION: No

R.10

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None

R.11 REFUELING, FUEL HANDLING AND STORAGE (DECAY TIME)

10 CFR 50.36 CRITERION: No

R.11

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None

R.12 REFUELING (MANIPULATOR CRANES AND SPENT FUEL CASK)

10 CFR 50.36 CRITERION: No

R.12

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.8.A Refueling Operations (Reactor Vessel & Containment)	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains within limits during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.	None
3.8.C Fuel Handling and Storage Operations (Spent Fuel Pit Operations)	None	None
T 4.1-3 Frequencies for Equipment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

R.13 SERVICE WATER ISOLATION VALVE LEAKAGE (0.36 GPM LEAKAGE LIMIT)

10 CFR 50.36 CRITERION: No

R.13

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
4.4 Containment Tests	CTS Surveillance Testing Requirements are maintained in the ITS.	None

R.14 RADIOACTIVE MATERIALS MANAGEMENT

10 CFR 50.36 CRITERION: No

R.14

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
3.9 Radioactive Materials Management	See CTS Relocated Item R.14	None

R.15 MOVABLE INCORE INSTRUMENTATION

10 CFR 50.36 CRITERION: No

R.15

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.11 Movable Incore Instrumentation	See Relocated Item R.15	None

R.16 RIVER LEVEL (FLOODING PROTECTION)

10 CFR 50.36 CRITERION: No

R.16

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
3.12 River Level	See Relocated Item R.16	None

R.17 SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

10 CFR 50.36 CRITERION: No

R.17

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.11 Safety-Related Shock Suppressors (Snubbers)	See Relocated Item R.17	None
3.13 Safety-Related Shock Suppressors (Snubbers)	See Relocated Item R.17	None

R.18 TOXIC GAS MONITORING

10 CFR 50.36 CRITERION: No

R.18

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None
4.5.A.5 Control Room Air Filtration System	CTS Surveillance Testing Requirements are maintained in the ITS.	None
3.3.H Control Room Ventilation System (including toxic gas monitoring)	Control room ventilation provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CRACS is designed so that the functional capability of the Control Room is maintained during a Design Basis Accident. Functional capability means that the ambient temperature for safety equipment located in this room will not exceed 108.2 F. Control Room safety equipment is specified to a temperature of 120 F and the 108.2 F limit for Control Room temperature is sufficient to account for the temperature rise in the enclosed cabinets.	Toxic gas monitoring is an alarm only function is relocated consistent with Generic Letter 95-1

R.19 REACTOR COOLANT SYSTEM INTEGRITY TESTING

10 CFR 50.36 CRITERION: No

R.19

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.3 Reactor Coolant System Integrity Testing	CTS Surveillance Testing Requirements are maintained in the ITS.	None

R.20 SEISMIC INSTRUMENTATION

10 CFR 50.36 CRITERION: No

R.20

CTS Requrements reflected in this ITS	CTS NOTES	ITS NOTES
4.10 Seismic Instrumentation	See Relocated Item R.20	None
6.0 Administrative Controls	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36	None

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Relocated Item No: R.1

REACTOR VESSEL HEAD VENTS

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.1-5	179	179	No TSCRs	No TSCRs for this Page	N/A
3.1-8	179	179	No TSCRs	No TSCRs for this Page	N/A
T 4.1-3(2)	148	148	No TSCRs	No TSCRs for this Page	N/A

R.1

7. REACTOR VESSEL HEAD VENTS

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to operable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

8. OVERPRESSURE PROTECTION SYSTEM (OPS)

a. When the RCS temperature is below 319°F,

(1) the OPS shall be armed and operable. Both OPS PORVs shall have lift settings not to exceed those given in Figure 3.1.A-2, or

(2) the RCS must be vented with an equivalent opening of 2.00 square inches.

b. The requirements of 3.1.A.8.a may be modified to allow one PORV and/or its series block valve to be inoperable for a maximum of seven (7) consecutive days.

SEE ITS 3.4.12

c. If the requirements of 3.1.A.8.a or 3.1.A.8.b cannot be met, then one of the following actions shall be completed within 8 hours.

(1) The RCS must be depressurized and vented with an equivalent opening of at least 2.00 square inches;

Or

(2) The RCS must be heated in accordance with the restrictions of Specification 3.1.A.1.h(3) and maintained above 411°F;

Or

(3) Restrict pressurizer level as per the curves on Figures 3.1.A-5 and 3.1.A-6.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path. R.1

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the limits established in Reg. Guide 1.99, Revision 2. The OPS is considered to be operable when the minimum number of required channels (per Table 3.5-3) are available to open the PORVs upon receipt of a high pressure signal which is based upon RCS T_{cold}, as shown in Figure 3.1.A-2. (The happy face icon contained on this and other Technical Specification figures indicates the side of the applicable curve in which operation is permissible. Conversely, the sad face icon indicates the side of the applicable curve in which operation is prohibited.) The OPS setpoint is based upon a comparative analysis of References 5 and 9, with allowances for metal/fluid temperature differences (as described below) and for the static head due to elevation differences and dynamic head effect of the operation of the reactor coolant and RHR pumps. "Arming" means that the motor operated block valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to 319°F or manually by the control room operator.

Relocated Item (R-1)

TABLE 4.1-3 (Sheet 2 of 2)

SEE ITS 3.4.11 ITS 3.4.14

13. RHR Valves 730 and 731	Automatic isolation and interlock action	24M
14. PORV Block Valves	Operability through 1 complete cycle of full travel	Quarterly (see Note 1)
15. PORV Valves	Operability	24M
16. Reactor Vessel Head Vents	Operability	24M

R.1

24M - At least once per 24 months

SEE ITS 3.4.11

Note 1. If the block valve is shut due to a leaking or inoperable PORV, Block Valve operability will be checked the next time the plant is in cold shutdown.

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Relocated Item No: R.1

REACTOR VESSEL HEAD VENTS

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.1: REACTOR VESSEL HEAD VENTS

CTS 3.1.A.7:

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

Discussion:

The basic function of the Reactor Vessel Head Vent System is to remove non-condensible gases or steam from the Reactor Vessel Head. This system is designed to mitigate a possible condition of inadequate core cooling due to a loss of natural circulation resulting from the accumulation of non-condensible gases in the Reactor Coolant System. The Reactor Heat Vent System is operated manually from the main control room. (IP3 FSAR 4.2.11) This function is consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," however, the operation of reactor vessel head vents is not assumed in the safety analysis because the operation of the vents is not part of the primary success path in the Final Safety Analysis Report (FSAR). The operation of these vents is an operator action after the event has occurred and is required only when there is indication that natural circulation is not occurring.

Comparison to Selection Criteria:

1. RCS vent paths are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. RCS vent paths are not a process variable, design feature, or operating restriction that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. RCS vent paths are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-44) and summarized in Table 1 of WCAP-11618, the RCS vent paths were found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation and considers it applicable to IP3. RCS vent paths are not important for any scenarios modeled in the IP3 Individual Plant Examination (IPE).

Conclusion:

Since the selection criteria have not been satisfied, the RCS Vent Paths LCO and Surveillances will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement for two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.2

STEAM GENERATOR SECONDARY SIDE MINIMUM TEMPERATURE FOR PRESSURIZATION)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.1-17	179	179	No TSCRs	No TSCRs for this Page	N/A

B. HEATUP AND COOLDOWN

Specifications

SEE ITS 3.4.3

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 13.3 effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

R.2

3. ~~The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.~~ R.2

SEE Relocated R.3

4. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

SEE ITS 3.4.3

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

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Relocated Item No: R.2

STEAM GENERATOR SECONDARY SIDE MINIMUM TEMPERATURE FOR PRESSURIZATION)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document

RELOCATED ITEM R.2: STEAM GENERATOR SECONDARY SIDE MINIMUM TEMPERATURE FOR PRESSURIZATION

CTS 3.B.3:

The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70^{0}F.

Discussion:

The limitation on steam generator pressures and temperature (i.e., P/T) ensures that pressure-induced stresses on the steam generators do not exceed the maximum allowable fracture toughness limits. These pressure and temperature limits are based on maintaining steam generator RT_{NDT} sufficient to prevent brittle fracture. As such, the TS places limits on variables consistent with structural analysis results. However, these limits are not initial condition assumptions of a FSAR accident analysis. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, the Final Policy Statement Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in TS.

Comparison to Selection Criteria:

1. The steam generator P/T limits are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Steam generator P/T limits are not a process variable, design feature, or operating restriction that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Steam generator P/T limits are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-55) and summarized in Table 1 of WCAP-11618, the steam generator P/T limits were found to be non-significant risk contributors to core damage frequency and offsite releases. This is, in large part, due to Steam Generator Tube Rupture (SGTR) events being negligible contributors in past PWR PRAs. For IP3, SGTR sequences are important in the IP3 Individual Plant Examination (IPE). However, the plant-specific IPE does not evaluate conditions below 70°F. In addition, it is also recognized that the likelihood of pressurizing the SG secondary side when RCS temperature is below 70°F is small.

Conclusion:

Since the selection criteria have not been satisfied, the steam generator P/T limits LCO and Surveillances will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement steam generator secondary side minimum temperature for pressurization. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.3

PRESSRIZER HEATUP AND COOLDOWN

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.1-17	179	179	No TSCRs	No TSCRs for this Page	N/A
3.1-21	121	121	No TSCRs	No TSCRs for this Page	N/A

B. HEATUP AND COOLDOWN

Specifications

SEE ITS 3.4.3

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 13.3 effective full-power years (EFPYs). The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr respectively.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in the Basis and results of surveillance specimens as covered in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens.

SEE RELOCATED R.2

3. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

R.3

4. ~~The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.~~ R.3

5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3.

Basis

Fracture Toughness Properties

SEE ITS 3.4.3

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code [6] and ASTM E185 [5] and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code [1], and the calculation methods described in WCAP-7924 [2].

Relocated Item (R-3)

R.3

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

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Relocated Item No: R.3

PRESSRIZER HEATUP AND COOLDOWN

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
RELOCATED ITEM R.3: PRESSURIZER HEATUP AND COOLDOWN

CTS 3.1.B.4:

The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Discussion:

Limits are placed on pressurizer operation to prevent a non-ductile failure. Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

Comparison to Selection Criteria:

1. The pressurizer temperature limits are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The pressurizer temperature limits are not a process variable, design feature, or operating restriction that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The pressurizer temperature limits are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a FSAR accident that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-41) and summarized in Table 1 of WCAP-11618, the pressurizer temperature limits were found to

be non-significant risk contributors to core damage frequency and offsite releases. NYPA has reviewed this evaluation and considers it applicable to IP3. The pressurizer temperature limits are outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the pressurizer temperature limits LCO and Surveillances will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the limits for pressurizer heatup and cooldown rates and use of spray if the temperature difference between the pressurizer and the spray fluid is greater than 320°F . Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.4

MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.1-29	121	121	No TSCRs	No TSCRs for this Page	N/A
3.1-30	121	121	No TSCRs	No TSCRs for this Page	N/A
T 4.1-2(1)	139	139	No TSCRs	No TSCRs for this Page	N/A

R.4

E. MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Specification

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

Contaminant	Normal Steady-State Operation (PPM)	Transient Not To Exceed 24 Hours (PPM)
a. Oxygen	0.10	1.00
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

2. If any of the normal steady-state operating limits as specified in 3.1.E.1, above, are exceeded, or if it is anticipated that they may be exceeded, corrective action shall be taken immediately.

3. If the concentrations of any of the contaminants cannot be controlled within the limits of Specification 3.1.E.1, namely, steady-state limit not restored within 24 hours or transient limit exceeded, the reactor shall be brought to the cold shutdown condition, utilizing normal operating procedures, and the cause of the out-of-specification operation ascertained and corrected. The reactor may then be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values. Otherwise, a safety review is required before startup.

4. Concentrations of contaminants in the reactor coolant shall not exceed the following maximum limits when the reactor coolant temperature is below 250°F:

Contaminant	Normal Concentration (PPM)	Transient Not To Exceed 48 Hours (PPM)
a. Oxygen	Saturated	Saturated
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

If the limits above are exceeded, namely, normal concentration limits not restored within 48 hours or the transient limits exceeded, the reactor shall be immediately brought to the cold shutdown condition and the cause of the out-of-specification condition ascertained and corrected.

Amendment No.121

R.4

5. For the purposes of correcting the contaminant concentrations to meet Specifications 3.1.E.1 and 3.1.E.4 above, increase in coolant temperature consistent with operation of reactor coolant pumps for a short period of time to assure mixing of the coolant shall be permitted. This increase in temperature to assure mixing shall in no case cause the coolant temperature to exceed 250°F.

Basis

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in 3.1.E.1 and 3.1.E.4, the integrity of the reactor coolant system is assured against stress corrosion cracking under all operating conditions. (1)

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank (2) during power operation. Because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately, as the condition can be corrected. Thus, the periods of either 24 hours or 48 hours for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the proper period (24 hours or 48 hours), then the reactor will be brought to the cold shutdown condition and the corrective action will continue.

R.4

The effects of contaminants in the reactor coolant are time and temperature dependent. It is consistent, therefore, to permit a transient concentration to exist for a longer period of time and still provide the assurance that the integrity of the primary coolant system will be maintained.

In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

References

1) FSAR Section 4.2

2) FSAR Section 9.2

R.4

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS			
Sample	**Analysis**	**Frequency**	**Maximum Time Between Analysis**
1. Reactor Coolant	Gross Activity[1]	5 days/week[1][4]	3 days[4]
	Tritium Activity	Weekly[1]	10 days
	Boron concentration	2 days/week	5 days
	Radiochemical (gamma)[2] Spectral Check	Monthly	45 days
	Oxygen and Chlorides Concentration	3 times per 7 days	3 days
	Fluorides Concentration	Weekly	10 days
	$\bar{E}$ Determination [3]	Semi-Annually	30 weeks
	Isotopic Analysis for I-131, I-133, I-135	Once per 14 days[5]	20 days
2. Boric Acid Tank	Boron Concentration, Chlorides	Weekly	10 days
3. Spray Additive Tank	NaOH Concentration	Monthly	45 days
4. Accumulators	Boron Concentration	Monthly	45 days
5. Refueling Water Storage Tank	Boron Concentration pH, Chlorides	Monthly	45 days
	Gross Activity	Quarterly	16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis)	Monthly	45 days
	Gross Activity	3 times per 7 days	3 days
7. Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8. Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days

R.4

R.4

R.4

SEE CTS MASTER MARKUP

Relocated Item (R-4)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: **R.4**

MAXIMUM REACTOR COOLANT OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.4: MAXIMUM RCS OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

CTS 3.1.E

Concentrations of contaminants in the reactor shall not exceed the following limits when the reactor coolant is above 250°F:

Contaminant	Normal Steady-State Operation (PPM)	Transient Not To Exceed 24 Hours (PPM)
a. Oxygen	0.10	1.00
b. Chloride	0.15	1.50
c. Fluoride	0.15	1.50

Discussion:

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

The effects of contaminants in the reactor coolant are time and temperature dependent. It is consistent, therefore, to permit a transient concentration to exist for some period of time and still provide the assurance that the integrity of the primary coolant system will be maintained. Additionally, if these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank during power operation. Because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately.

Justification for Relocation of CTS Requirement to
Licensee Controlled Document

Relocated Item R.4: MAXIMUM RCS OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Comparison to Selection Criteria:

1. Reactor coolant water chemistry is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Reactor coolant water chemistry is not a process variable, design feature, or operating restriction that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Reactor coolant water chemistry is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-40) and summarized in Table 1 of WCAP-11618, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation and considers it applicable to IP3. Effects of RCS chemistry are outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the RCS Chemistry LCO and Surveillances will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the limits for Oxygen, chloride and fluoride in the reactor coolant system. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Justification for Relocation of CTS Requirement to
Licensee Controlled Document

Relocated Item R.4: MAXIMUM RCS OXYGEN, CHLORIDE AND FLUORIDE CONCENTRATION

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.5

CHEMICAL AND VOLUME CONTROL SYSTEM

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.2-1	139	139	No TSCRs	No TSCRs for this Page	N/A
3.2-2	139	139	No TSCRs	No TSCRs for this Page	N/A
3.2-3	139	139	No TSCRs	No TSCRs for this Page	N/A
3.2-4	139	139	No TSCRs	No TSCRs for this Page	N/A
T 4.1-2(1)	139	139	No TSCRs	No TSCRs for this Page	N/A
T 4.1-3(1)	178 TSCR 97-156, 98-043	178	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated
T 4.1-3(1)	178 TSCR 97-156, 98-043	178	IPN 97-156	SR Freq for Main Turbine Stop and Control Valves	Incorporated

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

R.5

Specification

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be brought above the cold shutdown condition unless the following requirements are met:

1. Two charging pumps shall be operable.

2. Two boric acid transfer pumps shall be operable.

3. The boric acid storage system shall contain a minimum of 6100 gallons of 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) boric acid solution at a temperature of at least 145°F.

4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.

5. The appropriate boric acid storage tank level indicator(s) shall be operating.

6. Two channels of heat-tracing shall be operable for the flow path from the boric acid storage system to the Reactor Coolant System.

R.5

7. City water piping and valves shall be operable to the extent required to provide emergency cooling water to the charging pumps and flush water for the concentrated boric acid piping from the outlet of the boric acid storage tanks to the charging pump suction.

C. The requirements of 3.2.B may be modified to allow any one of the following components to be inoperable at any one time:

1. One of the two operable charging pumps may be removed from service provided a second charging pump is restored to an operable status within 24 hours.

2. One boric acid transfer pump may be inoperable for a period not to exceed 48 hours.

3. The boric acid storage system may be inoperable for a period not to exceed 48 hours provided that the RWST is operable.

4. One channel of heat tracing for the flow path from the boric acid storage system to the Reactor Coolant System may be out of service provided the failed channel is restored to an operable status within 7 days and the redundant channel is demonstrated to be operable daily during that period.

D. If the Chemical and Volume Control System is not restored to meet the requirements of 3.2.B within the time period specified in 3.2.C, then:

1. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

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3. In either case, if the requirements of 3.2.B are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

BASIS

The Chemical and Volume Control System[1] provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with either one of the two boric acid transfer pumps. An alternate method of boration will be to use the charging pumps taking suction directly from the refueling water storage tank. A third method will be to depressurize and use the safety injection pumps.

There are three sources of borated water available for injection through 3 different paths:

1. The boric acid transfer pumps can deliver the contents of the boric acid storage system to the charging pumps.

2. The charging pumps can take suction from the refueling water storage tank.

3. Injection of borated water from the refueling water storage tank with the safety injection pumps[2].

The quantity of boric acid in storage from either the boric acid storage system or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life.

R.5

A combined minimum deliverable volume of 6100 gallons with an averaged concentration of the 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) of boric acid are required to meet cold shutdown conditions. An upper concentration limit of 13% (22,735 ppm of boron) boric acid in the boric acid storage system is specified to maintain solution solubility at the specified low temperature limit of 145°F. One channel of heat tracing is sufficient to maintain the specified low temperature limit. The second channel of heat tracing provides backup for continuous plant operation when one channel is inoperable. Should both channels of heat tracing become inoperable, the reactor will be shutdown and can easily be borated before the line temperature is reduced near the boric acid precipitive temperature.

The city water system is used as a source of water for emergency cooling of the charging pumps and as a source of flush water to remove concentrated boric acid from the piping between the outlet of the boric acid storage tanks and the inlet to the charging pumps in the unlikely event of a complete loss of electrical power and/or a complete loss of service water resulting from turbine missiles.

References

1) FSAR - Section 9.2
2) FSAR - Section 6.2
3) "Revised Feasibility Report For BIT Elimination For Indian Point Unit 3," July 1988 (Westinghouse report).

R.5

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS

Sample	Analysis	Frequency	Maximum Time Between Analysis
1. Reactor Coolant	Gross Activity[1]	5 days/week[1][4]	3 days[4]
	Tritium Activity	Weekly[1]	10 days
	Boron concentration	2 days/week	5 days
	Radiochemical (gamma)[2] Spectral Check	Monthly	45 days
	Oxygen and Chlorides Concentration	3 times per 7 days	3 days
	Fluorides Concentration	Weekly	10 days
	$\bar{E}$ Determination [3]	Semi-Annually	30 weeks
	Isotopic Analysis for I-131, I-133, I-135	Once per 14 days[3]	20 days
2. ~~Boric Acid Tank~~	~~Boron Concentration, Chlorides~~	~~Weekly~~	~~10 days~~ R5
3. Spray Additive Tank	NaOH Concentration	Monthly	45 days
4. Accumulators	Boron Concentration	Monthly	45 days
5. Refueling Water Storage Tank	Boron Concentration pH, Chlorides	Monthly	45 days
	Gross Activity	Quarterly	16 weeks
6. Secondary Coolant	I-131 Equivalent (Isotopic Analysis)	Monthly	45 days
	Gross Activity	3 times per 7 days	3 days
7. Component Cooling Water	Gross Activity, Corrosion Inhibitor and pH	Monthly	45 days
8. Spent Fuel Pool (when fuel stored)	Gross Activity Boron Concentration, Chlorides	Monthly	45 days

SEE CTS MASTER MARKUP

Relocated Item (R-5)

Relocated Item (R-5)

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS		
	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M*
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop Control Valves	Closure	Yearly
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Quarterly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

SEE CTS MASTER MARKUP

R.5

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* Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.

Indian Point 3
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Relocated Item No: R.5

CHEMICAL AND VOLUME CONTROL SYSTEM

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.5: CHEMICAL AND VOLUME CONTROL SYSTEM

CTS 3.2:

A. When fuel is in the reactor there shall be at least one flow path to the core for boric acid injection.

B. The reactor shall not be brought above the cold shutdown condition unless the following requirements are met:
   1. Two charging pumps shall be operable.
   2. Two boric acid transfer pumps shall be operable.
   3. The boric acid storage system shall contain a minimum of 6100 gallons of 11 1/2% to 13% by weight (20,112 ppm to 22,735 ppm of boron) boric acid solution at a temperature of at least 145°F.
   4. System piping and valves shall be operable to the extent of establishing one flow path from the boric acid storage system and one flow path from the refueling water storage tank (RWST) to the Reactor Coolant System.
   5. The appropriate boric acid storage tank level indicator(s) shall be operating.
   6. Two channels of heat tracing shall be operable for the flow path from the boric acid storage system to the Reactor Coolant System.
   7. City water piping and valves shall be operable to the extent required to provide emergency cooling water to the charging pumps and flush water for the concentrated boric acid piping from the outlet of the boric acid storage tanks to the charging pump suction.

Discussion:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin.

The Chemical and Volume Control System ensures negative reactivity control is available for normal operation (normal makeup and chemical shim reactivity control) and provides an alternate method for borating the reactor coolant system. However, this system is not assumed to mitigate any design basis

accident or transient. Other systems (e.g., Safety Injection pumps) and other borated water sources (RWST) are assumed in the safety analysis.

In the case of a malfunction of the CVCS, which causes a boron dilution event, the automatic response, or that required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before the shutdown margin is lost. Operations of the boration subsystem is not assumed to mitigate this event.

Comparison to Screening Criteria:

1. The Chemical and Volume Control System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

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

3. The Chemical and Volume Control System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, pages A-8, A-9 and A-10) and summarized in Table 1 of WCAP-11618, the loss of the Chemical and Volume Control System was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation and considers it applicable to IP3. The conclusions from this generic analysis are consistent with the plant-specific IPE.

Conclusion:

Since the screening criteria have not been satisfied, the Chemical and Volume Control System LCO and Surveillances will be relocated to the Final Safety Analysis Report and requirements will be implemented by plant procedures. There is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the

requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.6

Weld Channel and Penetration Pressurization System (WC & PPS)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.3-8	145	145	No TSCRs	No TSCRs for this Page	N/A
3.3-18	145	145	No TSCRs	No TSCRs for this Page	N/A
4.4-3	174	174	No TSCRs	No TSCRs for this Page	N/A

R.6

D. Weld Channel and Penetration Pressurization System (WC & PPS)

1. The reactor shall not be brought above the cold shutdown condition unless:

   a. All required portions of the four WC & PPS zones are pressurized above 43 psig.*

   b. The uncorrected air consumption for the WC & PPS is less than or equal to 0.2% of the containment volume per day.

2. The requirements of 3.3.D.1 may be modified as follows:

   a. Any one of the four WC & PPS zones may be inoperable for a period not to exceed seven consecutive days.

   b. The uncorrected air consumption for the WC & PPS may not be in excess of 0.2% of the containment volume per day except for a period not to exceed seven consecutive days. If at any time it is determined that this limit is exceeded, repairs shall be initiated immediately.

3. If the WC & PP System is not restored to an operable status within the time period specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

   c. In either case, if the WC & PP System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

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* Certain portions of the Weld Channel Pressurization System have become inoperable and are not practicably accessible for repair. These portions of the Weld Channel Pressurization System have been disconnected from the system and are no longer considered required portions of the four WC & PPS zones.

SEE ITS 3.6.6

Due to the distribution of the five fan cooler units and two containment spray pumps on the 480 volt buses, the closeness to which the combined equipment approaches minimum safeguards varies with which particular component is out of service. Accordingly, the allowable out of service periods vary according to which component is out of service. Under no conditions do the combined equipment degrade below minimum safeguards.

The seven day out of service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is consistent with W Standardized Technical Specifications. This is allowable because no credit has been taken for operation of these systems in the calculation of off-site accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems.[11] The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems for the maximum containment calculated peak accident pressure of 42.42 psig.[15] A WC & PPS zone is considered that portion of piping downstream of the air receiver discharge check valve up to the last component pressurized by that system portion.

Some portions of the Weld Channel Pressurization System (WCPS) piping would not be practicably accessible for repair if they became inoperable. A section of WCPS piping is considered to be inoperable if it brings the air consumption of the WC & PPS above the required 0.2% of the containment volume per day or if the section can not maintain a pressure above the required 43 psig. If it is determined, by written evaluation, that an inoperable section of piping is not practicably accessable for repair, then that portion of the WCPS may be disconnected from the system. Inoperable sections of WCPS piping which can be considered for disconnection will satisfy one of the following criteria: 1) the piping is covered by concrete and repairs of the piping would involve the removal of some portion of the containment structure; or 2) the piping is located behind plant equipment in the containment building and repairs of the piping would involve the relocation of the equipment. The integrity of the welds associated with any disconnected portions of the WCPS is verified by integrated leak rate testing. The provision that allows for the disconnection of portions of the WCPS piping does not apply to any other WC & PPS piping.

R.6

SEE ITS 3.7.8

The Component Cooling System is not required during the injection phase of a loss-of-coolant accident. The component cooling pumps are located in the Primary Auxiliary Building and are accessible for repair after a loss-of-coolant accident.[6] During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards.[7]

R.6

C. Sensitive Leakage Rate

~~Verify the leakage rate for the Containment Penetration and Weld Channel Pressurization System is ≤ 0.2 percent of the containment free volume per day when pressurized to ≥ 43 psig and the containment pressure is atmospheric. The testing shall be performed at intervals no greater than 3 years.~~

D. Air Lock Tests

SEE ITS 3.6.2

Perform required Containment Air Lock leak rate testing in accordance with the Containment Leakage Rate Testing Program.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.6

Weld Channel and Penetration Pressurization System (WC & PPS)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.6: Weld Channel and Penetration Pressurization System

CTS 3.3.D:

The reactor shall not be brought above the cold shutdown condition unless:

a. All required portions of the four WC & PPS zones are pressurized above 43 psig.*

b. The uncorrected air consumption for the WC & PPS is less than or equal to 0.2% of the containment volume per day.

* Certain portions of the Weld Channel Pressurization System have become inoperable and are not practicably accessible for repair. These portions of the Weld Channel Pressurization System have been disconnected from the system and are no longer considered required portions of the four WC & PPS zones.

Discussion:

The Weld Channel and Penetration Pressurization System (WC & PPS) is incorporated into the design of Indian Point 3 as an engineered safety feature. Its purpose is to provide pressurized gas to some containment penetrations and most liner inner weld seams such that, in the event of a LOCA, there would be no leakage through these potential leakage paths from the containment to the atmosphere. Spaces between selected isolation valves are also served by the (WC & PPS). By maintaining the (WC & PPS) at some pressure level above the peak accident pressure, any postulated leakage would be into the Containment rather than out of the Containment.

Although the WC & PPS is an engineered safety feature, no credit is taken for its operation in calculating the amount of radioactivity released for offsite dose evaluations. Containment leakage rate testing is performed such that the WC & PPS is not used to satisfy leakage rate requirements. For Indian Point 3, offsite dose calculations were performed to demonstrate compliance with 10 CFR 100 guidelines without the benefit of the WC & PPS and the results were well within those guidelines. In those calculations, it was assumed that the Containment leaked at a rate of 0.1% per day of Containment free volume for the first 24 hours and 0.045% per day of Containment free volume thereafter. (FSAR 6.6.1)

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.6: Weld Channel and Penetration Pressurization System

Comparison to Screening Criteria:

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

2. The WC & PPS is not a process variable, design feature or operating restriction that is an initial condition of a DBA or transient.

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

4. Effects of WC & PPS are outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the screening criteria have not been satisfied, the WC & PPS LCO and surveillances will be relocated to the Final Safety Analysis Report and requirements will be implemented by plant procedures. There is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.7

STEAM AND POWER CONVERSION SYSTEM
(Turbine Generator)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.4-2	92	92	No TSCRs	No TSCRs for this Page	N/A
3.4-5	1-18-95	1-18-95	No TSCRs	No TSCRs for this Page	N/A
F 3.4-1	0	0	No TSCRs	No TSCRs for this Page	N/A
F 3.4-2	0	0	No TSCRs	No TSCRs for this Page	N/A
T 4.1-3(1)	178 TSCR 97-156, 98-043	178	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated
T 4.1-3(1)	178 TSCR 97-156, 98-043	178	IPN 97-156	SR Freq for Main Turbine Stop and Control Valves	Incorporated

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

B. Except as modified by E. below, if during power operations any of the conditions of 3.4-A above, except Items (1) and (2), cannot be met within 48 hours, the operator shall start to shutdown and cool the reactor below 350°F using normal operation procedures.

C. If during power operations, the requirement of 3.4.A.2 is not satisfied, the following actions shall be taken:

SEE ITS 3.7.7 ITS 3.7.5 ITS 3.7.6

1) With one auxiliary feedwater pump inoperable, restore the pump to operable status within 72 hours or be in hot shutdown within the next 12 hours.

2) With two auxiliary feedwater pumps inoperable, be in hot shutdown within 12 hours.

3) With three auxiliary feedwater pumps inoperable, maintain the plant in safe stable mode which minimizes the potential for a reactor trip and, immediately initiate corrective action to restore at least one auxiliary feedwater pump to operable status as soon as possible.

R.7

D. ~~The gross turbine-generator electrical output at all times shall be within the limitation of Figure 3.4-1 or Figure 3.4-2 for the application conditions of turbine overspeed setpoint, number of operable low pressure steam dump lines, and condenser back pressure as noted thereon.~~ R.7

E. The reactor shall not be heated above 350°F unless both valves in the single auxiliary feedwater supply line from the Condensate Storage Tank are open. If, during power operations, it is discovered that one or both of the valves are closed, the following action shall be taken:

SEE ITS 3.7.5 ITS 3.7.6 ITS 3.7.7

1) Immediately place the auxiliary feedwater system in the manual mode,

2) Within one hour either:

a) reopen the closed valve(s),

or

b) open the valves to the alternate city water supply,

and

3) Once a water supply has been restored, return the system to the automatic mode.

R7

The limitations placed on turbine-generator electrical output due to conditions of turbine overspeed setpoint, number of operable steam dump lines, and condenser back pressure are established to assure that turbine overspeed (during conditions of loss of plant load) will be within the design overspeed value considered in the turbine missile analysis. [2] In the preparation of Figures 3.4-1 and 3.4-2, the specified number of operable L.P. steam dump lines is shown as one (1) greater than the minimum number required to act during a plant trip. The limitations on electrical output, as indicated in Figures 3.4-1 and 3.4-2, thus consider the required performance of the L.P. Steam Dump System in the event of a single failure for any given number of operable dump lines.

Relocated Item (R-7)

R7

INDIAN POINT UNIT NO.3

Curve of Power Level versus Number of Operable L.P. Dump Lines with Parameters of Trip Set Point Required to Limit Maximum Overspeed to 132% Based on 1.5" Hg abs. Condenser Pressure

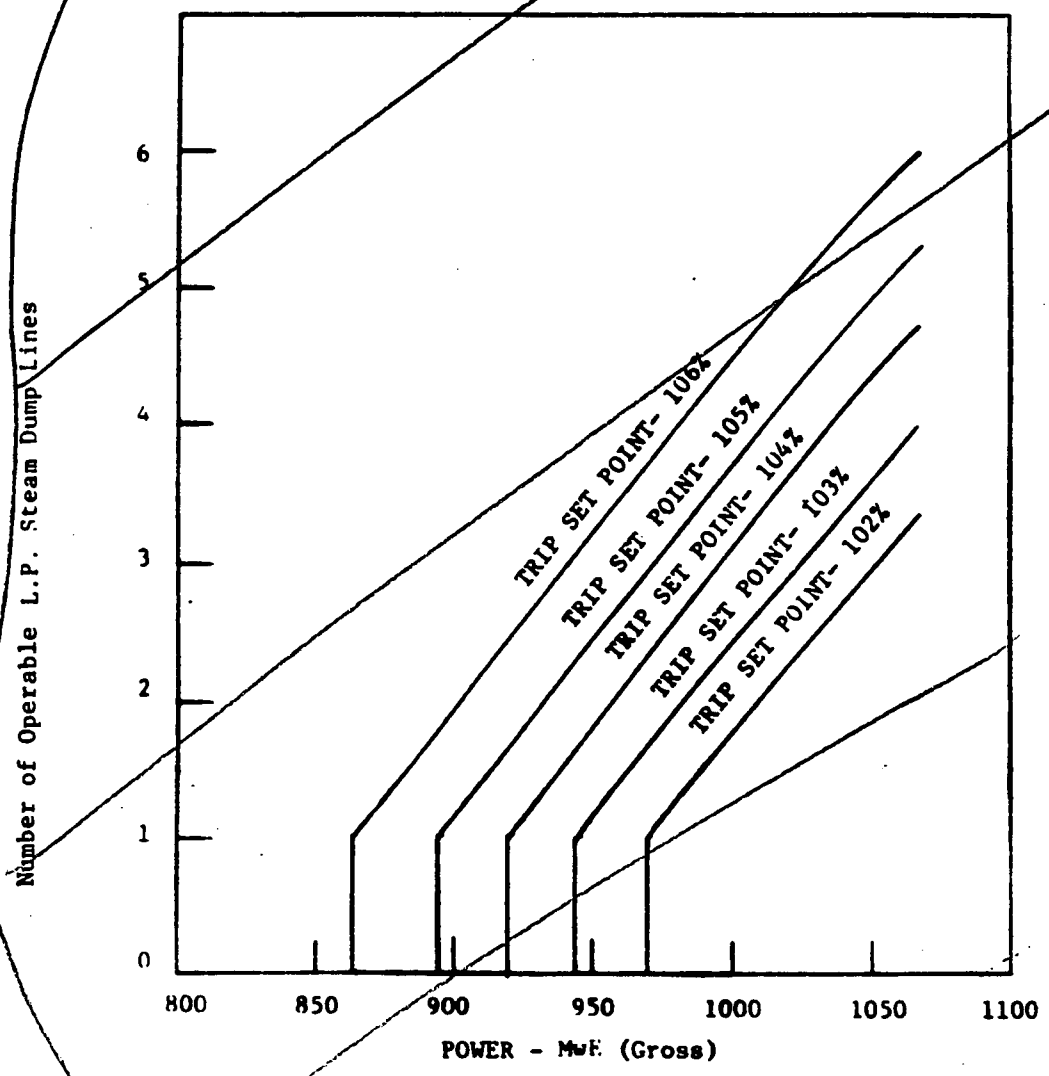

Figure 3.4-2 Gross Electrical Output
1.5 inch Hg Backpressure

Relocated Item (R-7)

R.7

INDIAN POINT UNIT NO.3

Curve of Power Level versus Number of Operable Dump Lines with Parameters of Trip Set Point Required to Limit Maximum Overspeed to 132% Based on 1.0" Hg abs. Condenser Pressure

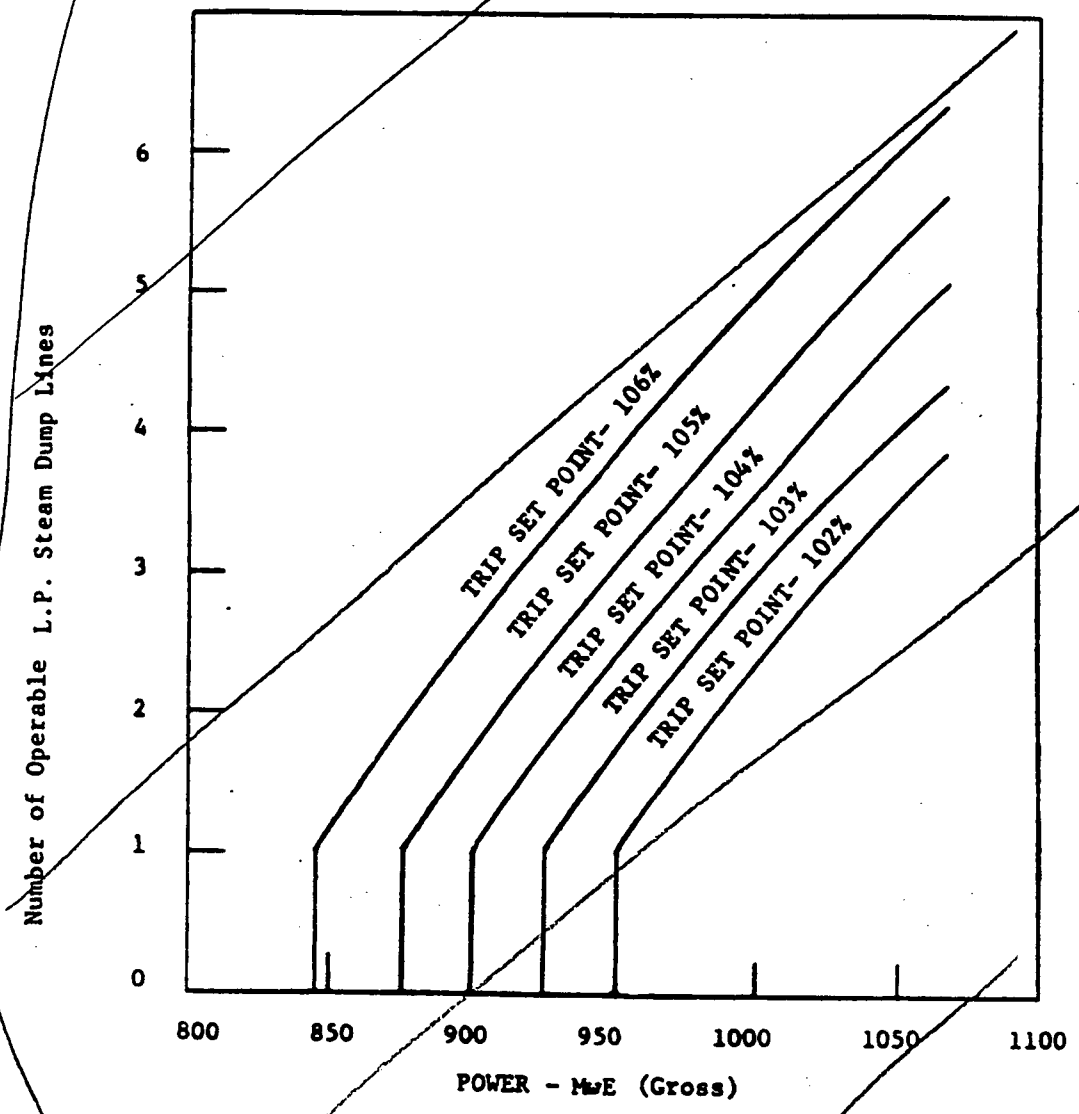

Figure 3.4-1 Gross Electrical Output
1.0 inch Hg Backpressure

Relocated Item (R-7)

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS		
	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M*
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop Control Valves	Closure	Yearly
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Quarterly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

SEE CTS MASTER MARKUP

R.7

R.7

SEE CTS MASTER MARKUP

* Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.7

STEAM AND POWER CONVERSION SYSTEM
(Turbine Generator)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.7: STEAM AND POWER CONVERSION SYSTEM (Turbine Generator)

CTS 3.4.D

The gross turbine-generator electrical output at all times shall be within the limitation of Figure 3.4-1 or Figure 3.4-2 for the application conditions of turbine overspeed setpoint, number of operable low pressure steam dump lines, and condenser back pressure as noted thereon.

Discussion:

The limitations placed on turbine-generator electrical output due to conditions of turbine overspeed setpoint, number of operable steam dump lines, and condenser back pressure are established to assure that turbine overspeed (during conditions of loss of plant load) will be within the design overspeed value considered in the turbine missile analysis.

Comparison to Screening Criteria:

1. The turbine-generator overspeed protection features are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The turbine-generator overspeed protection features are not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The turbine-generator overspeed protection features are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-30) and summarized in Table 1 of WCAP-11618, the loss of turbine-generator overspeed protection features was found to be a non-significant risk contributor to core damage frequency and offsite releases. Additionally, after the completion of the assessment referenced in Section 4.0 (Appendix A, page A-30), Indian Point 3 has installed three low pressure turbines that are significantly improved in design than the previous low pressure

turbines. This new design reduces the probability of a low pressure turbine rotor failure which generates an external turbine missile. The new designed rotors are of a welded discs type. These two major design changes have demonstrated excellent results in operating experiences with no stress corrosion cracking and yields a low probability of external missile generation. NYPA has reviewed this evaluation and considers it applicable to IP3.

Conclusion:

Since the screening criteria have not been satisfied, the turbine-generator electrical output limits will be relocated to the Final Safety Analysis Report and requirements will be implemented by plant procedures. There is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.8

AREA RADIATION MONITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; Plant Wide Range Vent Monitor

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
T 3.5-4(2)	151	151	No TSCRs	No TSCRs for this Page	N/A
3.8-1	86	86	No TSCRs	No TSCRs for this Page	N/A
3.8-3	114	114	No TSCRs	No TSCRs for this Page	N/A
T 4.1-1(2)	169	169	No TSCRs	No TSCRs for this Page	N/A
T 4.1-1(4)	169 TSCR 98-043	169 TSCR 98-043	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated

3.8 Refueling, Fuel Handling and Storage

Applicability

Applies to operating limitations during refueling, fuel handling, storage operations, and when heavy loads are moved over the reactor when the head is removed.

Objective

To ensure that no incident could occur during refueling, fuel handling, and storage operations that would adversely affect public health and safety.

Specification

A. During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met: R.8

1. The equipment door and at least one door in each personnel air lock shall be properly closed. When the closure plate with a personnel door that prevents direct air flow from the containment is used, it shall be properly closed.

2. At least one isolation valve shall be operable, locked closed or blind flanged in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

3. Radiation levels in the containment and spent fuel storage areas shall be monitored continously. R.8

4. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service.

5. At least one residual heat removal pump and heat exchanger shall be operating except during those core alternations in which the residual heat removal flow interferes with component positioning.

6. During reactor vessel head removal and while loading and unloading fuel in the reactor, T_{avg} shall be < 140°F.

7. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

SEE CTS MASTER MARKUP

a. No. 31 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

b. No. 32 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

c. The water level in the refueling cavity above the top of the reactor vessel flange is equal to or greater than 23 feet.

B. If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

R.8

C. During fuel handling and storage operations, the following conditions shall be met:

1. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used. — R.8

2. The spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, no loads in excess of 2,000 pounds shall be moved over any region of the spent fuel pit.

3. During periods of spent fuel cask or fuel storage building cask crane movement over the spent fuel pit, or during periods of spent fuel movement in the spent fuel pit when the pit contains irradiated fuel, the pit shall be filled with borated water at a concentration of >1000 ppm.

4. Whenever movement of irradiated fuel in the spent fuel pit is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated in the storage rack.

TABLE 3.5-4 (Page 2 of 2)

	No. FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS IN COLUMN 3 OR 4 CANNOT BE MET
SEE ITS 3.3.2 3.7.3	3. FEEDWATER LINE ISOLATION					
	a. Safety Injection	See	Item	No. 1	of	Table 3.5-3
SEE ITS 3.3.6	4. CONTAINMENT VENT AND PURGE					
	a. Containment Radioactivity High (R11 and R12 monitor)	2	1	1	0	close all containment vent and purge valves when above cold shutdown
R.8	5. ~~PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING (sample line common with monitor R13)~~	~~1~~	~~NA~~	~~1~~	~~0~~	~~(see note 3)~~
SEE ITS 3.3.3	6. Main Steam Line Radiation Monitors	1/line	NA	1/line	0	(see note 3)
R.8	7. Wide Range Plant Vent Monitor (R27)	1	NA	1	0	(see note 3)

R.8

Relocated Item (R.8)

NOTES

SEE CTS MASTER MARKUP

1. If the conditions of Columns 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition if applicable, within an additional 24 hours.

2. Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

3. ~~If the plant vent sampling capability, the wide-range vent monitor or the main steam line radiation monitors is/are: determined to be inoperable when the reactor is above the cold shutdown condition, then restore the sampling/monitoring capability within 72 hours or:~~
   ~~a) Initiate a pre-planned alternate sampling/monitoring capability as soon as practical, but no later than 72 hours after identification of the failures. If the capability is not restored to operable status within 7 days, then,~~
   ~~b) Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system.~~

R.8

SEE CTS MASTER MARKUP

TABLE 4.1-1 (Sheet 2 of 6)

Channel Description	Check	Calibrate	Test	Remarks
8. 6.9 KV Voltage	N.A.	18M	Q	Reactor protection circuits only
6.9 KV Frequency	N.A.	24M	Q	Reactor protection circuits only
9. Analog Rod Position	S	24M	M	
10. Steam Generator Level	S	24M	Q	
11. Residual Heat Removal Pump Flow	N.A.	24M	N.A.	
12. Boric Acid Tank Level	S	24M	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level				
a. Transmitter	W	18M	N.A.	Low level alarm
b. Indicating Switch	W	6M	N.A.	Low level alarm
14a. Containment Pressure - narrow range	S	24M	Q	High and High-High
14b. Containment Pressure - wide range	M	18M	N.A.	
15. Process and Area Radiation Monitoring:				
SEE ITS 3.3.8 a. Fuel Storage Building Area Radiation Monitor (R-5)	D	24M	Q	
SEE ITS 3.3.6 b. Vapor Containment Process Radiation Monitors (R-11 and R-12)	D	24M	Q	
c. Vapor Containment High Radiation Monitors (R-25 and R-26)	D	24M	Q	
d. Wide Range Plant Vent Gas Process Radiation Monitor (R-27)	D	24M	Q	

R.8

Relocated Item (R-8)

Amendment No. 8, 38, 63, 68, 74, 93, 107, 125, 137, 140, 144, 148, 150, 154, 169

SEE CTS MASTER MARKUP

TABLE 4.1-1 (Sheet 4 of 6)

Channel Description	Check	Calibrate	Test	Remarks
25. Level Sensors in Turbine Building	N.A.	N.A.	24M	
26. Volume Control Tank Level	N.A.	24M	N.A.	
27. Boric Acid Makeup Flow Channel	N.A.	24M	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Level	S	24M	Q	Low-Low
b. Undervoltage	N.A.	24M	24M	
c. Main Feedwater Pump Trip	N.A.	N.A.	24M	
29. Reactor Coolant System Subcooling Margin Monitor	D	18M****	N.A.	
30. PORV Position Indicator	N.A.	N.A.	24M	Limit Switch
31. PORV Position Indicator	D	24M	24M	Acoustic Monitor
32. Safety Valve Position Indicator	D	24M	24M	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	18M	N.A.	
R.8 34. Plant Effluent Radioiodine/Particulate Sampling	N.A.	N.A.	18M	Sample line ~~common~~ with monitor R-13 R8
35. Loss of Power				
a. 480v Bus Undervoltage Relay	N.A.	24M	M	
b. 480v Bus Degraded Voltage Relay	N.A.	18M	M	
c. 480v Safeguards Bus Undervoltage Alarm	N.A.	24M	M	
36. Containment Hydrogen Monitors	D	Q	M	

Amendment No. 38, 44, 54, 65, 67, 74, 93, 125, 136, 137, 142, 144, 150, 168, 169

Relocated Item (R-8)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: **R.8**

AREA RADIATION MONITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING; Plant Wide Range Vent Monitor

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.8: AREA RADIATION MONITORING and PLANT EFFLUENT RADIOIODINE/PARTICULATE SAMPLING

CTS 3.8.A.3, Area Raditiation Monitoring during fuel handling;
CTS 3.8.C.1, Area Raditiation Monitoring during fuel handling;
CTS Table 3.5-4, Plant Effluent Radiation Monitors;
CTS Table 4.1-1, Plant Effluent Radioiodine/Particulate Sampling and Area Monitoring.

Discussion:

All gaseous and particulate effluent from accident releases of radioactivity external to the reactor containment (e.g., the spent fuel pit and waste handling equipment) will be exhausted from the plant vent. Various Air particulate monitors are provided to detect air particulate gamma radioactivity discharges through the plant vent to the atmosphere. The purpose of the Radioactive Gaseous Effluent Instrumentation is to monitor and control radioactive releases. This instrumentation provides a surveillance of release points and initiates automatic alarm/trip functions to terminate the release prior to exceeding the limits of 10 CFR 20. The alarm/trip functions are set in accordance with the ODCM.

Requirements to monitor the containment and spent fuel storage areas using either installed or portable instrumentation is not assumed in the analysis of any event.

Comparison to Selection Criteria:

1. Radioactive gaseous effluent instrumentation and area radiation monitors are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Radioactive gaseous effluent instrumentation and area radiation monitors are not a process variables, design feature, or operating restrictions that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Radioactive gaseous effluent instrumentation and area radiation monitors are not structures, systems, or components that are part of the primary success path and functions or actuates in the mitigation of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, radioactive gaseous effluent instrumentation and area radiation monitors are non-significant risk contributors to core damage frequency and offsite releases. NYPA has reviewed this evaluation and considers it applicable to IP3. Effects of radioactive gaseous effluent instrumentation are outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, radioactive effluent instrumentation and area radiation monitors will be relocated to the Final Safety Analysis Report and requirements will be implemented by plant procedures. There is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.9

AUXILIARY ELECTRICAL SYSTEMS (A.C. Circuit Inside Containment)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.7-3	34	34	No TSCRs	No TSCRs for this Page	N/A
3.7-6	153 TSCR 98-044	153	IPN 98-044	DG Testing when a DG is Inoperable	

SEE ITS 3.8.1

C. If the electrical distribution system is not restored to meet the requirements of 3.7.A within the time periods specified in 3.7.B, then:

1. If the reactor is critical, it shall be in the hot shutdown condition within six hours and in the cold shutdown condition within the following 30 hours.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

D. The requirements of Specification 3.7.A.1 may be modified during an emergency system-wide blackout condition as follows:

Two of the three 13.8 KV feeders (13W92, 13W93 and/or 13W94) to the Buchanan substation 138 KV buses operable with at least 37 MV power from any combination of gas turbines (nameplate rating at 80°F) at the Buchanan Substation and onsite and onsite available for exclusive use on Indian Point Unit No. 3.

R.9

E. ~~Whenever the reactor critical, the circuit breaker on the electrical feeder to emergency lighting panel 318 inside containment shall be locked open except when containment access is required.~~ R.9

SEE ITS 3.8.2

F. As a minimum, under all conditions including cold shutdown, the following A.C. electrical power sources shall be operable:

1. One transmission circuit to Buchanan Substation, except for testing.

2. Either:

a. 6.9 KV buses 5 or 6 energized from the 138 KV feeder 95331 or 95332,

or

b. 13.8 KV feeder 13W92 or 13W93 and its associated 13.8/6.9 KV transformer available to supply 6.9 power,

3. Two of the four 480-volt buses 2A, 3A, 5A and 6A energized.

SEE ITS 3.8.7

Since the backup lighting supply is stripped on safety injection, the requirement that not more than one 120 volt A.C. instrument bus be energized from the backup lighting supply is to assure minimum operable containment spray actuation channels.

~~As a result of an investigation of the effect components that might become submerged following a LOCA may have on ECCS, containment isolation and other safety-related functions, a fuse and a locked open circuit breaker were provided on the electrical feeder to emergency lighting panel 318 inside containment. With the circuit breaker in the open position, containment electrical penetration H-70 is de-energized during the accident condition. Personnel access to containment may be required during power operation. Since it is highly improbable that a LOCA would occur during this short period of time, the circuit breaker may be closed during that time to provide emergency lighting inside containment for personnel safety.~~ R.9

SEE ITS 3.8.1

When the 138 KV source of offsite power is out of service and the 13.8KV power source is being used to feed Buses 5 and 6, the automatic transfer of 6.9 KV Buses 1, 2, 3 and 4 to offsite power after a unit trip could result in overloading of the 20 MVA 13.8 KV/6.9 KV auto-transformer. Accordingly, the intent of specification 3.7.B.3 is to prevent the automatic transfer when only the 13.8 KV source of offsite power is available. However, this specification is not intended to preclude subsequent manual operations or bus transfers once sufficient loads have been stripped to assure that the 20 MVA auto-transformer will not be overloaded by these manual actions.

References

1) FSAR - Section 8.2.1
2) NYPA Calculation, IP3-CALC-EG-00217, Revision 3, dated May 25, 1994.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.9

AUXILIARY ELECTRICAL SYSTEMS (A.C. Circuit Inside Containment)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.9: AUXILIARY ELECTRICAL SYSTEMS (Circuits Inside Containment)

LCO 3.7.E AUXILIARY ELECTRICAL SYSTEMS (A.C. Circuit Inside Containment)

Whenever the reactor is critical, the circuit breaker on the electrical feeder to emergency lighting panel 318 inside containment shall be locked open except when containment access is required.

Discussion:

This specification provides a fuse and a locked open circuit breaker on the electrical feeder to emergency lighting panel 318 inside containment to mitigate any potential effects on ECCS, containment isolation and other safety-related functions if the circuit becomes submerged following a LOCA. The circuits covered by this specification are provided for equipment that is not used during normal plant operation or for accident mitigation. Neither the circuit nor the equipment powered by the circuit have any safety function.

Comparison to Selection Criteria:

1. The circuit described in CTS 3.7.E is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The circuit described in CTS 3.7.E is not a process variable, design feature, or operating restriction that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The circuit described in CTS 3.7.E is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a FSAR accident that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-62) and summarized in

Table 1 of WCAP-11618, the A.C. Circuits Inside Containment were found to be non-significant risk contributors to core damage frequency and offsite releases. NYPA has reviewed this evaluation and considers it applicable to IP3. The circuit described in CTS 3.7.E is outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.]

Conclusion:

Since the selection criteria have not been satisfied, LCO 3.7.E, Auxiliary Electrical Systems (A.C. Circuit Inside Containment) will be relocated to the Final Safety Analysis Report and requirements will be implemented by plant procedures. There is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.10

Refueling, Fuel Handling and Storage (Communications)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.8-1	86	86	No TSCRs	No TSCRs for this Page	N/A

3.8 Refueling, Fuel Handling and Storage

Applicability

Applies to operating limitations during refueling, fuel handling, storage operations, and when heavy loads are moved over the reactor when the head is removed.

Objective

To ensure that no incident could occur during refueling, fuel handling, and storage operations that would adversely affect public health and safety.

Specification

A. During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met: R.10

SEE CTS MASTER MARKUP

1. The equipment door and at least one door in each personnel air lock shall be properly closed. When the closure plate with a personnel door that prevents direct air flow from the containment is used, it shall be properly closed.

2. At least one isolation valve shall be operable, locked closed or blind flanged in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

3. Radiation levels in the containment and spent fuel storage areas shall be monitored continously.

4. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service.

5. At least one residual heat removal pump and heat exchanger shall be operating except during those core alternations in which the residual heat removal flow interferes with component positioning.

6. During reactor vessel head removal and while loading and unloading fuel in the reactor, T_{avg} shall be < 140°F.

R-10

7. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place. R.10

Amendment No. 39, 69, 86

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.10

Refueling, Fuel Handling and Storage (Communications)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.10: Refueling, Fuel Handling and Storage (Communications)

3.8.A, Refueling, Fuel Handling and Storage (Communications)

During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met:

7. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

DISCUSSION:

Communication between the control room personnel and personnel performing Core Alterations is maintained to ensure that personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and containment personnel. However, the refueling system design accident or transient response does not take credit for communications.

COMPARISON TO SCREENING CRITERIA:

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

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

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

4. As discussed in Section 4.0 (Appendix A, page A-67) and summarized in Table 1 of WCAP-11618, the loss of communications was found to be a non-significant risk contributor to core damage frequency and offsite

releases. NYPA has reviewed this evaluation and considers it applicable to IP3. Communication during refueling operations is outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.

CONCLUSION:

Since the screening criteria have not been satisfied, the Communications LCO and Surveillances will be relocated to the Final Safety Analysis Report and requirements will be implemented by plant procedures. There is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.11

Refueling, Fuel Handling and Storage (Decay Time)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.8-2	175	175	No TSCRs	No TSCRs for this Page	N/A
3.8-6	175	175	No TSCRs	No TSCRs for this Page	N/A

SEE ITS 3.9.3 ITS 3.3.6

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

R.11

9. ~~No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 145 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 421* hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal absorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications.~~ R.11

SEE ITS 3.9.3 3.3.6

R.11

~~In the event that more than 76 assemblies are to be discharged from the reactor, those assemblies in excess of 76 shall not be discharged earlier than 267 hours after shutdown.~~ R.11

SEE CTS MASTER MARKUP

10. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

11. Hoists or cranes utilized in handling irradiated fuel shall be dead- load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test and prior to fuel handling. A test of interlocks and overload cutoff devices on the manipulator shall also be performed.

12. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

13. To ensure redundant decay heat removal capability, at least two of the following requirements shall be met:

* Movement of irradiated VANTAGE + fuel assemblies before the reactor has been subcritical for ≥550 hours requires operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers.

R.11

~~The waiting time of 267 hours required following plant shutdown before unloading more than 76 assemblies from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR. The calculations confirming this are based on an inlet river temperature of 95°F, consistent with the FSAR assumptions[2].~~

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45 day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. Fuel Storage Building isolation is actuated upon receipt of a signal from the area high activity alarm or by manual operation. The emergency ventilation bypass assembly is manually isolated, using manual isolation devices, prior to movement of any irradiated fuel. This ensures that all air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers. The ventilation system is tested prior to all fuel handling activities to ensure the proper operation of the filtration system.

When fuel in the reactor is moved before the reactor has been subcritical for at least 421 hours (See footnote on page 3.8-2), the limitations on the containment vent and purge system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The limit to have at least two means of decay heat removal operable ensures that a single failure of the operating RHR System will not result in a total loss of decay heat removal capability. With the reactor head removed and 23 feet of water above the vessel flange, a large heat sink is available for core cooling. Thus, in the event of a single component failure, adequate time is provided to initiate diverse methods to cool the core.

The minimum spent fuel pit boron concentration and the restriction of the movement of the spent fuel cask over irradiated fuel were specified in order to minimize the consequences of an unlikely sideways cask drop.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.11

Refueling, Fuel Handling and Storage (Decay Time)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.11: Refueling, Fuel Handling and Storage (Decay Time)

LCO 3.8.A Refueling, Fuel Handling and Storage (Decay Time)

No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 145 hours.

In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before the interval of 267 hours has elapsed after shutdown.

Discussion:

The requirement that the reactor must be subcritical for 145 hours before moving irradiated fuel in the reactor is needed to meet assumptions for the source term in the dose analysis for a fuel handling accident as stated in FSAR 14.2.

The requirement the reactor must be subcritical for 267 hours before discharging more than 72 assemblies is needed to limit fuel pool the heat load. This decay time ensures that the maximum pool water temperature will be within design objectives as stated in FSAR 9.5.2.

COMPARISON TO SCREENING CRITERIA:

1. Requirements to delay movement of irradiated fuel for a specified period after reactor shutdown are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Requirements to delay movement of irradiated fuel for a specified period after reactor shutdown are assumed in the analysis of a fuel handling accident and spent fuel pool cooling limitations. Although this Specification satisfied criterion 2, the activities necessary prior to commencing movement of irradiated fuel normally provide a significant delay before the movement of irradiated fuel. Administrative controls

have been demonstrated to be very effective in ensuring these requirements are met. Therefore, this Specification has been relocated as per Industry/NRC agreement during the development of NUREG-1431.

3. Not permitting movement of irradiated fuel for a specified period following shutdown is not a primary success path in the mitigation of a DBA or transient.

4. Not permitting movement of irradiated fuel for a specified period following shutdown is outside the scope of the IP3 IPE, and therefore, the plant-specific IPE provides no information to supplement the conclusions from the generic analysis.

CONCLUSION:

Although this Specification satisfied criterion 2 as an initial condition for a fuel handling accident and fuel pool cooling limitations, Although this Specification satisfied criterion 2, the activities necessary prior to commencing movement of irradiated fuel normally provide a significant delay before the movement of irradiated fuel. Administrative controls have been demonstrated to be very effective in ensuring these requirements are met. Therefore, this Specification has been relocated to the Final Safety Analysis Report in accordance with Industry/NRC agreement during the development of NUREG-1431.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.12

Refueling (Manipulator Cranes and Spent Fuel Cask)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.8-1	86	86	No TSCRs	No TSCRs for this Page	N/A
3.8-2	175	175	No TSCRs	No TSCRs for this Page	N/A
3.8-3	114	114	No TSCRs	No TSCRs for this Page	N/A
3.8-4	173	173	No TSCRs	No TSCRs for this Page	N/A
T 4.1-3(1)	182 TSCR 98-043	173	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated

3.8 Refueling, Fuel Handling and Storage

~~**Applicability**~~

~~Applies to operating limitations during refueling, fuel handling, storage operations, and when heavy loads are moved over the reactor when the head is removed.~~

~~**Objective**~~

~~To ensure that no incident could occur during refueling, fuel handling, and storage operations that would adversely affect public health and safety.~~

Specification

R.12 A. ~~During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met:~~ R.12

SEE CTS MASTER MARKUP

1. The equipment door and at least one door in each personnel air lock shall be properly closed. When the closure plate with a personnel door that prevents direct air flow from the containment is used, it shall be properly closed.

2. At least one isolation valve shall be operable, locked closed or blind flanged in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

3. Radiation levels in the containment and spent fuel storage areas shall be monitored continously.

4. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service.

5. At least one residual heat removal pump and heat exchanger shall be operating except during those core alternations in which the residual heat removal flow interferes with component positioning.

6. During reactor vessel head removal and while loading and unloading fuel in the reactor, T_{avg} shall be < 140°F.

7. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

SEE CTS MASTER MARKUP

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

9. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 145 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 421* hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal absorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than 76 assemblies are to be discharged from the reactor, those assemblies in excess of 76 shall not be discharged earlier than 267 hours after shutdown.

10. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

R.12

11. ~~Hoists or cranes utilized in handling irradiated fuel shall be dead- load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test and prior to fuel handling. A test of interlocks and overload cutoff devices on the manipulator shall also be performed.~~

R.12

SEE CTS MASTER MARKUP

12. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

13. To ensure redundant decay heat removal capability, at least two of the following requirements shall be met:

Movement of irradiated VANTAGE + fuel assemblies before the reactor has been subcritical for ≥550 hours requires operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers.

a. No. 31 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

b. No. 32 residual heat removal pump and heat exchanger, together with their associated piping and valves are operable.

c. The water level in the refueling cavity above the top of the reactor vessel flange is equal to or greater than 23 feet.

B. If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

C. During fuel handling and storage operations, the following conditions shall be met:

1. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used.

R-12

2. ~~The spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, no loads in excess of 2,000 pounds shall be moved over any region of the spent fuel pit.~~ R.12

3. ~~During periods of spent fuel cask or fuel storage building cask crane movement over the spent fuel pit, or during periods of spent fuel movement in the spent fuel pit when the pit contains irradiated fuel, the pit shall be filled with borated water at a concentration of >1000 ppm.~~

4. Whenever movement of irradiated fuel in the spent fuel pit is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated in the storage rack.

SEE CTS MASTER MARKUP

R-12

5. ~~Hoists or cranes utilized in handling irradiated fuel shall be deadload tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the fuel handling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test prior to fuel handling.~~

6. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the periods of inoperability.

7. The spent fuel storage racks consist of two regions, as shown on Figure 3.8-3: Region 1 (Columns SS-ZZ, Rows 35-64) and Region 2 (Columns A-RR, Rows 1-34). Fuel storage is restricted in each region as follows:

   a. As specified in Figure 3.8-2, fuel assemblies to be stored in Region 2 shall have a minimum burnup exposure as a function of initial enrichment.

   b. As specified in Figure 3.8-1, fuel assemblies to be stored in Region 1 consist of 3 types (Type A, B, C), depending on their initial enrichment and current burnup. Restrictions on location of fuel in Region 1 are as follows:

      1. Type A assemblies may be stored anywhere in Region 1.

      2. A Type B assembly may be stored anywhere in Region 1, provided it is not face-adjacent to a Type C assembly.

      3. Type C assemblies may not be stored in Row 64 or Column ZZ of Region 1. A Type C assembly may be stored in any other Region 1 location provided that all surrounding (face-adjacent) locations are occupied by Type A assemblies, non-fuel components or empty.

D. When any fuel assemblies are in the reactor vessel and the reactor vessel head bolts are less than fully tensioned, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

SEE CTS MASTER MARKUP

TABLE 4.1-3 (Sheet 1) Relocated Item R.12

SEE CTS MASTER MARKUP

SEE CTS MASTER MARKUP

FREQUENCIES FOR EQUIPMENT TESTS		
	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M*
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
~~6. Refueling System Interlocks~~	~~Functioning~~	~~Each refueling, prior to movement of core components~~ R.12
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop And Control Valves	Closure	Not to exceed 6 months**
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Quarterly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

* Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probablistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.12

Refueling (Manipulator Cranes and Spent Fuel Cask)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.12: Refueling, Fuel Handling and Storage (Manipulator Cranes)

LCO 3.8 Refueling, Fuel Handling and Storage (Manipulator Cranes)

3.8.A During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met:

11. Hoists or cranes utilized in handling irradiated fuel shall be dead load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead load test and prior to fuel handling. A test of interlocks and overload cutoff devices on the manipulator shall also be performed.

3.8.C During fuel handling and storage operations, the following conditions shall be met:

5. Hoists or cranes utilized in handling irradiated fuel shall be dead load tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the fuel handling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead load test prior to fuel handling.

DISCUSSION:

Operability of Hoists or cranes utilized in handling irradiated fuel ensures that the equipment used to handle fuel within the reactor pressure vessel functions as designed and that the equipment has sufficient load capacity for handling fuel assemblies and/or drive rods. Although the interlocks designed to provide the above capabilities can prevent damage to the refueling equipment and fuel assemblies, they are not assumed to function to mitigate the consequences of a design basis accident.

Justification for Relocation of CTS Requirement to
Licensee Controlled Document

Relocated Item R.12: Refueling, Fuel Handling and Storage (Manipulator Cranes)

COMPARISON TO SCREENING CRITERIA:

1. The hoists or cranes utilized in handling irradiated fuel are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The hoists or cranes utilized in handling irradiated fuel are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. Additionally, at IP3, restrictions governing heavy loads were established in accordance with unnumbered generic letter dated December 22, 1980, "Control of Heavy Loads." This generic letter requested that licensees implement the heavy load control guidelines in NUREG-0612 including safe load paths, crane design and inspection, operator training, and procedures.

3. The hoists or cranes utilized in handling irradiated fuel are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-68) and summarized in Table 1 of WCAP-11618, the hoists or cranes utilized in handling irradiated fuel were found to be a non-significant risk contributor to core damage frequency and offsite releases.

CONCLUSION:

Since the screening criteria have not been satisfied, LCO and Surveillances for the hoists and cranes and movement of the spent fuel cask will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement for hoists and cranes and movement of the spent fuel cask. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the

requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.13

Service Water Isolation Valve Leakage (0.36 GPM Lealage Limit)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
4.4-4	174	174	No TSCRs	No TSCRs for this Page	N/A
6-22	174 TSCR 98-018, 98-043	174 TSCR 98-018, 98-043	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated
6-22	174 TSCR 98-018, 98-043	174 TSCR 98-018, 98-043	IPN 98-018	Generic Letter 89-01 and 10 CFR 20 Generic Letter	Incorporated

E. Containment Isolation Valves

SEE ITS 3.6.3 and 3.6.9

1. Verify the combined leakage rate for all containment bypass leakage paths, Table 4.4-1 lists required isolation valves, is ≤ 0.6La when pressurized ≥ Pa, in accordance with the Containment Leakage Rate Testing Program.

2. Verify the leakage rate of water from the Isolation Valve Seal Water System is ≤ 14,700 cc/hr when pressurized ≥ 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program.

R-13

3. ~~Verify the leakage rate of water into the containment from isolation valves sealed with the service water system is ≤ 0.36 gpm per fan cooler unit when pressurized ≥ 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program.~~ R.13

SEE ITS 5.7

6.12.2* In addition to the requirements of 6.12.1 above, areas accessible to individuals with radiation levels such that an individual could receive in 1 hour a dose greater than 1000 mrem**, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the plant Radiological and Environmental Services Manager or his designee.

SEE ITS 5.4

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 Environmental qualification of electric equipment important to safety shall be in accordance with the provisions of 10 CFR 50.49. Pursuant to 10 CFR 50.49, Section 50.49 (d), the EQ Master List identifies electrical equipment requiring environmental qualification.

6.13.2 Complete and auditable records which describe the environmental qualification method used, for all electrical equipment identified in the EQ Master List, in sufficient detail to document the degree of compliance with the appropriate requirements of 10 CFR 50.49 shall be available and maintained at a central location. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

SEE ITS 5.5.15

~~6.14~~ CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, Dated September 1995" as modified by the following exception:

a. ANS 56.8 - 1994, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 42.39 psig. The minimum test pressure is 42.42 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

See TSCR 98-043 Next Page.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤0.60 L_a for the Type B and C tests and ≤0.75 L_a for Type A tests;

b. Air lock acceptance criteria are:

1) Overall the air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a,

2) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ P_a.

R-13 :. Isolation valves sealed with the service water system leakage rate into containment acceptance criterion is ≤0.36 gpm per fan cooler unit R-13

SEE ITS 5.7

* Health Physics Personnel shall be exempt from the RWP issuance requirements for entries into high radiation areas during the performances of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

** Measured at 30 centimeters (12 inches) from the source of radioactivity.

TSCR 98-018

Amendment No. 11, 59 (Order dated October 24, 1980), 88, 181, 183, 116, 117, 162, 174

See Also TSCR 98-043

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.13

Service Water Isolation Valve Leakage (0.36 GPM Lealage Limit)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
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Relocated Item R.13:

Surveillance Requirement 4.4.E.3

Verify the leakage rate of water into the containment from isolation valves sealed with the service water system is < 0.36 gpm per fan cooler unit when pressurized > 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program.

Discussion:

This requirement exists to ensure that inleakage from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation will not result in flooding the internal recirculation pumps.

Comparison to Selection Criteria:

1. Limits for the leakage rate of water into the containment from isolation valves sealed with the service water system are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The leakage rate of water into the containment from isolation valves sealed with the service water system is not a process variable, design feature, or operating restriction that is an initial condition of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The leakage rate of water into the containment from isolation valves sealed with the service water system is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a FSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Justification for Relocation of CTS Requirement to
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Relocated Item R.13:

4. The leakage rate of water into the containment from isolation valves sealed with the service water system is not important for any scenarios modeled in the IP3 Individual Plant Examination (IPE).

Conclusion:

Since the selection criteria have not been satisfied, the leakage rate of water into the containment from isolation valves sealed with the service water system will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain this requirement. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.14

RADIOACTIVE MATERIALS MANAGEMENT

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.9-1	0	0	No TSCRs	No TSCRs for this Page	N/A
3.9-2	0	0	No TSCRs	No TSCRs for this Page	N/A
6-18	157 TSCR 98-018	157 TSCR 98-018	IPN 98-018	Generic Letter 89-01 and 10 CFR 20 Generic Letter	Incorporated

Relocated Item (R-14)

3.9 RADIOACTIVE MATERIALS MANAGEMENT

Applicability

Applies to the handling and use of sealed special nuclear, source and by-product material.

Objective

To assure that leakage from by-product, source, and special nuclear radioactive material sources does not exceed allowable limits.

R.14

Specification

A. Tests for leakage and/or contamination shall be performed as follows.

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen-3, with a half life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date or use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Startup sources shall be leak tested prior to being subjected to core flux and following repair or maintenance to the source.

R.14

B. Sealed sources are exempt from Specification 3.9.A when the source contains 100 microcuries or less of beta and/or gamma emitting material or 5 microcuries or less of alpha emitting material.

C. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with Commission regulations.

R.14

SEE ITS 5.6

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety limits are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

R.14

SEE CTS MASTER MARKUP

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Deleted

g. Release of radioactive effluents in excess of limits ~~(Appendix B Specifications 2.3, 2.4, 2.5, 2.6)~~ TSCR 98-018

Amendment No. ~~10~~, ~~11~~, ~~12~~, ~~31~~, ~~44~~, ~~51~~, ~~59~~, ~~63~~, ~~68~~, ~~67~~, ~~83~~, ~~88~~, ~~103~~, ~~108~~, ~~118~~, ~~117~~, 157

TSCR 98-018

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RADIOACTIVE MATERIALS MANAGEMENT

PART 2:

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Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.14: RADIOACTIVE MATERIALS MANAGEMENT

LCO 3.9 RADIOACTIVE MATERIALS MANAGEMENT

Tests for leakage and/or contamination of sealed sources

DISCUSSION:

The limitations on sealed source contamination are intended to ensure that the total body and individual organ irradiation doses do not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limit for removable contamination on each sealed source. This requirement and the associated surveillance requirements are not conditions or limits necessary for safe reactor operation.

COMPARISON TO SCREENING CRITERIA:

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

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

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

4. As discussed in Section 4.0 (Appendix A, page A-59) and summarized in Table 1 of WCAP-11618, sealed source contamination not within limits was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to IP3, and concurs with the assessment.

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.14: RADIOACTIVE MATERIALS MANAGEMENT

CONCLUSION:

Since the selection criteria have not been satisfied, tests for leakage and/or contamination of sealed sources will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the tests for leakage and/or contamination of sealed sources. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.15

MOVABLE INCORE INSTRUMENTATION

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.11-1	122	122	No TSCRs	No TSCRs for this Page	N/A
3.11-2	122	122	No TSCRs	No TSCRs for this Page	N/A

3.11 MOVABLE INCORE INSTRUMENTATION

~~Applicability~~

~~Applies to the operability of the movable detector instrumentation system.~~

~~Objective~~

~~To specify functional requirements on the use of the incore instrumentation system, for the recalibration of the excore axial off-set detection system.~~

Specification

A. ~~A minimum of 2 thimbles per quadrant and sufficient movable incore detectors shall be operable during recalibration of the excore axial off-set detection system.~~ R.15

SEE ITS 3.2.3 ITS 3.3.1

B. Power shall be limited to 90% of rated power if recalibration requirements for the excore axial off-set detection system, identified in Table 4.1-1, are not met.

C. ~~During the incore/excore calibration procedure, all full core flux maps will be made only when at least 38 of the movable detector guide thimbles are operable.~~ R.15

~~Basis~~

~~The Movable Incore Instrumentation System[1] has six drives, six detectors, and 58 movable detector guide thimbles in the core. Fifty (50) of these thimbles were provided as part of the original design basis of the plant. The other eight thimbles are supplemental thimbles that were connected during the 8/9 refueling outage. The eight supplemental thimbles are maintained to the same standards as the original 50 thimbles. These eight supplemental thimbles can be used to satisfy the 38 thimble requirement for flux mapping. An appropriate evaluation will be performed prior to the initial use of the supplemental thimbles to satisfy technical specification requirements for flux mapping. The eight supplemental thimbles improve the reliability of the Movable Incore Instrumentation System. Each of the six movable incore detectors can be routed to sixteen or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.~~

~~To calibrate the excore detectors, it is only necessary that the Movable Incore Instrumentation System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.~~

Relocated Item (R-15)

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor as it will compensate for an error of 10% in the excore protection system. Experience at Beznau No. 1 and R.E. Ginna plants has shown that drift due to changes in the core or instrument channels is very slight. Thus the 10% reduction is considered to be very conservative.

R.15

Reference

(1) FSAR - Section 7.6

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MOVABLE INCORE INSTRUMENTATION

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.~~14~~ 15: MOVABLE INCORE INSTRUMENTATION

LCO 3.11 MOVABLE INCORE INSTRUMENTATION

A. A minimum of 2 thimbles per quadrant and sufficient movable incore detectors shall be operable during recalibration of the excore axial off-set detection system.

C. During the incore/excore calibration procedure, all full core flux maps will be made only when at least 38 of the movable detector guide thimbles are operable.

DISCUSSION:

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

COMPARISON TO DETERMINISTIC SCREENING CRITERIA:

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

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

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

4. As discussed in Section 4.0 (Appendix A, page A-12) and summarized in Table 1 of WCAP-11618, the loss of Movable Incore Detectors was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to IP3, and concurs with the assessment.

Justification for Relocation of CTS Requirement to
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Relocated Item R.~~14~~ 15: MOVABLE INCORE INSTRUMENTATION

CONCLUSION:

Since the screening criteria have not been satisfied, the requirements associated with movable incore detectors will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement for movable incore detectors. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.16

RIVER LEVEL (Flooding Protection)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.12-1	34	34	No TSCRs	No TSCRs for this Page	N/A

3.12 RIVER LEVEL

Applicability

Applies to water elevation of the Hudson River as measured at the Indian Point Unit No. 3 intake structure.

Objective

To specify the maximum water elevation of the Hudson River for safe operation of the reactor.

R.16

Specification

R-16

When the Hudson River water elevation as measured at the Indian Point Unit Unit No. 3 intake structure reaches 11'-0" above mean sea level, sandbagging the service water pumps will be initiated. If the Hudson River water elevation reaches 12'-5" above mean sea level at the Indian Point Unit No. 3 intake structure, the reactor will be in the hot shutdown condition within six hours and in the cold shutdown condition within the following 30 hours.

R.16

Basis

Analyses have been performed wich indicate that the river water elevation would have to reach 15'-3" above mean sea level before it would seep into the lowest floor elevation of any of the buildings housing equipment vital for safe shutdown of the reactor. (1) Monitoring of the Hudson River water elevation will not be required until there is a flood warning notice disseminated by the New York City National Oceanographic and Atmosphere Administration (NOAA) office.

References:

(1) FSAR, Section 2.5

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RIVER LEVEL (Flooding Protection)

PART 2:

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Justification for Relocation of CTS Requirement to
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Relocated Item R.16: RIVER LEVEL (Flooding Protection)

LCO 3.12 **RIVER LEVEL** (Flooding Protection)

When the Hudson River water elevation as measured at the Indian Point Unit No. 3 intake structure reaches 11'-0" above mean sea level, sandbagging the service water pumps will be initiated. If the Hudson River water elevation reaches 12'-5" above mean sea level at the Indian Point Unit No. 3 intake structure, the reactor will be in the hot shutdown condition within six hours and in the cold shutdown condition within the following 30 hours.

Discussion:

Analyses have been performed which indicate that the river water elevation would have to reach 15'-3" above mean sea level before it would seep into the lowest floor elevation of any of the buildings housing equipment vital for safe shutdown of the reactor.

Comparison to Screening Criteria:

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

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

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

4. As discussed in Section 4.0 (Appendix A, page A-56) and summarized in Table 1 of WCAP-11618, Flooding Protection was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to IP3, and concurs with the assessment.

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.16: RIVER LEVEL (Flooding Protection)

Conclusion:

Since the screening criteria have not been satisfied, flooding protection requirements will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement for flooding protection. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.17

SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.13-1	83	83	No TSCRs	No TSCRs for this Page	N/A
3.13-2	83	83	No TSCRs	No TSCRs for this Page	N/A
3.13-3	83	83	No TSCRs	No TSCRs for this Page	N/A
4.11-1	111	111	No TSCRs	No TSCRs for this Page	N/A
4.11-2	111	111	No TSCRs	No TSCRs for this Page	N/A
4.11-3	165 TSCR 98-043	165	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated
4.11-4	111	111	No TSCRs	No TSCRs for this Page	N/A
4.11-5	125	125	No TSCRs	No TSCRs for this Page	N/A

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SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

4.11-6	**111**	**111**	**No TSCRs**	No TSCRs for this Page	**N/A**
4.11-7	**111**	**111**	**No TSCRs**	No TSCRs for this Page	**N/A**

3.13 SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the operability of snubbers required for protection of safety related components.

Objective

To define the time during which reactor operation is permitted after detection of inoperable snubbers.

Specification

1. During any mode of operation for which a safety-related system is required to be operable, the snubbers in such systems shall be OPERABLE except as noted in 3.13.2 and 3.13.3 below. The requirements of snubber operability shall be satisfied within 7 days for the residual heat removal system when the unit is in cold shutdown and snubbers are being removed for scheduled testing or routine maintenance.

2. If one or more safety-related snubbers are determined to be inoperable in a system which at that time is required to be operable, then within 72 hours, perform section 3.13.2.a and 3.13.2.b:

   a.(1)Replace or restore the inoperable snubber(s) to OPERABLE status,

   or

   (2)perform an engineering evaluation which shows that the inoperable snubber is not required.

---

* Safety-related snubbers include those snubbers installed on safety-related systems and snubbers on non safety-related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety-related system.

R.17

R.17

b. Perform an engineering evaluation per Technical Specification 4.11.B.4 on the supported system or component.

If the requirements of section 3.13.2 cannot be met or the results of the applicable evaluations performed by section 3.13.2 are unacceptable, then the supported system shall be declared inoperable and the appropriate limiting condition for operation action statement for that system shall be followed. If an engineering evaluation demonstrates that the component or system is still operable, i.e., not degraded by the inoperability of the subject snubber(s), the supported system or component need not be declared inoperable.

3. If one or more safety-related snubbers, are determined to be inoperable in a system which at that time is <u>not</u> required to be OPERABLE, then prior to bringing the reactor to that condition for which such system is required to be operable, perform sections 3.13.3.a and 3.13.3.b:

a.(1) Replace or restore the inoperable snubbers(s) to OPERABLE status,

or

(2) perform an engineering evaluation which shows that the inoperable snubber is not required.

b. Perform an engineering evaluation per Technical Specification 4.11.B.4 on the supported system or component.

If the requirements of section 3.13.3 cannot be met or the results of the applicable evaluations performed by section 3.13.3. are unacceptable, then the supported system shall be declared inoperable and the appropriate limiting condition for operation action statement for that affected system shall be followed. If an engineering evaluation demonstrates that the component or system is still operable, i.e., not degraded by the inoperability of the subject snubber(s), the supported system or component need not be declared inoperable.

R.17

Basis

Snubbers are required to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion. The consequences of an inoperable snubber can be an increase in the probability of structural damage to piping in the event of dynamic or thermal loads. It is therefore required that snubbers necessary to protect the primary coolant system or any other safety system or component be operable. Because the snubber lockup protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements before the system must be declared inoperable unless an engineering evaluation can prove otherwise. The engineering evaluations from items 3.13.2.a.(2) and 3.13.3.a.(2) shall determine whether or not the operability of a system or component may be affected by eliminating a redundant inoperable snubber. The engineering evaluations from paragraphs 3.13.2.b and 3.13.3.b shall determine if the system or component supported by a failed snubber experienced degradation that would prevent the system or component from performing its intended function in its intended manner assuming that the required action statements of sections 3.13.2.a and 3.13.3.a were performed as necessary.

References

1) Generic Letter 84-13, "Technical Specifications For Snubbers"

4.11 SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the periodic inspection and testing requirements for all safety-related hydraulic snubbers that are required to protect the primary coolant system or any other safety-related system or component.

Objective

To verify that safety-related snubbers will perform their design functions in the event of a seismic or other transient dynamic event.

Specification

A. Visual Inspection

1. Safety-related snubbers shall be visually inspected in accordance with the following schedule:

Size of Population or Category (Notes 1 & 2)	Number Of Unacceptable Snubbers		
	Column A Extend Interval (Note 3)	Column B Repeat Interval (Note 4)	Column C Reduce Interval (Note 5)
1	0	0	1
20	0	0	1
40	0	0	1
60	0	0	1
80	0	0	2
90	0	0	3
100	0	1	4
120	0	1	5
130	0	2	6
140	0	2	7
150	0	3	8
160	0	3	9
170	0	3	10
180	1	4	11
190	1	4	12
200	2	5	13

Amendment No. ~~6~~, ~~57~~, ~~83~~, 111,

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Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. This decision shall be made and documented before any inspection and shall be used as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. The next lower integer for the value of the limit for Columns A, B, C shall be used if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B, but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

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2. Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundations or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for the particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.11.B.5. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable via functional testing for the purpose of establishing the next visual inspection period. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

B. Functional Testing

1. At least once per 24 months* during plant shutdown, a representative sample of 10% of all the safety-related hydraulic snubbers shall be functionally tested for operability, either in place or on a bench test. For each snubber that does not meet the requirement of 4.11.B.5, an additional 10% of the total installed of that type of hydraulic snubber shall be functionally tested. This additional testing will continue until no failures are found or until all snubbers of the same type have been functionally tested. The representative sample shall include each size and type of snubber in use in the plant.

2. The representative sample selected for functional testing should include the various configurations, operating environments, sizes and capacities of snubbers. At least 25% or the maximum possible if less than 25%, of the snubbers in the representative sample should include snubbers from the following three categories:

   a. The first snubber away from each reactor vessel nozzle.

* Snubber functional testing due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.

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b. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)

c. Snubbers within 10 feet of the discharge from a safety or relief valve.

Snubbers identified as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative samples".

Snubber selection for functional testing is developed from an engineering evaluation and is based on a rotating basis. In addition to the regular sample, snubber locations which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the previously failed snubber (if it is repaired and currently installed in another position) and the installed spare snubber shall be retested. Test results of these snubbers may not be included for the sampling required by Specification 4.11.B.1.

3. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same manufacturer and model, subject to the same defect and located in a similar environment, shall be functionally tested.

4. For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the inoperable snubber(s) remain capable of performing their intended function in their intended manner after the action statements of Specification 3.13.2.a or 3.13.3 a were performed as necessary.

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* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions.

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5. The hydraulic snubber functional test shall verify that:

   a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

   b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

C. Snubber Service Life Monitoring

1. A record of the service life of each snubber, the date at which the designated service life commences, as well as the installation and maintenance records on which the designated service life is based shall be maintained as required by specification 6.10.2.o. The service life may be modified based on a performance evaluation.

2. At least once per 24 months the installation and maintenance records for each safety-related snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

Basis

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Performance of periodic visual inspections of snubbers complements the existing functional testing and provides additional confidence in snubber operability. The visual inspection interval for the snubbers is based on the number of unacceptable snubbers found during the previous inspection in proportion to the sizes of the various populations or categories and may be as long as two refueling cycles with good overall visual inspection results. The visual inspection interval will not exceed 48 months. However, as for all surveillance activities, unless otherwise noted, allowable tolerances of 25% are applicable for snubbers. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The words "will not exceed" associated with a surveillance interval does not negate this allowable tolerance. Inspections performed before the interval has elapsed may be used as a new reference point to determine the next scheduled inspection; however, the results of such early

inspections performed before the original required time interval has elapsed may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule. The results of random inspections of individual snubbers, conducted at other than scheduled inspection intervals, will be evaluated on a case-by-case basis to determine if they should impact the scheduled interval.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified operable by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, and are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system by determining if the system or component was exposed to a dynamic transient which required the inoperable snubber to mitigate the transient.

To provide assurance of snubber functional reliability, a representative sample of 10% of the installed snubbers will be functionally tested during plant shutdowns. The representative sample selected for functional testing includes various configurations, operating environments, locations and the range of size and capacity of snubbers. An engineering evaluation which addresses snubber performance environments and history selects the representative sample which is based on a rotating basis. Selection of a representative sample of hydraulic snubbers provides a confidence level within acceptable limits that these supports will be in an operable condition. Observed failures of these sample snubbers shall require functional testing of additional units of the same type.

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If a snubber fails a functional test, that snubber location will be retested during the next snubber testing period to determine if the failure was environmentally caused. If the failed snubber was repaired and re-installed elsewhere in the system, during the functional test effort the snubber will be retested during the next testing period to verify if the repair addressed the cause of a failure. If a failed snubber is repaired and not reinstalled in the system during the functional test effort it shall be retested before it is subsequently installed in the system as added assurance that the repair addressed the cause of failure. The results of these augmented testing efforts are intended to address previous failure modes and these test results (passing or failure) may not be included in the specification 4.11.B.1 sample selection.

The service life of a snubber is evaluated via engineering evaluation, test data, service data, manufacturer input, snubber service conditions and snubber service history (newly installed snubber, seal replaced, spring replaced, in high radiation area, high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

References

1) Generic Letter 84-13, "Technical Specifications For Snubbers."

2) Generic Letter 90-09, "Alternative Requirements For Snubber Visual Inspection Intervals and Corrective Actions."

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Relocated Item No: R.17

SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.17: SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

LCO 3.13 SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)
LCO 4.11 SAFETY-RELATED SHOCK SUPPRESSORS (SNUBBERS)

LCO Statement:

During any mode of operation for which a safety-related system is required to be operable, the snubbers in such systems shall be OPERABLE except as noted in 3.13.2 and 3.13.3 below. The requirements of snubber operability shall be satisfied within 7 days for the residual heat removal system when the unit is in cold shutdown and snubbers are being removed for scheduled testing or routine maintenance.

Discussion:

Snubbers are required to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion. The consequences of an inoperable snubber can be an increase in the probability of structural damage to piping in the event of dynamic or thermal loads.

Comparison to Screening Criteria:

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

2. Shock suppressors are not a process variable, design feature or operating restriction that is an initial condition of a DBA or transient.

3. Shock suppressors are not part of a primary success path in the mitigation of a DBA or transient. However, the consequences of an inoperable snubber can be an increase in the probability of structural damage to piping in the event of dynamic or thermal loads. It is required that snubbers necessary to protect the primary coolant system

or any other safety system or component be operable.

4. As discussed in Section 4.0 (Appendix A, page A-57) and summarized in Table 1 of WCAP-11618, Shock suppressors were found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the snubber LCO and surveillances may be relocated to other plant controlled documents outside the Technical Specifications. However, although this LCO does not satisfy any of the TS selection criteria, snubbers are being retained in the ISI/IST programs as negotiated by the Industry and NRC during development of NUREG-1431. NYPA as reviewed this evaluation, considers it applicable to IP3, and concurs with the assessment. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.18

Toxic Gas Monitoring

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.3-11	154	154	No TSCRs	No TSCRs for this Page	N/A
3.3-12	115	115	No TSCRs	No TSCRs for this Page	N/A
4.5-5	125	125	No TSCRs	No TSCRs for this Page	N/A
6-19	117 TSCR 98-018	117 TSCR 98-018	IPN 98-018	Generic Letter 89-01 and 10 CFR 20 Generic Letter	Incorporated

G. Containment Hydrogen Monitoring Systems

1. One hydrogen monitor including a flow path and associated containment fan cocler unit shall be OPERABLE whenever the reactor T_{avg} exceeds 350°F.

   a. The requirements of 3.3.G.1 can be modified to allow both containment hydrogen monitoring systems to be inoperable for a period not to exceed 7 days.

H. Control Room Ventilation System

SEE ITS 3.3.3 ITS 3.3.7 ITS 3.7.11

1. The control room ventilation system shall be operable at all times when containment integrity is required as per specification 3.6.

2. The requirements of 3.3.H.1 may be modified as follows:

   a. The control room ventilation system may be inoperable for a period not to exceed seventy-two hours. At the end of this period, if the mal-condition in the control room ventilation system has not been corrected, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If after an additional 48 hours the mal-condition still exists, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3. Two independent toxic gas monitoring systems, with separate channels for detecting chlorine, ammonia, and oxygen shall be operable in accordance with 3.3.H.1 except as specified below. The alarms for ammonia and chlorine shall be adjusted to actuate at ≤35 ppm and ≤3 ppm, respectively.

   a. With any channel for a monitored toxic gas inoperable, restore the inoperable channel to operable status within 7 days.

   b. If 3.a above cannot be satisfied within the specified time, then within the next 8 hours initiate and maintain operation in the control room of alternate monitoring capability for the inoperable channel.

   c. With both channels for a monitored gas inoperable, within 8 hours initiate and maintain operation in the control room of an alternate monitoring system capable of detecting the gas monitored by the inoperable channel.

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AND

1. Within 72 hours after identification of the inoperability of both installed monitoring channels, restore one monitoring channel to operable status.

OR

2. Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the monitoring systems.

I. Electric Hydrogen Recombiner System

1. Two independent Hydrogen Recombiner Systems shall be OPERABLE whenever the reactor T_{avg} exceeds 350°F.

a. With one Hydrogen Recombiner System inoperable, restore the inoperable system to operable status within 30 days, or be in the HOT SHUTDOWN CONDITION within the next 6 hours and subsequently reduce T_{avg} to less than or equal to 350°F within the following 30 hours.

b. The reactor operating condition may be escalated while one Hydrogen Recombiner System is inoperable provided the requirements of section 3.3.I.1.a. above, are satisfied.

SEE ITS 3.6.8

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e. ~~Each toxic gas monitoring system shall be demonstrated operable by performance of a channel check at least once per day, a channel test at least once per 31 days and a channel calibration at least once per 18 months.~~

6. Fuel Storage Building Emergency Ventilation System

a. The fuel storage building emergency ventilation system fan shall be operated for a minimum of 15 minutes every month when there is irradiated fuel in the spent fuel pit.

b. Prior to handling of irradiated fuel, the following conditions shall be demonstrated before the system can be considered operable:

(1) The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at ambient conditions and accident design flow rates.

(2) Using either direct or indirect measurements, the flow rate of the system fans shall be shown to be at least 90% of the accident design flow rate.

(3) The filtration system bypass assembly shall be isolated and leak tested to assure that it is properly sealed.

c. Prior to handling of irradiated fuel, or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal adsorber use since the last test, the following conditions shall be demonstrated before the system can be considered operable:

(1) Charcoal shall have a methyl iodine removal efficiency ≥ 90% at ± 20% of the accident design flow rate, 0.05 to 0.15 mg/m^3 inlet methyl iodine concentration, ≥ 95% relative humidity and ≥ 125°F.

(2) A halogenated hydrocarbon (freon) test on charcoal adsorbers at ± 20% of the accident design flow rate and ambient conditions shall show ≥ 99% halogenated hydrocarbon removal.

SEE ITS 3.7.13 ITS 5.5.10

h. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24)

i. Radioactive environmental sampling results in excess of reporting levels (~~Appendix B Specification 2.7, 2.8, 2.9~~ **Technical Specification 6.8.4.b**)

j. Operation of Overpressure Protection System (Specification 3.1.A.8.c)

R.18 k. Operation of Toxic Gas Monitoring Systems (Specification 3.3.H.3.) R.18

~~l. Inoperable explosive gas monitoring instrumentation~~ **(Appendix B, Technical Specification 1.1.1)**

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacements of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

SEE CTS MASTER MARKUP

Amendment No. 11, 12, 47, 51, 59, 88, 101, 103, 108, 116, 117,

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Relocated Item No: R.18

Toxic Gas Monitoring

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.18: Toxic Gas Monitoring

CTS 3.3.H

Two independent toxic gas monitoring systems, with separate channels for detecting chlorine, ammonia, and oxygen shall be operable.

Discussion:

Toxic gas detection systems ensure that sufficient capability is available to promptly detect an accidental chlorine release of toxic material. These alarms allow operators to initiate protective action to isolate the control room. NRC requirements regarding the relationship of the toxic gas detection systems to the general design criteria (GDC) appear in NUREG-0800, "Standard Review Plan"; Regulatory Guide (RG) 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated hazardous Chemical Release"; and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

Comparison to Selection Criteria:

Generic Letter 95-10, Relocation of Selected Technical Specifications Requirements Related to Instrumentation, provides the following justification for relocating requirements for toxic gas monitoring out of the Technical Specifications:

Chlorine detection systems may serve an important role in protecting control room personnel from internal or external hazards related to toxic gases. However, the release of chlorine or other hazardous chemicals is not part of an initial condition of a design basis accident or transient analysis that assumes a failure of or presents a challenge to the integrity of a fission product barrier. Since the release of toxic gases is not assumed to initiate or occur simultaneously with design basis accidents or transients involving challenges to fission product barriers, the chlorine detection system is not part of a success path for the mitigation of those accidents or transients. The staff has, therefore, concluded that requirements for this system do not

meet the 10 CFR 50.36 criteria and need not be included in Technical Specifications.

Conclusion:

Since the selection criteria have not been satisfied, toxic gas monitoring requirements will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement for toxic gas monitoring. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Relocated Item No: R.19

REACTOR COOLANT SYSTEM INTEGRITY TESTING

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
4.3-1	179	179	No TSCRs	No TSCRs for this Page	N/A

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4.3 REACTOR COOLANT SYSTEM (RCS) TESTING

A. Reactor Coolant System Integrity Testing

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following refueling, repair, replacement or modification.

Specification

a) The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of ASME Section XI.

b) Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of ASME Section XI.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first 13.3 EFPYs of operations. Figure 4.3-1 will be recalculated periodically. Allowable pressures during cooldown from the leak test temperature shall be in accordance with Figure 3.1-2.

Basis

Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during operation. The test frequency and conditions are specified in the Code.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operation.

The inservice leak test temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

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REACTOR COOLANT SYSTEM INTEGRITY TESTING

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.19: REACTOR COOLANT SYSTEM INTEGRITY TESTING

LCO 4.3.b REACTOR COOLANT SYSTEM INTEGRITY TESTING

When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of ASME Section XI.

DISCUSSION:

The inspection and repair programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components life. Other Technical Specifications require important systems to be operable (for example, ECCS) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this Specification to ensure immediate operability of safety systems. Further, this Technical Specification prescribes inspection (and repair) requirements which are performed during plant shutdown.

COMPARISON TO SCREENING CRITERIA:

1. The inspection and repair programs for ASME Code Class 1, 2, and 3 components stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The inspection and repair programs for ASME Code Class 1, 2, and 3 components stipulated by this Specification are not a process variable, design feature, or operating restriction that is an initial assumption in a DBA or transient.

3. The ASME Code Class 1, 2, and 3 components inspected and/or repaired under this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification only specifies inspection requirements for these components; and these inspections can

only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.

4. As discussed in Section 4.0 (Appendix A, page A-43) and summarized in Table 1 of WCAP-11618 the assurance of operability of the entire system as verified in the system operability Specification dominates the risk contribution of the system. The lack of a long term assurance of structural integrity as stipulated by this Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to IP3, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Reactor Coolant System Integrity Testing LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. Chapter 5.0 of the IP3 Improved Technical Specifications will contain a section which provides a programmatic approach to the requirements relating to the structural integrity of ASME Code Class 1, 2, and 3 components.

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Relocated Item No: R.20

SEISMIC INSTRUMENTATION

PART 1:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show Relocated CTS Requirement

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
4.10-1	0	0	No TSCRs	No TSCRs for this Page	N/A
4.10-2	0	0	No TSCRs	No TSCRs for this Page	N/A
T 4.10-1	0	0	No TSCRs	No TSCRs for this Page	N/A
T 4.10-2	164	164	No TSCRs	No TSCRs for this Page	N/A
6-18	157 TSCR 98-018	157 TSCR 98-018	IPN 98-018	Generic Letter 89-01 and 10 CFR 20 Generic Letter	Incorporated

4.10 SEISMIC INSTRUMENTATION

Applicability

Applies to testing of seismic monitoring instruments.

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required.

Specification

4.10.1 Each of the seismic monitoring instruments in Table 4.10-1 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.10-2.

4.10.2 If the number of OPERABLE seismic monitoring instruments is less than that required by Table 4.10-1, restore the inoperable instrument(s) to OPERABLE status within 30 days.

4.10.3 With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

4.10.4 Each of the seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed within 48 hours following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

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Basis

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility and is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes", April, 1974.

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Relocated Item (R-20)

TABLE 4.10-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelographs		
a. EL 46'-0" VC Base Mat	0 to ± 1G	1*
b. EL 99'-0" VC Wall	0 to ± 1G	1*
2. Triaxial Peak Accelographs		
a. STM GEN # 31	0 to ± 2G	1
b. RC Pump # 31	0 to ± 2G	1
c. Pressurizer	0 to ± 2G	1
3. Triaxial Response-Spectrum Recorders		
a. EL 46'-0" VC Base Mat	0 to ± 1G	1*

R.20

* With reactor control room indication

Relocated Item (R-20)

TABLE 4.10-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS			
INSTRUMENTS AND SENSOR LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. EL 46' -0" VC Base Mat	M*	24M	SA
b. EL 99' -0" VC Wall	M*	24M	SA
2. Triaxial Peak Accelographs			
a. STM GEN #31	NA	24M	NA
b. RC Pump #31	NA	24M	NA
c. Pressurizer	NA	24M	NA
3. Triaxial Response-Spectrum Recorders			
a. EL 46' -0" VC Base Mat**	M	24M	SA

24M - At least once per 24 months

* Except seismic trigger.
** With reactor control room indications.

R.20

Amendment No. ~~125~~, 164

SEE ITS 5.6

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety limits are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator-Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification; R.20

SEE R.14

a. Sealed source leakage on excess of limits (Specification 3.9)

R.20

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

R.20

SEE ITS 5.6

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Deleted

g. Release of radioactive effluents in excess of limits ~~(Appendix B Specifications 2.3, 2.4, 2.5, 2.6)~~ TSCR 98-018

Amendment No. 10, 11, 12, 31, 44, 51, 59, 63, 66, 67, 73, 88, 103, 108, 118, 117, 157

TSCR 98-018

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Relocated Item No: R.20

SEISMIC INSTRUMENTATION

PART 2:

Justification for Relocation of CTS Requirement to Licensee Controlled Document

Justification for Relocation of CTS Requirement to
Licensee Controlled Document
Relocated Item R.20: SEISMIC INSTRUMENTATION

LCO 4.10 SEISMIC INSTRUMENTATION

Each of the seismic monitoring instruments shall be demonstrated OPERABLE.

Discussion:

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

Comparison to Deterministic Screening Criteria:

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

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

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

4. As discussed in Section 4.0 (Appendix A, page A-22), and summarized in Table 1 of WCAP-11618, the loss of seismic monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to IP3, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Seismic Instrumentation LCO and Surveillances will be relocated to the Final Safety Analysis Report and will be implemented by administrative programs and plant

procedures.

This change is acceptable because the FSAR and plant procedures will maintain the requirement for Seismic monitoring. Therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

Maintaining this requirement in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, is designed to assure that changes to the FSAR do not result in any of the following: changes to the Technical Specification requirements; significant increases in the probability or consequences of accidents previously evaluated; the possibility of a new or different kind of accident; or a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review. There, this change will maintain an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3

Improved

Technical Specifications

Conversion Submittal

New York Power Authority

New York Power Authority

Improved Technical Specifications Conversion Submittal

VOLUME 2

SECTION III

IP3 ITS 1.0
through
IP3 ITS 3.0

INDIAN POINT 3

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 1.0:
"USE AND APPLICATION"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed

(continued)

1.1 Definitions

CHANNEL CALIBRATION (continued)

sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL OPERATIONAL TEST (COT)

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev.1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".

$\bar{E}$ – AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total noniodine activity in the coolant.

L_a

The maximum allowable primary containment leakage rate, L_a, shall be 1% of primary containment air weight per day at the calculated peak containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

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

(Leakage into closed systems is leakage that can be accounted for and contained by a

(continued)

1.1 Definitions

LEAKAGE (continued)

system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and leakage past the safety injection pressure isolation valves are examples of reactor coolant system leakage into closed systems.)

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

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

b. Unidentified LEAKAGE

All LEAKAGE (except for leakage into closed systems and RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

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

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant loop temperature, and reactor

(continued)

1.1 Definitions

MODE (continued)	vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in FSAR Chapter 13, Initial Tests and Operations; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)."

(continued)

1.1 Definitions

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3025 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power level.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

(continued)

1.1 Definitions

THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

1.1 Definitions

Table 1.1-1 (page 1 of 1)
MODES

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

(a) Excluding decay heat.

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

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1

ACTIONS

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

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

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

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

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each

(continued)

1.3 Completion Times

DESCRIPTION (continued)

additional failure, with Completion Times based on initial entry into the Condition.

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

a. Must exist concurrent with the first inoperability; and

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

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

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

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

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

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition

(continued)

1.3 Completion Times

DESCRIPTION (continued)

entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR-3.0.3 would apply.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 1.0:
"USE AND APPLICATION"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
1-1	170	170	IPN 96-063	Leakage Limits for RCS and SIS	
1-2	86	86	No TSCRs	No TSCRs for this Page	N/A
1-3	0	0	No TSCRs	No TSCRs for this Page	N/A
1-4	34 TSCR 97-070	34 TSCR 97-070	IPN 97-070	Clarification of Containment Integrity	Incorporated
1-5	97	97	No TSCRs	No TSCRs for this Page	N/A
1-6	112 TSCR 98-018	112 TSCR 98-018	IPN 98-018	Generic Letter 89-01 and 10 CFR 20 Generic Letter	Incorporated
3.8-1	86		No TSCRs	No TSCRs for this Page	N/A
T 4.1-1(6)	181 TSCR 98-043	170 TSCR 97-118, 98-043	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

Add Note from 1.1 — A.1

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

~~1.1 REACTOR CONDITIONS~~ A.1

RTP

~~1.1.1~~ Rated Thermal Power (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3025 MWt. ~~("Rated Power" and "Rated Thermal Power" are used interchangeably throughout the Technical Specifications).~~ — A.1

THERMAL POWER

~~1.1.2~~ Thermal Power

Thermal Power shall be the total reactor core heat transfer rate to the reactor coolant.

~~1.1.3~~ ~~Reactor Pressure~~

~~The pressure in the steam space of the pressurizer.~~

~~1.1.4~~ ~~Tavg~~

~~Average temperature across the reactor vessel as measured by the hot and cold leg temperature detectors.~~ — A.2

~~1.2~~ REACTOR OPERATING CONDITIONS

Table 1.1-1 Mode 5

~~1.2.1~~ Cold Shutdown Condition

When the reactor is subcritical by at least 1% Δk/k and T_{avg} is ≤ 200°F.

keff < 0.99 — A.1

Table 1.1-1 Mode 3 Mode 4

~~1.2.2~~ Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is > 200°F but ≤ 555°F.

keff < 0.99 — A.4

Mode 3, Temp ≥ 350
Mode 4, Temp > 200 and < 350 — A.5

Add Definitions for the following:
Actions,
Axial Flux Difference; — A.3

A.1

Table 1.1-1 Mode 2

1.2.3 ~~Reactor Critical~~ Startup - Mode 2

When the neutron chain reaction is self-sustaining and k_{eff} ~~= 1.0~~. ≥ 0.99 A.6

Table 1.1-1 Mode 1

1.2.4 Power Operation Condition

Table 1.1-1, Note a A.7

When the reactor is critical and the ~~neutron flux power / range instrumentation indicates~~ greater than ~~2%~~ of rated power. 5% L.1

Table 1.1-1 Mode 6

1.2.5 Refueling Operation Condition

When the reactor is ~~subcritical by at least 5%~~ A.8 ~~ΔK/K~~ and ~~T_{avg} is ≤ 140°F~~ and ~~core alterations are being made~~ A.9 with the head completely ~~unbolted~~. Table 1.1-1, Notes b & c M.1

LA.1

~~1.3 REFUELING OUTAGE~~

~~An outage in which core alterations are performed in order to compensate for fuel burnup.~~ A.2

Core Alteration

~~1.4~~ CORE ALTERATION

Insert ITS Definition A.10

~~The addition, removal, relocation or other movement of fuel, controls, or installed equipment or material in a reactor core; except for functions normally performed during conventional reactor operation in accordance with intended design of equipment, such as control rod or instrument detector movement or performance of flux scans.~~ N.2

OPERABLE-OPERABILITY

~~1.5~~ OPERABLE

ITS SR 3.0.3 A.11

Properly installed in the system and capable of performing the intended functions in the intended manner ~~as verified by testing and tested at the frequency required by the Technical Specifications~~. Implicit in this definition shall be the assumption that all necessary attendant controls, electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Add Definitions for the following:
La;
Leakage;
Master Relay Test;
Mode;
Physics Tests; A.3

A.1

1.6 ~~OPERATING/INSERVICE~~

~~Performing the intended functions in the intended manner.~~ A.2

1.7 PROTECTION INSTRUMENTATION AND LOGIC A.12

The protection system consists of the actuation devices of both the reactor protection system and the engineered safety features systems.

1.7.1 Instrument Channel

An arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. An instrument channel loses its identity where single action signals are combined.

1.7.2 Logic Channel

A group of relay contact matrices which operate in response to the instrument channels signals to generate a protective action signal.

1.8 DEGREE OF REDUNDANCY A.2

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

~~1.9 INSTRUMENTATION SURVEILLANCE~~ A.1

~~1.9.1~~ Instrument Channel Check

CHANNEL CHECK

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel with other independent channels measuring the same variable.

Add Defenitions for the following:

PTLR;
Stave Relay Test;
Staggered Test Basis;
TADOT;

A.3

~~1.9.2~~ ~~Instrument Channel Functional Test~~ (or actual) — A.12

CHANNEL OPERATIONAL TEST (COT)

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating actions. (Add ITS Definition of COT) — A.13

~~1.9.3~~ Instrument Channel Calibration — A.14

CHANNEL CALIBRATION

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. (required) Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test. (Add ITS Definition) — A.14

~~1.9.4~~ ~~Logic Channel Functional Test~~

ACTUATION LOGIC TEST

The operation of relays or switch contacts, in all the combinations required, to produce the required output. — A.15

(Add ITS Definition)

1.10 CONTAINMENT INTEGRITY

Containment integrity is defined to exist when:

1.10.1 Non-automatic containment isolation valves (Table 3.6-1) are closed or may be opened under administrative control and only as long as necessary to perform their intended function.

1.10.2 Blind flanges, that provide an isolation function which are shown in FSAR drawings, are maintained installed.

1.10.3 Any test connection, vent or drain valve that is located within the isolation boundary and is required to perform an isolation function is closed and capped (threaded) or blind flanged as shown in FSAR drawings.

SEE ITS 3.6.1 3.6.2 3.6.3

1.10.4 The equipment door is properly closed.

1.10.5 Both doors in each personnel air lock are properly closed unless being used for entry, egress or maintenance, at which time at least one air lock door shall be closed.

1.10.6 All automatic containment isolation valves are either operable or in the closed position, or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

(Add ITS 1.2, Logical Connectors) — A.16

(Add ITS 1.3, Completion Times) — A.16

TSCR 97-070

(Add ITS 1.4, Frequency) — A.16

QPTR

~~1.11~~ QUADRANT POWER TILT RATIO

The quadrant power tilt ratio shall be the ratio of the maximum upper excore dector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

SEE ITS 3.2.4

1.12 SURVEILLANCE INTERVAL

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

SEE ITS 3.0

1.13 OPERATION IN A DEGRADED MODE

The plant is said to be operating in a degraded mode when it is operating with one or more systems listed herein inoperable as permitted by the Technical Specifications. If inoperable components or systems are subsequently made operable, the action statements requiring plant shutdown no longer apply.

$\bar{E}$

1.14 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY

~~Noble gas~~ $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 10 minutes, making up at least 95% of the total activity in the coolant. M.3

DOSE EQUIVALENT I-131

1.15 DOSE EQUIVALENT I-131

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

Add additional References L.2

~~1.16 REPORTABLE EVENT~~

~~A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR 50.~~ A.2

1.17 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6 5.6.5. Plant operation within these operating limits is addressed in individual specifications.

1.18 SHUTDOWN MARGIN A.20

SHUTDOWN MARGIN (SDM) is the instantaneous amount of negative reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn. add: RCCA not capable of insertion A.18

add: SDM: Part b-3 A.19

1.19 EFFLUENT CONCENTRATION

The EFFLUENT CONCENTRATION is that concentration of a radionuclide specified in 10 CFR 20, Table 2 of Appendix B.

1.20 MEMBER(S) OF THE PUBLIC

SEE ITS 5.5.1 ITS 5.5.4

MEMBER OF THE PUBLIC means any individual except when that individual is receiving an OCCUPATIONAL DOSE.

1.21 OCCUPATIONAL DOSE

OCCUPATIONAL DOSE means the dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. OCCUPATIONAL DOSE does not include dose received from background radiation, from any medical administration the individual has received, from exposure administered to individuals administered radioactive material and released in accordance with 35.75, from voluntary participation in medical research programs, or as a MEMBER OF THE PUBLIC.

1.22 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL SHALL contain the current methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Appendix A Technical Specification 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Appendix B Technical Specifications 4.3.2.1 and 4.3.2.2.

1.23 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way to assure compliance with 10 CFR Parts 20, 61 and 71, and Federal and State regulations and other requirements governing the disposal of solid radioactive waste.

Amendment No. ~~89~~, ~~103~~, ~~112~~,

TSCR 98-018

Table Notation

SEE ITS 3.0

*	By means of the movable incore detector system
**	Quarterly when reactor power is below the setpoint and prior to each startup if not done previous month.
***	This surveillance requirement may be extended on a one time basis to no later than April 26, 1997
****	This surveillance requirement may be extended on a one time basis to no later than May 12, 1997.
*****	This surveillance requirement may be extended on a one time basis to no later than May 14, 1997.
#	These requirements are applicable when specification 3.3.F.5 is in effect only.

TSCR 98-043

SEE ITS 3.4.1

##	The "each shift" frequency also requires verification that the DNB parameters (Reactor Coolant Temperature, Reactor Coolant Flow, and Pressurizer Pressure) are within the limits of Technical Specification 3.1.H.

S	- Each Shift (i.e., at least once per 12 hours)
W	- Weekly
P	- Prior to each startup if not done previous week
M	- Monthly
NA	- Not Applicable
Q	- Quarterly
D	- Daily
18M	- At least once per 18 months
TM	- At least every two months on a staggered test basis (i.e., one train per month)
24M	- At least once per 24 months
6M	- At least once per 6 months

A.17

TSCR 98-043

3.8 Refueling, Fuel Handling and Storage

~~**Applicability**~~

~~Applies to operating limitations during refueling, fuel handling, storage operations, and when heavy loads are moved over the reactor when the head is removed.~~

~~**Objective**~~

~~To ensure that no incident could occur during refueling, fuel handling, and storage operations that would adversely affect public health and safety.~~

Specification

SEE CTS MASTER MARKUP

A. During handling operations, reactor vessel head removal or installation, or movement of heavy loads over the reactor vessel with the head removed, the following conditions shall be met:

1. The equipment door and at least one door in each personnel air lock shall be properly closed. When the closure plate with a personnel door that prevents direct air flow from the containment is used, it shall be properly closed.

2. At least one isolation valve shall be operable, locked closed or blind flanged in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

3. Radiation levels in the containment and spent fuel storage areas shall be monitored continously.

4. The core subcritical neutron flux shall be continuously monitored by the two source range neutron monitors, ~~each with continuous~~ visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one source range neutron flux monitor shall be in service.

5. At least one residual heat removal pump and heat exchanger shall be operating except during those core alternations in which the residual heat removal flow interferes with component positioning.

Table 1.1-1, Mode 6

6. ~~During reactor vessel head removal and while loading and unloading fuel in the reactor, T_{avg} shall be < 140°F.~~ LA.1

7. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 1.0:
"USE AND APPLICATION"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 The definitions listed below are deleted because the current Technical Specifications that use these definitions are not retained in the ITS, or the equivalent ITS Specification will not use the defined term.

1.1.3	Reactor Pressure;
1.1.4	Tavg;
1.3	Refueling Outage;
1.6	Operating/Inservice;
1.7	Protection Instrumentation and Logic;
1.8	Degree of Redundancy; and,
1.16	Reportable Event.

The removal of definitions that are not used in the ITS is an administrative change with no impact on safety because there is no effect on any Technical Specifications retained in the ITS. Descriptions and justifications for changing the CTS that use these definitions are included elsewhere in the IP3 ITS conversion package.

A.3 The IP3 Improved Technical Specifications use the following defined terms that are not part of the CTS:

Actions	Physics Tests
Trip Actuating Device Operational Test (TADOT)	Pressure and Temperature Limits Report (PTLR)
Mode	Slave Relay Test
La	Staggered Test Basis
Leakage	Master Relay Test
Axial Flux Difference (AFD)	

Although changes in definitions could change Technical Specification requirements, adding definitions used in ITS LCOs or Surveillances but not used in the CTS is an administrative change because any technical changes to existing definitions created by the adoption of these definitions are identified and justified with the applicable LCOs. Therefore, this is an administrative change with no significant adverse impact on safety.

A.4 The CTS 1.2.2 definition of Hot Shutdown Condition specifies that the reactor is in Hot Shutdown only if the reactor is subcritical by an amount greater than or equal to the margin specified in CTS 3.10 (i.e., Mode 3 and/or Mode 4). Therefore, consistent with CTS 3.10.1.1 requirements, the reactor must have shutdown margin (SDM) ≥ 1.3% $\Delta k/k$ to be considered in Mode 3 and/or Mode 4.

The equivalent ITS definition in ITS Table 1.1-1 specifies the reactivity condition needed to be in Mode 3 and/or 4 is Keff < 0.99; however, ITS LCO 3.1.1, Shutdown Margin, maintains the requirement to have SDM ≥ 1.3% $\Delta k/k$ when in Modes 3 and 4, otherwise Required Actions must be implemented (See ITS LCO 3.1.1).

The difference between the CTS definition and the ITS definition is the core reactivity level that defines the transition between Mode 2 (CTS 1.2.3 definition of Reactor Critical) and Mode 3 (CTS 1.2.2 definition of Hot Shutdown Condition). CTS defines the transition as the point

when the reactor is subcritical by less than the required SDM of 1.3% $\Delta k/k$. ITS defines the transition as when the reactor is subcritical by less than 1.0% $\Delta k/k$ with Limiting Condition for Operation (LCO) 3.1.1, Shutdown Margin, which maintains the requirement to have SDM ≥ 1.3% $\Delta k/k$ when in Modes 3 and 4 and in Mode 2 when Keff is < .99.

Eliminating the required SDM from the definition of the Mode is needed to differentiate between the status of the reactor (i.e., reactor subcritical (Keff < 0.99)) and an associated Limiting Condition for Operation that the SDM must be ≥ 1.3% k/k (i.e., a parameter assumed as an initial condition for the analysis of reactivity addition events in Modes 3 and 4). This change makes IP3 consistent with industry practice of considering the reactor subcritical when Keff < 0.99 with SDM requirements governed by an LCO.

This is an administrative change with no significant adverse impact on safety because ITS LCO 3.1.1, Shutdown Margin, maintains the requirement to have SDM ≥ 1.3% $\Delta k/k$ when in Modes 3 and 4.

A.5 The CTS 1.2.2 definition of Hot Shutdown Condition specifies that the reactor is in Hot Shutdown only if the reactor is subcritical (See ITS 1.0, DOC A.4) and reactor coolant average temperature (Tavg) is greater than 200°F and less than or equal to 555°F. However, CTS Limiting Conditions for Operation (LCOs) and associated Actions frequently differentiate between Hot Shutdown when > 350°F and Hot Shutdown < 350°F.

The equivalent ITS definition in ITS Table 1.1-1 provides formal recognition of the CTS practice of differentiating between "Hot Standby" when > 350°F (Mode 3) and "Hot Shutdown" when < 350°F (Mode 4). Differentiating between Hot Standby (Mode 3) and Hot Shutdown (Mode 4) does not change any existing requirements. Changes to the Applicability of LCOs during the conversion from CTS to ITS are identified and justified with the applicable LCOs. Therefore, this is an administrative change with no significant adverse impact on safety.

A.6 The CTS 1.2.3 definition of Reactor Critical specifies that the reactor is critical when the neutron chain reaction is self-sustaining and keff = 1.0. However, CTS LCOs that are Applicable when the reactor is

critical are considered Applicable during approaches to criticality and power level changes.

The equivalent ITS definition in ITS Table 1.1-1 identifies the same condition as "Startup" (i.e., Mode 2) and specifies the reactivity condition that places the reactor in Mode 2 is Keff ≥ 0.99. This change makes IP3 consistent with industry practice of considering the reactor critical during approaches to criticality when Keff is ≥ 0.99.

This is an administrative change with no significant adverse impact on safety because the ITS reactivity limits for Mode 2 are consistent with a reasonable interpretation of the existing definition.

A.7 The CTS 1.2.4 definition of Power Operation Condition specifies that neutron flux power range instrumentation is used to determine when reactor power is high enough such that the reactor is considered in the Power Operation Condition (i.e., Mode 1).

The equivalent ITS definition in ITS Table 1.1-1 maintains this clarification that neutron flux power range instrumentation is used to determine when reactor power is high enough such that the reactor is considered in Mode 1. Specifically, footnote (a) specifies that decay heat is excluded when determining if the reactor power level places the reactor in Mode 1 (i.e., neutron flux power range instrumentation is used without adjustment for decay heat variations related to recent power level changes). This is an administrative change with no significant adverse impact on safety because there is no change to the existing requirement.

A.8 The CTS 1.2.5 definition of Refueling Operation Condition (i.e., Mode 6) specifies that the reactor is in refueling when it is subcritical by at least 5% $\Delta k/k$.

The equivalent ITS definition in ITS Table 1.1-1 does not specify a reactivity condition needed to be in refueling (i.e., Mode 6); however, ITS LCO 3.9.1, requires that boron concentration in the RCS during refueling must maintain keff ≤ 0.95 (i.e., subcritical by 5% $\Delta k/k$) during fuel handling.

This change is needed to differentiate between the actual condition of

the reactor (i.e., reactor vessel head removed) and an associated Limiting Condition for Operation (i.e., SDM must be ≥ 5.0% $\Delta k/k$).

This is an administrative change with no significant adverse impact on safety because ITS LCO 3.9.1 maintains the requirement that boron concentrations in the RCS during refueling maintain an overall core reactivity of keff ≤ 0.95 during fuel handling. Therefore, there is no change to the existing requirement.

A.9 The CTS 1.2.5 definition of Refueling Operation Condition (i.e., Mode 6) specifies that the refueling condition exists only when core alterations are being made; however, CTS LCOs and Required Actions differentiate between refueling conditions with and without core alterations in progress.

The equivalent ITS definition in ITS Table 1.1-1 specifies that the refueling condition exists whenever one or more reactor vessel head closure bolts is less than fully tensioned without regard to the status of core alterations; however, similar to the CTS, ITS LCO Applicability statements and Required Actions differentiate between refueling conditions with and without core alterations in progress.

Deletion of the condition that core alterations are in progress from the definition of refueling that is an administrative change because any technical changes to existing requirements are identified and justified with the applicable LCOs. Therefore, this is an administrative change with no significant adverse impact on safety.

A.10 The CTS 1.4 definition of Core Alterations identifies the various activities classified as Core Alterations because there are LCOs and Required Actions that apply only during Core Alterations.

The ITS definition of a Core Alteration also identifies the activities classified as Core Alterations; however, the ITS definition includes the guidance that Required Actions to suspend core alterations do not preclude completion of movement of a component to a safe position.

This is an administrative change with no significant adverse impact on safety because it is a reasonable interpretation of the existing definition.

A.11 The CTS 1.5 definition of Operable specifies that Operability must be verified by testing and testing must be performed at the frequency required by the Technical Specifications.

The equivalent ITS definition for Operable-Operability does not specify that Operability must be verified by testing at a specified Frequency; however, this CTS requirement is maintained in ITS SR 3.0.3. Therefore, this is an administrative change with no significant adverse impact on safety.

A.12 The CTS 1.9.2 definition of Instrument Channel Functional Test requires injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating actions.

The equivalent ITS Definition, Channel Operation Test, maintains this requirement except that the ITS definition allows use of either an actual or simulated signal to verify that a channel is Operable. This change is acceptable because use of an actual instead of a simulated or "test" signal will not affect the performance of the test because the equipment being tested cannot discriminate between an actual or simulated signal. This is an administrative change with no impact on safety because the use of an actual or simulated signal does not change the validity of the test as a verification of plant response to the event.

A.13 The CTS 1.9.2 definition of Instrument Channel Functional Test requires injection of a signal into the channel to verify that it is operable, including alarm and/or trip initiating actions.

The equivalent ITS Definition, Channel Operation Test (COT), maintains this requirement except as follows:

The ITS Definition of COT clarifies that the signal must be injected into the channel as close to the sensor as practicable;

The ITS Definition of COT clarifies that only required alarms, interlocks, displays, and trip functions must be verified Operable;

The ITS Definition of COT clarifies that the COT must include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

These are administrative changes with no significant adverse impact on safety because each is a reasonable interpretation of the existing definition.

A.14 The CTS 1.9.3 definition of Instrument Channel Calibration requires adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter that the channel measures. CTS 1.9.3 also specifies that calibration must encompass the entire channel, including alarm or trip, and must include the channel functional test.

The equivalent ITS Definition, Channel Calibration, maintains these requirements except for the following:

The ITS Definition of Channel Calibration clarifies that only required alarms, interlocks, displays, and trip functions must be verified Operable;

The ITS Definition of Channel Calibration clarifies that a calibration may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated;

The ITS Definition of Channel Calibration clarifies that calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibrations of the remaining adjustable devices in the channel.

The ITS Definition of Channel Calibration clarifies that whenever a sensing element is replaced, the next required Channel Calibration must include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

These are administrative changes with no significant adverse impact on safety because each is a reasonable interpretation of the existing definition.

A.15 The CTS 1.9.4 definition of Logic Channel Functional requires operation of relays or switch contacts, in all the combinations required, to produce the required output.

The equivalent ITS Definition, Actuation Logic Test, maintains these requirements except for the following:

The ITS Definition of Actuation Logic Test clarifies that the verification of the required logic output must include each possible interlock logic state; and

The ITS Definition of Actuation Logic Test clarifies that the test must include at least a continuity check of output devices.

These are administrative changes with no significant impact on safety because each is a reasonable interpretation of the existing definition.

A.16 The ITS includes the following sections that are not included in the CTS: Section 1.2 - Logical Connectors; Section 1.3 - Completion Times; and, Section 1.4 - Frequency.

The reason for these additional sections is that some conventions in the application of Technical Specifications to unusual situations have been the subject of debate and different interpretations between the licensees and the NRC Staff. Because the guidance in these proposed sections establishes positions not previously documented, the guidance is considered administrative, with no impact on plant safety. These sections are consistent with the Westinghouse STS, NUREG-1431, Rev. 1.

A.17 CTS Table 4.1-1 (Sheet 6 of 6) includes abbreviations for Surveillance Requirement (SR) Frequencies used in the CTS. ITS SRs do not use these abbreviations to identify SR Frequencies. Therefore, deletion of this list of abbreviations is an administrative change with no significant adverse impact on safety.

A.18 CTS 1.18 is the definition of Shutdown Margin. The ITS maintain the CTS definition (See ITS 1.0, DOCs A.19 and A.20) but includes the clarification that if any RCCA is not capable of being fully inserted, then the reactivity worth of that RCCA must be accounted for in the determination of SDM. This is an administrative change with no significant adverse impact on safety because the clarification provided in the ITS definition is consistent with requirements in CTS 3.10.7 and CTS 3.10.4 for operation with an inoperable control rod (See ITS 3.1.5 and 3.1.6).

A.19 CTS 1.18 is the definition of Shutdown Margin. The ITS maintain the CTS definition (See ITS 1.0, DOCs A.18 and A.20) but includes the clarification that if in Modes 1 and 2, then the fuel and moderator temperatures are assumed to be at the hot zero power level. This is consistent with current assumption and practice. This is an administrative change with no significant adverse impact on safety because it is a reasonable interpretation of the existing requirement.

A.20 The CTS 1.18 definition of Shutdown Margin specifies that SDM is based on the reactivity worth of all "full length" rod cluster assemblies (i.e., an implicit requirement that the reactivity worth of partial length rods may not be credited). The ITS definition of SDM does not include a requirement that the reactivity worth of partial length rods may not be credited. This is an administrative change with no significant adverse impact on safety because the IP3 design does not use partial length control rod assemblies.

MORE RESTRICTIVE

M.1 The CTS 1.2.5 definition of Refueling Operation Condition (i.e., Mode 6) specifies that the refueling condition exists only when the vessel head is completely unbolted. The equivalent ITS definition in ITS Table 1.1-1 specifies that the refueling condition exists when one or more reactor vessel head closure bolts less than fully tensioned.

This change is needed because assumptions regarding the RCS boundary may not be met when one or more bolts are detensioned. This change is acceptable because ITS LCO Applicability statements carefully

differentiate between LCOs that apply during refueling mode (i.e., one or more reactor vessel head closure bolts less than fully tensioned) and those that apply whenever the reactor vessel head is seated on the vessel (e.g., Low Temperature Overpressure Protection). Any technical changes to existing requirements are identified and justified with the applicable LCOs. Therefore, this change has no significant adverse impact on safety.

M.2 The CTS 1.4 definition of Core Alterations does not classify any functions normally performed during conventional reactor operation in accordance with intended design of equipment (e.g., control rod movement) as a core alteration.

The ITS Definition, Core Alteration, includes the movement of any fuel, sources, or reactivity control components; therefore, control rod movement is a core alteration under ITS.

This change is needed because the term Core Alteration is used to describe conditions in Mode 6 when reactivity excursions and/or fuel handling accidents are more likely to occur. Therefore, control rod movement is classified as a core alteration because of the potential to add positive reactivity.

This change is acceptable because preventing positive reactivity insertion by control rod movement when core alterations are prohibited is conservative.

M.3 CTS 1.14, the definition of $\bar{E}$-Average Disintegration Energy, limits $\bar{E}$ to noble gases with half-lives greater than 10 minutes when calculating the acceptance criteria for reactor coolant gross activity in CTS 3.1.D.1.

ITS LCO 3.4.16 and the acceptance criteria for ITS 3.4.16.1 are based on the ITS Definition, $\bar{E}$-Average Disintegration Energy. The ITS Definition of $\bar{E}$ differs from the CTS definition in that the ITS definition includes all isotopes (not just Noble gases) in the reactor coolant, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant. This change, including all isotopes except iodines when measuring gross specific activity, is needed because the ITS definition ensures that contributions from

isotopes other than Noble gases, although typically not significant, are counted. (Maintaining the CTS allowance permitting the exclusion of isotopes with half lives > 10 minutes rather than adopting the ITS allowance permitting the exclusion of isotopes with half lives > 15 minutes is needed to ensure that Xenon-138 is included in $\bar{E}$-Average Disintegration Energy consistent with current analysis assumptions.) This change, excluding iodines from the definition of $\bar{E}$ and gross specific activity, is acceptable because the dose contribution of iodines are limited by the ITS SR 3.4.16.2 limits for Dose Equivalent I-131. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 The CTS 1.2.4 definition of the Power Operation Condition specifies that the reactor is in the Power Operation Condition (i.e., Mode 1) whenever reactor power is > 2% of rated power. Therefore, CTS LCOs applicable in the Power Operation Condition are applicable whenever reactor power is > 2% of RTP.

The equivalent definition in ITS Table 1.1-1 specifies that the reactor is in the Power Operation Condition (i.e., Mode 1) whenever reactor power is > 5% of rated power. Therefore, ITS LCOs applicable in the Power Operation Condition are applicable whenever reactor power is > 5% of rated power. This change increases the power level used as the transition point from Mode 2 to Mode 1 from 2% RTP to 5% RTP and, therefore, increases the power level at which Mode 1 LCOs become applicable.

This change is needed to provide a wider band of operation in Mode 2 (i.e., 0% RTP t0 5% RTP) before Mode 1 LCOs become applicable. This change is acceptable because there are no analyzed events where assumptions of initial reactor power or the Operability of equipment required in Mode 1 but not Mode 2 are affected by this change in the Mode 1/Mode 2 transition point from 2% to 5% RTP. Therefore, this change has no significant adverse impact on safety.

L.2 The CTS 1.15 definition of Dose Equivalent I-131 defines this term as that concentration of I-131 (micro curies/gram) that alone would produce

the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 present. The thyroid dose conversion factors used for this calculation must be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

The equivalent ITS Definition, Dose Equivalent I-131, maintains this definition except that the ITS definition specifies that thyroid dose conversion factors from either of the following references are also acceptable:

Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or,

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

This change is needed because it allows more recently approved standards to be used for future calculations while documenting that existing calculations based on conversion factors in TID 14844 are still acceptable. This is a less restrictive change because the more recently approved conversion factors may result in a conclusion that a lower dose will result from a given release of radioactive iodine.

This change is acceptable because all three references identified in the ITS Definition contain thyroid dose conversion factors determined to be acceptable for use in calculation of the Dose Equivalent I-131.

REMOVED DETAIL

LA.1 The CTS 1.2.5 definition of Refueling Operation Condition (i.e., Mode 6) specifies that the reactor is in the refueling condition only when specified conditions including Tavg ≤ 140°F exist. However, no reactor condition is defined or Actions specified if Tavg is > 140°F but the reactor is otherwise in the refueling condition.

The equivalent definition in ITS Table 1.1-1 specifies that the refueling condition exists whenever one or more reactor vessel head closure bolts are less than fully tensioned whatever the RCS temperature. The requirement to maintain Tavg ≤ 140°F whenever the

reactor is in Mode 6 is not included in the ITS and is relocated to the FSAR and plant procedures.

This change is needed to differentiate between the condition of the reactor (i.e., reactor vessel head closure bolts less than fully tensioned) and a Technical Requirement (Tavg ≤ 140°F) associated with this condition.

Relocating to the FSAR (and plant procedures) the requirement to maintain Tavg ≤ 140°F when in Mode 6 is acceptable because it is not an initial condition for an accident analysis that assumes the failure of or presents a challenge to the integrity of a fission product barrier. Specifically, consistent with Regulatory Guide 1.25, March 23, 1972, the fuel handling accident analysis assumes a decontamination factor of 100 whatever water temperature based solely on the 23-foot water level in the refueling canal and the refueling cavity. Additionally, Tavg ≤ 140°F is not a limiting assumption for the detection or mitigation of a boron dilution event.

This change is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, are designed to assure that changes to the FSAR do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because Tavg ≤ 140°F is not a limiting initial condition for either event postulated to occur in Mode 6: a fuel handling event or a boron dilution event. This change is acceptable because an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 1.0:
"USE AND APPLICATION"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 1.0 - USE AND APPLICATION

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change increases the power level used as the transition point from Mode 2 to Mode 1 from 2% RTP to 5% RTP and, therefore, increases the power level at which Mode 1 LCOs become applicable. This change is needed to provide a wider band of operation in Mode 2 (i.e., 0% RTP to 5% RTP) before Mode 1 LCOs become applicable.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because there are no analyzed events where assumptions of initial reactor power or the Operability of equipment required in Mode 1 but not Mode 2 are affected by this change in the Mode 1/Mode 2 transition point from 2% to 5% RTP. Therefore, this change has no significant adverse impact on safety.

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

   The proposed change does not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This proposed change does not involve a significant reduction in a margin of safety because there are no analyzed events where assumptions

of initial reactor power or the Operability of equipment required in Mode 1 but not Mode 2 are affected by this change in the Mode 1/Mode 2 transition point from 2% to 5% RTP. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The ITS definition for Dose Equivalent I-131 would allow the use of the following references, in addition to Table III of TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," for thyroid dose conversion factors: Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, and ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity." All three references are acceptable. By including all three documents, more recent standards can be used for future calculations, while existing calculations based on conversion factors in TID 14844 are still acceptable.

   This change does not involve a significant increase in the probability or consequences of an accident previously evaluated, since all three references are approved methods for selecting dose conversion factors for calculating Dose Equivalent I-131. As such, these changes do not affect initiators of analyzed events or alter any assumptions concerning mitigation of accident or transient events.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC), nor involve changes in normal plant operation. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because all three references are approved methods for selecting dose conversion factors for calculating Dose Equivalent I-131.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 1.0:
"USE AND APPLICATION"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 1.0

This ITS Specification is based on NUREG-1431 Specification No. **1.0**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
BWOG-002 R1	**039 R1**	ALLOW CFTS TO BE PERFORMED BY SEQUENTIAL, OVERLAPPING OR TOTAL CHANNEL STEPS	Rejected by NRC	Not Incorporated	**N/A**
BWROG-038	**205 R0**	REVISION OF CHANNEL CALIBRATION, CHANNEL FUNCTIONAL TEST, AND RELATED DEFINITIONS	NRC Review	Not Incorporated	**N/A**
CEOG-016	**064**	CLARIFICATION OF APPLICABILITY OF CHANNEL CALIBRATION AND CHANNEL FUNCTIONAL TEST	NRC Reviewer Rejects	Not Incorporated	**N/A**
WOG-001.2	**003 R1**	RELOCATE REFERENCES TO THYROID DOSE CONVERSION FACTORS TO THE BASES.	Rejected by NRC	Not Incorporated	**N/A**

Technical Specification 1.0:
"USE AND APPLICATION"

WOG-001.3 R1	**004 R1**	MOVE THE PORV LIFT SETTINGS AND THE ENABLE TEMPERATURE TO THE PTLR	Rejected by NRC	Not Incorporated	**N/A**
WOG-005		ELIMINATE "REQUIRED DISPLAY" FROM THE CHANNEL CALIBRATION DEFINITION	Rejected by TSTF	Not Incorporated	**N/A**
WOG-006 R1	**019 R1**	RELOCATE THE DETAILS OF RTD AND THERMOCOUPLE CALIBRATION FROM THE CHANNEL CALIBRATION DEFINITION TO BASES OF INST. SPECS	Approved by NRC	Not Incorporated	**N/A**
WOG-042	**052**	IMPLEMENT 10 CFR 50, APPENDIX J, OPTION B	TSTF to Rewrite	Not Incorporated	**N/A**
WOG-047	**088 R0**	NUMBER OF REQUIRED REACTOR VESSEL HEAD CLOSURE BOLTS	Rejected by NRC	Not Incorporated	**N/A**
WOG-052 R1	**111 R1**	REVISE BASES FOR SR 3.3.1.16 AND 3.3.2.10 TO ELIMINATE PRESSURE SENSOR RESPONSE TIME TESTING	NRC Review	Not Incorporated	**N/A**
WOG-067 R1		RELOCATE LTOP ARMING TEMPERATURE TO PTLR	Rejected by TSTF	Not Incorporated	**N/A**
WOG-074 R2		ADD "ONLY REQUIRED TO BE PERFORMED" EXAMPLE TO 1.4	Rejected by TSTF	Not Incorporated	**N/A**
WOG-090		ADD A SECTION 1.4 EXAMPLE OF FREQUENCY BASED ON PLANT CONDITIONS	TSTF Review	Not Incorporated	**N/A**
WOG-110		REVISE EXAMPLE 1.4-2 TO REFERENCE SR 3.0.3	TSTF Review	Not Incorporated	**N/A**

1.0 USE AND APPLICATION

<CTS>

1.1 Definitions

NOTE

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

Term	Definition
<DOC A.3> ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
<1.9.4> <DOC A.15> ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
<DOC A.3> AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
<1.9.3> <DOC A.14> CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

(continued)

<CTS> 1.1 Definitions (continued)

<1.9.1>	CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
<1.9.2> <DOC A.12> <DOC A.13>	CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
<1.4> <DOC M.2> <DOC A.10>	CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
<1.17>	CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
<1.15> <DOC L.2>	DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page

(continued)

<CTS> 1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".
<1.14> $\bar{E}$—AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant. 10
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.) Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. CLB.2
<DOC A.3> L_a	The maximum allowable primary containment leakage rate, L_a, shall be 1% of primary containment air weight per day at the calculated peak containment pressure (P_a).
<DOC A.3> LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

Insert: 1.1-3-01

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

Insert: 1.1-3-02

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

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 1.0 - USE AND APPLICATION

INSERT: 1.1-3-01:

for leakage into closed systems and CLB.1

INSERT: 1.1-3-02:

CLB.1

(Leakage into closed systems is leakage that can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and leakage past the safety injection pressure isolation valves are examples of reactor coolant system leakage into closed systems.)

1.1 Definitions

<DOC A.3>	LEAKAGE (continued)	systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System; b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; (CLB.1) for leakage into closed systems and c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
<DOC A.3>	MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.
<DOC A.3>	MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. (PA.1) loop
<1.5> <DOC A.11>	OPERABLE—OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
<DOC A.3>	PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

(continued)

13, Initial Tests and Operations

FSAR

<CTS> 1.1 Definitions

<DOC A.3> PHYSICS TESTS (continued)

a. Described in Chapter ~~[14. Initial Test~~ Program] of the FSAR;

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

c. Otherwise approved by the Nuclear Regulatory Commission.

<DOC A.3> PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

<1.11> QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

<1.1.1> RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt. 3025

~~REACTOR TRIP SYSTEM (RTS) RESPONSE TIME~~

~~The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~

CLB.1

<1.18> SHUTDOWN MARGIN (SDM)

<DOC A.18>
<DOC A.19>
<DOC A.20>

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

(continued)

<CTS> 1.1 Definitions

<1.18>	SHUTDOWN MARGIN (SDM) (continued)	a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level. (not zero power level)
<Doc A.3>	SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
<Doc A.3>	STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
<1.1.2>	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
<Doc A.3>	TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1 (page 1 of 1)
MODES

	MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
<1.2.4, DOC A. , A. >	1	Power Operation	≥ 0.99	> 5	NA
<1.2.3, DOC A.6>	2	Startup	≥ 0.99	≤ 5	NA
<1.2.2, DOC A.4, A.5>	3	Hot Standby	< 0.99	NA	≥ ~~[350]~~
<1.2.2, DOC A.4, A.5>	4	Hot Shutdown[b]	< 0.99	NA	~~[350]~~ > T_{avg} > ~~[200]~~
<1.2.1>	5	Cold Shutdown[b]	< 0.99	NA	≤ ~~[200]~~
<1.2.5, DOC LA1, DOC A.8, A.9>	6	Refueling[c]	NA	NA	NA

<1.2.4> (a) Excluding decay heat.

<1.2.5> <DOC M.1> (b) All reactor vessel head closure bolts fully tensioned.

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

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

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

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

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

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

(continued)

1.3 Completion Times

DESCRIPTION (continued)

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

a. Must exist concurrent with the first inoperability; and

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

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

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

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

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time

(continued)

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------NOTE---------- Not required to be performed until 12 hours after ≥ 25% RTP. ----------	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 1.0:
"USE AND APPLICATION"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 1.0 - USE AND APPLICATION

1RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 IP3 ITS definition of Leakage differs from NUREG-1431 by maintaining the allowance in CTS 3.1.F.2 and CTS 3.1.F.3 that limits for unidentified and total (unidentified and identified) RCS Leakage do not apply to controlled leakage sources such as the reactor coolant pump controlled leakage seals and leakage into closed systems. Therefore, the definition of Leakage in ITS 1.0 defines leakage so that ITS LCO 3.4.13 limits are not applicable to controlled leakage sources such as the reactor coolant pump controlled leakage seals and leakage into closed systems. This change maintains the current licensing basis.

CLB.2 The definitions of Reactor Trip System (RTS) Response Time and Engineered Safety Feature (ESF) Response Time are not incorporated into the ITS because such response time testing is not part of the current licensing basis.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 IP3 ITS 1.0 clarifies the definition of MODE by adding the word "loop" to clarify that MODE changes are based on average reactor coolant "loop" temperature. This clarifies that determination of Mode is based on indicated loop temperature and does not include calculating the impact of the temperature and volume of water in the pressurizer.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

None

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 1.0 - USE AND APPLICATION

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 2.0:
"SAFETY LIMITS (SLs)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Vessel inlet temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, 5, and in MODE 6 when the reactor vessel head is on, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6, restore compliance within 5 minutes.

This curve does not provide allowable limits for normal operation.
(see Pressure, Temperature and Flow DNB limits, for DNB limits)

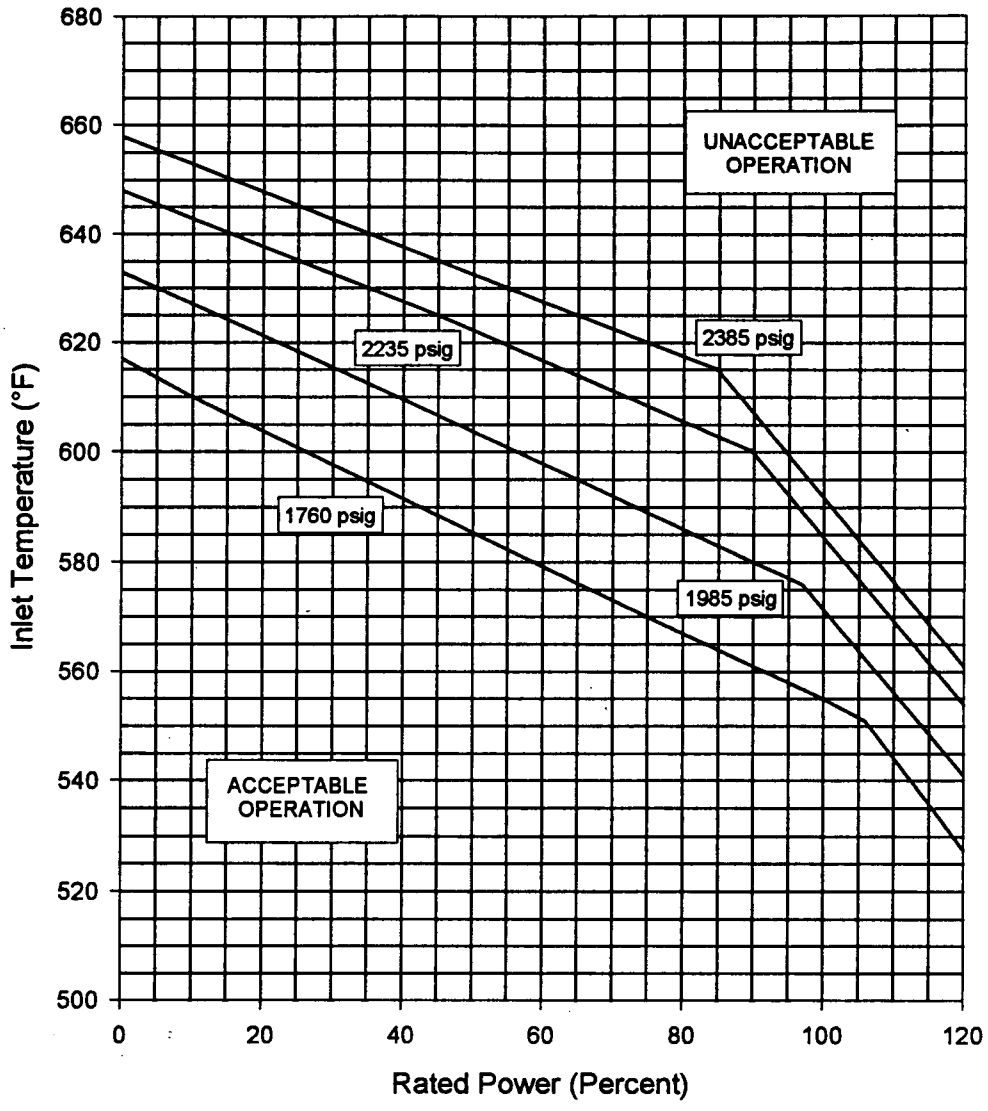


Figure 2.1-1
Rated Power (Percent of 3025 MWt)
100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt
Pressures and temperatures do not include allowance for instrument error.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

BASES

BACKGROUND (Continued)

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Protection System (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip;

d. Overpower ΔT trip;

e. Power Range Neutron Flux trip; and

BASES

APPLICABLE SAFETY ANALYSES (continued)

f. Steam generator safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits assumes:

1. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at RTP specified in the COLR;

2. An equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24% (Ref. 3);

3. Reactor coolant system total flow rate of greater than or equal to 375,600 gpm as measured at the plant; and,

4. A reference cosine with a peak of 1.55 for axial power shape.

BASES

SAFETY LIMITS (continued)

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP}(1 + PF_{\Delta H}(1-P))$$

Where
P is the fraction of Rated Thermal Power;
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at RTP specified in the COLR; and,
$PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.9% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (specified in the COLR) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

BASES

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A.

2. FSAR, Section 7.2.

3. WCAP-10705, Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators, October 1984.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2485 psig. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

BASES

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Protection System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer Power Operated Relief Valves (PORVs);

b. Atmospheric Dump Valves;

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer Spray.

BASES

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 6) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 and in MODE 6 when the reactor vessel head is on because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 when reactor vessel head is removed because the RCS can not be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

BASES

REFERENCES

1. 10 CFR 50, Appendix A.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.50433

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR 100.

5. FSAR, Section 7.2.

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 2.0:
"SAFETY LIMITS (SLs)"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
2.1-1	175	175	No TSCRs	No TSCRs for this Page	N/A
2.1-2	175 TSCR 98-036	175 TSCR 98-036	IPN 98-036	Steam Generator Tube Plugginhg Limit	
2.1-3	103 TSCR 98-036	103 TSCR 98-036	IPN 98-036	Steam Generator Tube Plugginhg Limit	
F 2.1-1	175	175	No TSCRs	No TSCRs for this Page	N/A
F 2.1-2	48	48	No TSCRs	No TSCRs for this Page	N/A
2.2-1	0	0	No TSCRs	No TSCRs for this Page	N/A
2.2-2	0	0	No TSCRs	No TSCRs for this Page	N/A
6-13	163 TSCR 97-051, 98-018	163 TSCR 97-051, 98-018	IPN 98-018	Generic Letter 89-01 and 10 CFR 20 Generic Letter	Incorporated
6-13	163 TSCR 97-051, 98-018	163 TSCR 97-051, 98-018	IPN 97-051	SRC Audit Requirements and Management Title Changes (See IPN 97-144)	Incorporated

2.0 Safety Limits and Limiting Safety System Settings

2.1 Safety Limits, Reactor Core

A.2

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure and coolant temperature during four-loop operation.

Objective

In Mode 1 and 2 A.5

To maintain the integrity of the fuel cladding.

Specification

A.3

pressure vessel inlet

2.1.1 The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 for four-loop operation. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System vessel inlet temperature and power level is at any time above the appropriate pressure line. A.4

Basis

A.1

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. The safety limits represent a design requirement for establishing the trip setpoints identified in Technical Specification 2.3. Technical Specification 3.1.H, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," provide more restrictive limits to ensure that the safety limits are not exceeded.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant Temperature and Pressure have been related to DNB through correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: There must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I (normal operation and operational transients) and Condition II (events of moderate frequency) events is greater than or equal to the Design DNBR limit.

A.1

In meeting the DNB criterion, uncertainties in operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analyses limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits includes:

1. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at Rated Thermal Power (RTP) specified in the COLR.

2.** an equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24%, [2]

3. a reactor coolant system total flow rate of greater than or equal to 375,600 gpm as measured at the plant,

4. a reference cosine with a peak of 1.55 for axial power shape.

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} \left(1 + PF_{\Delta H} \left(1-P\right)\right)$$

Where P is the fraction of Rated Thermal Power.

$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.9% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

** A lower SG plugging level of 2% is presumed for the Loss of Normal Feedwater and Loss of AC Power analyses. A reduction in assumed SG tube plugging levels makes the curves in Figure 2.1-1 more conservative (i.e., adds greater margin).

TSCR 98-036

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (specified in the COLR) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

References

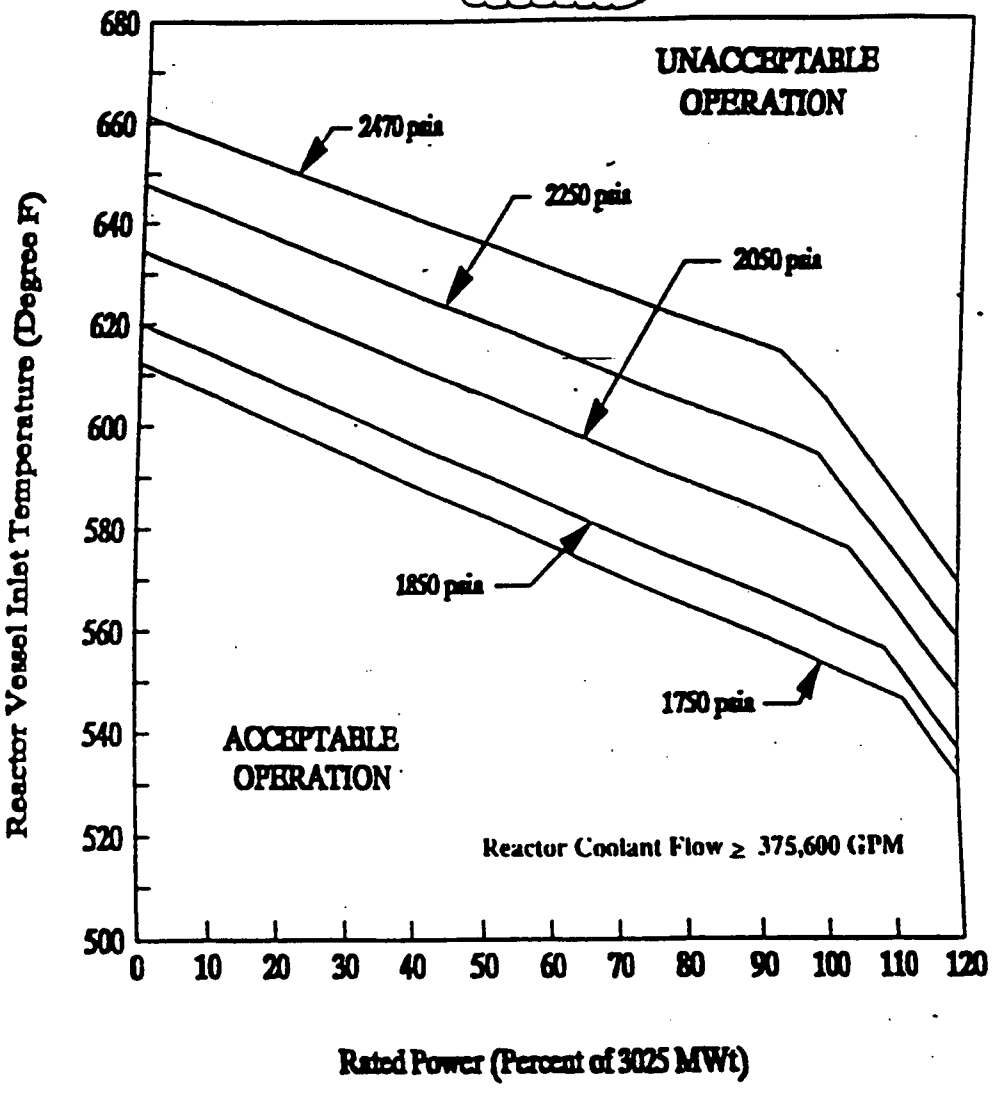
1. FSAR Section 3.2.2

2. "Safety Evaluation for Indian Point Unit 3 with Asymmetric Tube Plugging Among Steam Generators", WCAP-10705 (Westinghouse Non-Proprietary), October 1984.

TSCR 98-036

REACTOR CORE SAFETY LIMITS

This curve does not provide allowable limits for normal operation.
(See ~~Technical Specification 3.1.H~~ LCO 3.4.1 for DNB limits)



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt

Pressures and temperatures do not include allowance for instrument error.

FIGURE ~~2.1-1~~ 2.1.1-1

DELETED

A.1

Figure 2.1-2. Core Limits - Three Loop Operation

2.1.2 ~~2.2~~ SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System pressure.

A.2

Objective

To maintain the integrity of the Reactor Coolant System and to prevent the release of excessive amounts of fission product activity to the containment.

Specification

in Modes 1, 2, 3, 4, 5 and 6 with vessel head on — A.6

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis

The Reactor Coolant System (1) serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant system is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.

A.1

A.1

The setting of the power operated relief valves (2335 psig)[2] and the reactor high pressure trip (2385 psig)[2] have been established to assure that the Reactor Coolant System pressure limit is never reached and that the system pressure does not exceed the design limits of the fuel cladding.

In addition, the Reactor Coolant System safety valves[3] are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, the system is hydrotested in accordance with Power Piping Code USAS B31.1 (1967) prior to initial operation.

References

(1) FSAR Section 4

(2) FSAR Table 4.1-1

(3) FSAR Section 4.3.4

SEE ITS 5.1

b. Each REPORTABLE EVENT shall be reviewed by the PORC and a report submitted by the Site Executive Officer to the Chief Nuclear Officer, Vice President Regulatory Affairs and Special Projects, and the Chairman of the SRC.

~~6.7~~ SAFETY LIMIT VIOLATION

Add ITS 2.2.1
Add ITS 2.2.2

M.1

SL 2.2 ~~6.7.1~~ The following actions shall be taken in the event a Safety Limit is violated:

a. The reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36(c)(1)(i).

LA.1

b. The Safety Limit Violation shall be reported immediately to the Commission. The Chief Nuclear Officer, Vice President Regulatory Affairs and Special Projects, and the Chairman of the SRC will be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared by the PORC. This report shall describe (1) applicable circumstances preceding the occurrences, (2) effects of the occurrence upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

LA.1

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chief Nuclear Officer, the Vice President Regulatory Affairs and Special Projects, and the Chairman of the SRC by the Site Executive Officer.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

SEE ITS 5.4
ITS 5.5.3

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Process Control Program implementation.

g. Offsite Dose Calculation Manual implementation.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 2.0:
"SAFETY LIMITS (SLs)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 2.0 - SAFETY LIMITS (SLs)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 2.1 establishes a safety limit that the combination of thermal power, coolant pressure, and coolant temperature must not exceed the limits for four-loop operation shown in CTS Figure 2.1-1.

ITS 2.1.1 in conjunction with ITS Figure 2.1.1-1 maintains the same safety limit as CTS 2.1; however, ITS 2.1.1 is more specific than CTS 2.1 and specifies that the safety limits are based on pressurizer

A.6 CTS 2.1 establishes a safety limit for the maximum RCS pressure that is applicable whenever fuel assemblies are installed in the reactor vessel.

ITS 2.1.2 maintain the same safety limit; however, ITS 2.1.1 specifies this safety limit is applicable in Modes 1, 2, 3, 4 and 5 and in Mode 6 when the reactor vessel head is on. This change is acceptable because the ITS definition of Mode envelopes all conditions when fuel is in the reactor vessel. Additionally, there is no potential that this SL could be exceeded with the reactor head removed as is possible in Mode 6. This is an administrative change with no impact on safety because the change is a reasonable interpretation of the existing requirement.

MORE RESTRICTIVE

M.1 CTS 6.7.1.a specifies that the reactor will be shut down following any safety limit violation. No completion time is specified; therefore, in accordance with CTS 3.0, a normal reactor shutdown must be initiated immediately.

If the SL for the combination of power, pressure, and temperature is violated, ITS SL 2.2.1 specifies that compliance must be restored and the reactor placed in Mode 3 within 1 hour.

If the SL for maximum reactor pressure is violated in Mode 1 or 2, ITS SL 2.2.2 specifies that compliance must be restored and the reactor place in Mode 3 within 1 hour. Additionally, compliance must be restored within 5 minutes if the SL for maximum reactor pressure is violated in Mode 3, 4, 5 or 6.

This is a more restrictive change because it provides more explicit requirements for prompt restoration of compliance with SLs and more explicit requirements for a prompt reactor shutdown if SLs are violated. This more restrictive change is acceptable because it does not introduce any operation un-analyzed while requiring a more conservative response than is currently required when an SL is violated. Therefore, this change has no significant adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE

NONE

REMOVED DETAIL

LA.1 CTS 6.7.1.a specifies that reactor operation following a SL violation may be resumed in accordance with the provisions of 10 CFR 50.36(c)(1)(i). Additionally, CTS 6.7.1.b, 6.7.1.c and 6.7.1.d establish requirements for the reporting of safety limit violations both within the New York Power Authority organization and to the NRC.

ITS SL 2.2, Safety Limit Violations, (ISTS SL 2.2 as modified by Generic Change TSTF-05 (WOG-02), Rev.1), does not specify the internal or external reporting requirements created by a SL violation. Additionally, ITS SL 2.2 does not specify that 10 CFR 50 requirements govern reactor startup following a SL violation.

These changes are needed because 10 CFR 50 requirements for reporting SL violations and for reactor startup following a SL violation govern despite repetition in the Technical Specifications. Additionally, reporting requirements within the New York Power Authority organization are more appropriately controlled by NYPA administrative requirements.

Not including these requirements in the ITS is acceptable because these requirements are specified in 10 CFR 50 and cannot be changed by NYPA. Therefore, there is no change to the existing requirements and future changes are appropriately controlled. Additionally, adequate administrative controls exist to ensure these requirements are understood and properly implemented.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 2.0:
"SAFETY LIMITS (SLs)"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 2.0:
"SAFETY LIMITS (SLs)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 2.0

This ITS Specification is based on NUREG-1431 Specification No. **2.0**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
CEOG-017	**065**	USE OF GENERIC TITLES FOR UTILITY POSITIONS	NRC Review	Incorporated	**T.2**
WOG-002 R1	**005 R1**	DELETE SAFETY LIMIT VIOLATION NOTIFICATION REQUIREMENTS	Approved by NRC	Incorporated	**T.1**

2.0 SAFETY LIMITS (SLs)

<CTS>

2.1 SLs

<2.1>

2.1.1 Reactor Core SLs

<2.1>
<DOC A.3>
<DOC A.4>
<DOC A.5>

Vessel inlet

In MODES 1 and 2, the combination of THERMAL POWER, Reactor ~~Coolant System (RCS) highest loop average~~ temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1~~A~~-1.

DB.1

<2.2>
<DOC A.6>

2.1.2 RCS Pressure SL

Insert: 2.0-1-01

In MODES 1, 2, 3, 4, ~~and 5,~~ the RCS pressure shall be maintained ≤ ~~[~~2735~~]~~ psig.

<6.7>

2.2 SL Violations

<6.7.1>

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

<6.7.1>
<DOC M.1>

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

5, or 6

2.2.2.2 In MODE 3, 4, ~~or 5~~, restore compliance within 5 minutes.

~~2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President—Nuclear Operations].~~

T.1

~~2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President—Nuclear Operations].~~

T.2

~~2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.~~

Insert: 2.0-2-01

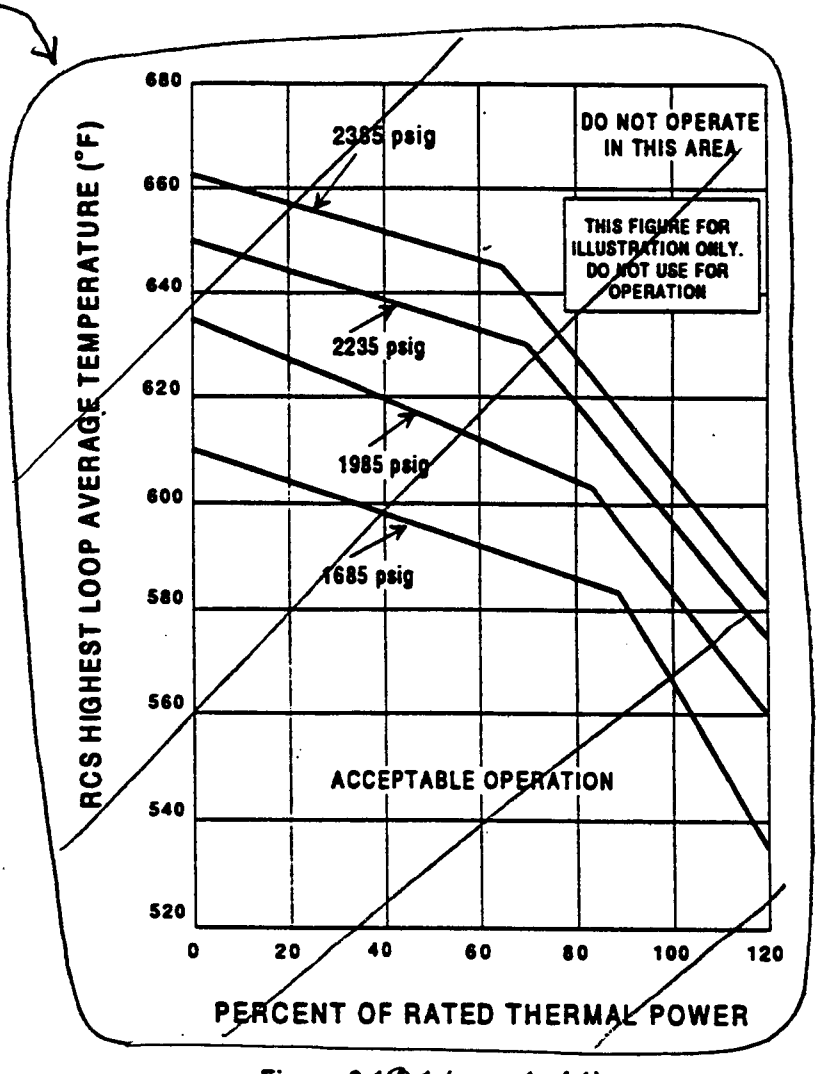
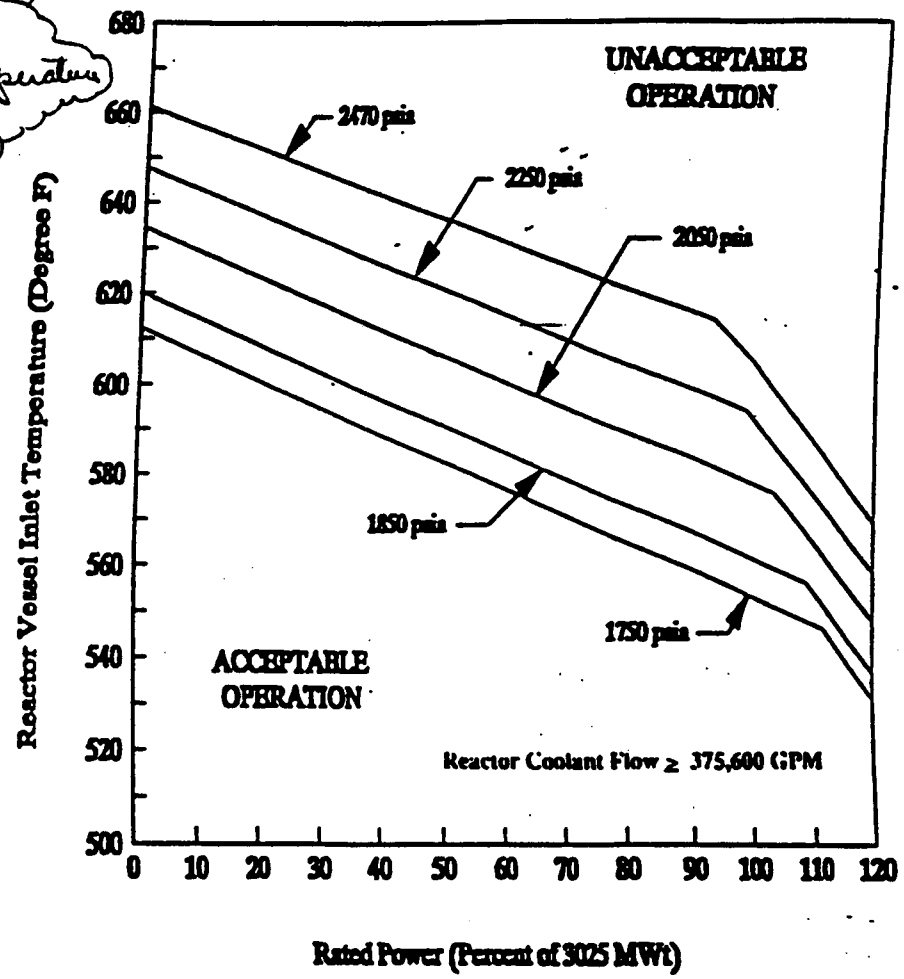


Figure 2.1.~~1~~A-1 (page 1 of 1)
Reactor Core Safety Limits

Insert: 2.0-2-01

This curve does not provide allowable limits for normal operation.
(See ~~Technical Specification 3.1.8~~ for DNB limits)

LCO 3.4.1, Pressure, Temperature and Flow DNB limits,



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt

Pressures and temperatures do not include allowance for instrument error.

Note: Based on Amendment 175

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

BACKGROUND (continued)

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

Protection

The Reactor Trip System ~~setpoints~~ (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

PA.1

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip;

d. Overpower ΔT trip;

e. Power Range Neutron Flux trip; and

f. Steam generator safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

CLB.1

Insert: B2.0-3-01

~~The curves provided in Figure B 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.~~

~~The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.~~

~~The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).~~

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RPS) Instrumentation." In MODES 3, 4,

Protection

P

(continued)

INSERT: B 2.0-3-01:

The curves provided in Figure 2.1.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits assumes:

1. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at RTP specified in the COLR;

2. An equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24% (Ref. 3);

3. Reactor coolant system total flow rate of greater than or equal to 375,600 gpm as measured at the plant; and,

4. A reference cosine with a peak of 1.55 for axial power shape.

Figure 2.1.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP}(1 + PF_{\Delta H}(1-P))$$

Where
P is the fraction of Rated Thermal Power;
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at RTP specified in the COLR; and,
$PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

INSERT: B 2.0-3-01: (continued)

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.9% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (specified in the COLR) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

BASES

APPLICABILITY (continued)

5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

PA.1

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

2.2.4

If SL 2.1.1 is violated, the Plant Superintendent and the Vice President—Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

T.1

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the Plant Superintendent and the Vice President—Nuclear Operations.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

T.1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section [7.2].
3. WCAP-8746-A, March 1977.
4. WCAP-9273-NP-A, July 1985.

Insert: B 2.0-5-01

5. 10 CFR 50.72.
6. 10 CFR 50.73.

T.1

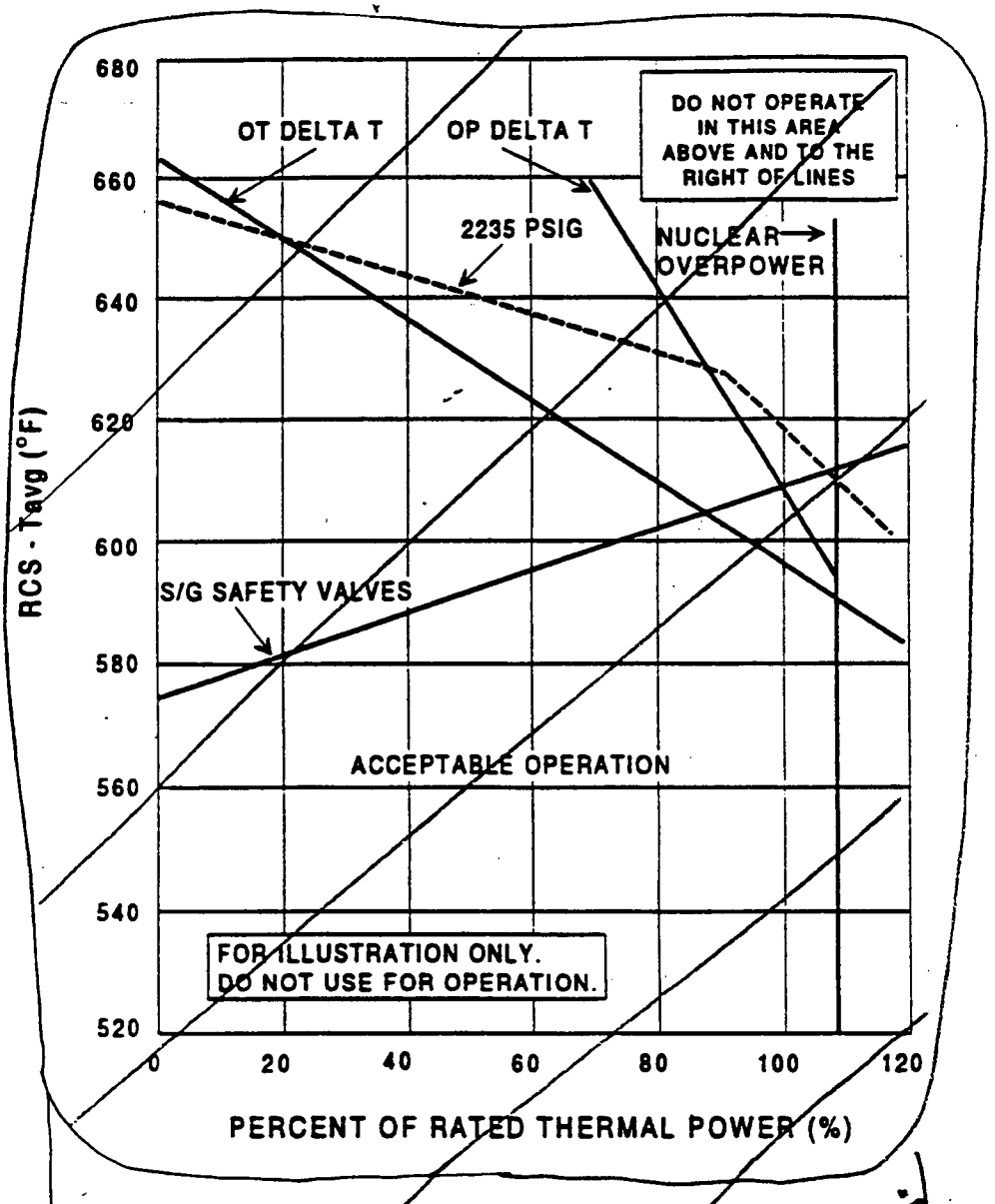


Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

2485 psig

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

Protection

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. ~~Steam line relief valve~~; Atmospheric Dump Valves

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray ~~valve.~~

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 6) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

(continued)

when the reactor vessel head is removed

and in Mode 6 when the reactor vessel head is on

BASES (continued)

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because ~~the reactor vessel head closure bolts are not fully tightened, making it unlikely that~~ the RCS can be pressurized.

not

SAFETY LIMIT VIOLATIONS

~~The following SL violations are applicable to the RCS pressure SL.~~

T.1

~~2.2.2.1~~

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

~~2.2.2.2~~

T.1

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.4

If the RCS pressure SL is violated, the Plant Superintendent and the Vice President-Nuclear Operations shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President-Nuclear Operations.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

T.1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.

(continued)

BASES

REFERENCES (continued)

5. FSAR, Section ~~[~~7.2~~]~~.

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

7. ~~10 CFR 50.72.~~

8. ~~10 CFR 50.73.~~

T.1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 2.0:
"SAFETY LIMITS (SLs)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 2.0 - SAFETY LIMITS (SLs)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 IP3 ITS Section 2.1.1 Bases for reactor core Safety Limits differs from NUREG-1431 by the retention of description of the safety limits from CTS Amendment 175. This change maintains the current licensing basis.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-05, Rev.1 (WOG-02), which deletes safety limit violation notification, reporting, and restart requirements if a safety limit is violated. This change is acceptable because it deletes requirements from the Technical Specifications that are contained in other regulations or required to comply with regulations (10 CFR 50.36 and 10 CFR 50.36(c)(1)(i)).

T.2 This change incorporates Generic Change TSTF-65 (CEOG-17), Use of generic titles for utility positions. This change allows plant specific

management titles in the CTS to be relocated to licensee controlled documents and replaced in the ITS with generic titles provided in ANSI/ANS 18.7. Personnel who fulfill these positions are required to meet the qualification requirements in ITS Specification 5.3. In addition, compliance details relating to the plant specific management position titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in accordance with ITS Specification 5.2.1.a. This approach is consistent with Generic Letter 88-06, which recommended relocation of the corporate and plant organization charts to licensee controlled documents. The intent of the Generic Letter, and of this change, is to reduce the unnecessary burden on NRC and licensee resources associated with processing license amendments.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.0:
"LCO APPLICABILITY and SR APPLICABILITY"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0 LCO APPLICABILITY

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7

Test Exception LCOs, such as 3.1.8, allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 SR APPLICABILITY

SR 3.0.3 (continued)

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is

BASES

LCO 3.0.2 (continued)

an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other

BASES

LCO 3.0.2 (continued)

specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation

BASES

LCO 3.0.3 (continued)

permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

BASES

LCO 3.0.3 (continued)

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.14, " Spent Fuel Pit Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the Spent Fuel Pit." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in the Spent Fuel Pit" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being

BASES

LCO 3.0.4 (continued)

required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry in MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for a continuous period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

LCO 3.0.4 (continued)

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

BASES

LCO 3.0.5 (continued)

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of reqired testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be

BASES

LCO 3.0.6 (continued)

declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs, such as LCO 3.1.8, allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be

BASES

LCO 3.0.7 (continued)

performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance

BASES

SR 3.0.1 (continued)

is norally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is

BASES

SR 3.0.2 (continued)

the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the

BASES

SR 3.0.3 (continued)

Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified

BASES

SR 3.0.3 (continued)

limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4

BASES

SR 3.0.4 (continued)

will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.0:
"LCO APPLICABILITY and SR APPLICABILITY"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
1-5	97	97	No TSCRs	No TSCRs for this Page	N/A
3.1-1	121	121	No TSCRs	No TSCRs for this Page	N/A
4.1-1	97	97	No TSCRs	No TSCRs for this Page	N/A
4.1-2	97	97	No TSCRs	No TSCRs for this Page	N/A
4.1-3	148	148	No TSCRs	No TSCRs for this Page	N/A

SEE ITS 1.0

1.11 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio shall be the ratio of the maximum upper excore dector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

SEE ITS 3.2.4

~~1.12~~ SURVEILLANCE INTERVAL

SR 3.0.2

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval. (A.11)

after the initial performance (M.2)

1.13 OPERATION IN A DEGRADED MODE

LCO 3.0.2

The plant is said to be operating in a degraded mode when it is operating with one or more systems listed herein inoperable as permitted by the Technical Specifications. (A.4) If inoperable components or systems are subsequently made operable, the action statements requiring plant shutdown no longer apply. (A.5)

SEE ITS 1.0

1.14 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY

Noble gas $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 10 minutes, making up at least 95% of the total activity in the coolant.

1.15 DOSE EQUIVALENT I-131

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

Add LCO 3.0.7 (A.9)

Add LCO 3.0.1 (A.3)

Add LCO 3.0.4 (M.1)

Add LCO 3.0.5 (L.1)

Add LCO 3.0.6 (A.8)

3. LIMITING CONDITIONS FOR OPERATION

Add LCO 3.0.3 A.6

~~For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.~~ A.7

3.1 REACTOR COOLANT SYSTEM

~~Applicability~~ A.2

~~Applies to the operating status of the Reactor Coolant System: operational components; heatup; cooldown; criticality; activity; chemistry and leakage.~~

~~Objective~~

~~To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.~~

Specification

A. OPERATIONAL COMPONENTS

SEE ITS 3.4.5 3.4.6 3.4.7 3.4.8

1. Coolant Pumps

a. When a reduction is made in the boron concentration of the reactor coolant, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation.

SEE ITS 3.4.5

b. (1) When the reactor coolant system T_{avg} is greater than 350°F and electrical power is available to the reactor coolant pumps, and as permitted during special plant evolutions, at least one reactor coolant pump shall be in operation. All reactor coolant pumps may be de-energized for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

(2) When the reactor is subcritical and reactor coolant system T_{avg} is greater than 350°F, control bank withdrawal shall be prohibited unless four reactor coolant pumps are operating.

SEE ITS 3.4.6

c. When the reactor coolant system T_{avg} is greater than 200°F and less than 350°F, and as permitted during special plant evolutions, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. All reactor coolant pumps may be de-energized with RHR not in service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

4 SURVEILLANCE REQUIREMENTS

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation. A.2

SR 3.0.1 Performance of any surveillance test outlined in these specifications is not required if the plant condition is the same as the condition into which the plant would be placed by an unsatisfactory result of that test. A.10

SR 3.0.2 Failure to perform a surveillance requirement within the allowed surveillance interval (including extensions specified in definition 1.12), shall constitute noncompliance with the operability requirements of the limiting conditions for operation (LCOs). A.11

SR 3.0.3 The time limits for associated action requirements are applicable at the time it is identified that a surveillance requirement has not been performed. Action requirements may be delayed for up to 24 hours to permit completion of the missed surveillance when the allowable outage time limits of the action requirements are less than 24 hours (i.e. for LCOs of less than 24 hours, a 24 hour delay period is permitted before entering the LCO; for LCOs greater than 24 hours, no delay period is permitted). L.2

a 24 hour L.2

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions. A.2

SEE ITS 3.3

Specification

A. Calibration, testing, and checking of analog channel and testing of logic channel shall be performed as specified in Table 4.1-1.

B. Sampling and equipment tests shall be conducted as specified in Table 4.1-2 and 4.1-3, respectively.

Basis

A surveillance test is intended to identify conditions in a plant that would lead to a degradation of reactor safety. Should a test reveal such a condition, then the Technical Specifications require that, either immediately or after a specified period of time, the plant be placed in a condition which mitigates or eliminates the consequences of additional related casualties or accidents. If the plant is already in a condition A.1

Add SR 3.0.4 M.1

A.1

condition which would satisfy the failure criteria of the test, then plant safety is assured and performance of the test yields either meaningless information or information that is not necessary to determine safety limits or limiting conditions for operation of the plant.

Likewise, systems and components are assumed to be operable as defined in paragraph 1.5, and satisfying safety limits or LCOs for a given plant operating condition, when surveillance requirements have been satisfactorily performed within the allowed surveillance interval and extensions as specified in definition 1.12. However, nothing in this provision shall be construed as implying that systems or components are operable when they are found or known to be inoperable although still meeting the surveillance requirements. LCO action requirements associated with operation in a degraded mode are applicable when surveillance requirements have not been completed within the allowed surveillance interval. The time limits of such LCOs apply from the point in time it is identified that a surveillance has not been performed and not at the time the allowed surveillance interval was exceeded.

For a missed surveillance, if the allowable outage time limits of the applicable LCO action requirements are less than 24 hours or a shutdown is required, then a 24-hour delay is permitted in implementing the action requirements. The purpose of the delay is to permit the completion of a missed surveillance before a shutdown or some other remedial measure precludes completion of the surveillance. This allowance of a delay includes consideration of the plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. If a surveillance is not completed within the 24-hour delay, than the time limits of the associated action requirements are applicable at the time. When a surveillance is performed within the 24 hour delay and the Surveillance Requirements are not met (e.g. the system or component is declared inoperable), the time limits of the LCO action requirements are applicable at that time.

Failure to perform the surveillance within the allowed surveillance interval and extension as specified in definition 1.12 is still a violation of the LCO operability requirement subject to enforcement and reportability requirements as may be applicable.

Definition 1.12 establishes the limit for which the specified time interval for Surveillance Requirements may be extended.

A.1

It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g. transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month or 24-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed on an 18-month or 24-month basis. Likewise, it is not the intent that 24 month surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Definition 1.12 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval. The phrase "at least" associated with a surveillance frequency does not negate the 25% extension allowance of Definition 1.12; instead, it permits the performance of more frequent surveillance activities.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency of once per shift is deemed adequate for reactor and steam system instrumentation.

Calibration

Calibrations are performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of 18 or 24 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and 18 or 24 months for the process system channels is considered acceptable.

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PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.0 - LCO AND SR APPLICABILITY

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 ITS Limiting Condition for Operation (LCO) 3.0.1 (as modified by Generic Change TSTF-6, Revision 1) is added to provide an explicit statement that each of the ITS LCOs must be met during the Modes or other specified conditions in that LCO's Applicability. There is no equivalent statement in the CTS.

ITS LCO 3.0.1 is needed because the ITS uses a format in which an

Applicability statement is used to specify all conditions in which an LCO must be met; whereas, the CTS specifies the conditions in which an LCO must be met as part of the statement of the LCO requirement. The addition of ITS LCO 3.0.1 is an administrative change with no impact on safety because ITS LCO 3.0.1 does not change any existing requirements.

A.4 CTS Definition 1.13, Operation in a Degraded Mode, specifies that the plant is operating in a degraded mode when it is operating with one or more systems inoperable. CTS 1.13 includes an implied requirement that Actions are required when operating in a degraded mode; however, the defined term Operation in a Degraded Mode is not used in any CTS.

ITS LCO 3.0.2 is added to provide an explicit statement that if a failure to meet any LCO is discovered, then the associated Required Actions must be met. Additionally, ITS LCO 3.0.2 clarifies that if the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

By this change, CTS Definition 1.13 is replaced by ITS LCO 3.0.2 which substitutes the term "failure to meet an LCO" for "Operation in a degraded mode" and then provides an explicit statement of the actions required to be taken upon discovery of a failure to meet an LCO. This is an administrative change with no adverse impact on safety because the change only clarifies the existing requirement.

A.5 CTS Definition 1.13, Operation in a Degraded Mode, specifies that the action statements "requiring plant shutdown" no longer apply if inoperable components or systems are subsequently made operable.

ITS LCO 3.0.2 specifies that if an LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. ITS LCO 3.0.2 extends the provisions of CTS 1.13 to apply to any Actions and not just to those Actions requiring plant shutdown. This is an administrative change because a reasonable interpretation of CTS 1.13 is that it is not intended to exclude Actions that do not require plant shutdown.

This change is acceptable because restoration of LCO requirements or placing the plant in a condition where the LCO is not applicable ensures that the design or analysis assumptions enforced by the LCO are satisfied. In those cases where restoration of LCO requirements or placing the plant in a condition where the LCO is not applicable may not satisfy design or analysis assumptions (e.g., violation of pressure temperature limits), ITS stipulate that required actions must be completed although the associated conditions no longer exist. Therefore, this change has no adverse impact on safety.

A.6 CTS does not include any explicit requirements when Actions are not specified or when the Actions specified are not met; however, there is an implicit assumption in the CTS that immediate plant shutdown is required.

ITS LCO 3.0.3 is added to provide an explicit statement that the plant must be placed in cold shutdown condition (i.e., Mode 5) when Actions are not specified or the specified Actions are not met. Additionally, ITS LCO 3.0.3 specifies that the plant shutdown must be initiated within 1 hour and completed within 37 hours of the discovery of the condition. These Completion Times include time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower Modes of operation permit the shutdown to proceed in a controlled and orderly manner that is within the maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is Operable.

This is an administrative change with no significant adverse impact on safety because the Actions specified in LCO 3.0.3 are a reasonable interpretation of the existing requirements.

A.7 CTS 3.0 specifies that for the cases where no exception time is specified for inoperable components, this time is assumed to be zero. This requirement is not retained in ITS because ITS LCOs include allowable out of service times (AOTs) and Completion Times for all identified conditions; otherwise, ITS LCO 3.0.3 is applicable (See ITS

3.0, DOC A.6). Therefore, deletion of CTS 3.0 is an administrative change with no adverse impact on safety.

A.8 ITS LCO 3.0.6 is added to provide an exception to ITS LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is needed because LCO 3.0.2 would require that the Conditions and Required Actions of an inoperable supported system be entered solely due to the inoperability of the support system. This change is acceptable because each LCO's Required Actions ensure the unit is maintained in a safe condition including the potential impact on supported systems; otherwise, an LCO's Required Actions will direct entry into the Required Actions of the supported system. Therefore, this change has no significant adverse impact on safety. This is an administrative change because any differences between CTS and ITS needed to support the implementation of ITS LCO 3.0.6 are addressed in this conversion package with each ITS LCO.

A.9 ITS LCO 3.0.7 is added to provide guidance for meeting the test exception ITS LCO 3.1.8. This test exception LCO allows specified Technical Specification requirements to be changed (made applicable in part or whole, or suspended) to permit the performance of special tests or operations that otherwise could not be performed. LCO 3.0.7 eliminates the confusion about which LCOs apply during the performance of a special test or operation. The addition of LCO 3.0.7 is an administrative change with no adverse impact on safety because it provides an explicit statement that clarifies the existing interpretation that these test exception LCOs change the requirements of another LCO. Allowances provided by the test exception, ITS LCO 3.1.8, are addressed in the Discussion of Changes associated with this specification.

A.10 CTS 4.1 specifies that performance of any surveillance is not required if the plant condition is the same as the condition into which the plant would be placed by an unsatisfactory result of that test (i.e., performance is required in the LCO's Applicable Modes).

ITS Surveillance Requirement (SR) 3.0.1 maintains this requirement by

stating that SRs must be met during the Modes or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Additionally, failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

This is an administrative change with no significant adverse impact on safety because ITS SR 3.0.1 is consistent with existing requirements in CTS 4.1 and CTS definition 1.12, Surveillance Interval. The second sentence of ITS SR 3.0.1, "Failure to meet a Surveillance, whether such failure is experienced during the performance of Surveillance or between performances of the Surveillance, shall be failure to meet the LCO," is a requirement that is consistent with the intent but not explicitly stated in the CTS. The addition of ITS SR 3.0.1 is an administrative change with no adverse impact on safety because it is a more explicit statement of existing requirements.

A.11 CTS 4.1 and CTS Definition 1.12, Surveillance Interval, specify that failure to perform an SR within the allowed surveillance interval (including extensions specified in definition 1.12), constitutes noncompliance with the Operability requirements of the LCOs. CTS Definition 1.12, specifies that each SR must be performed within the surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

ITS SR 3.0.2 maintains the same requirement (See ITS 3.0, DOC M.2); however, ITS SR 3.0.2 clarifies that this allowance also applies to the Required Actions with Completion Times that require periodic performance except that the initial performance must be performed within the specified Completion Time (See ITS 3.0, DOC M.2). This is an administrative change with no adverse impact on safety because Required Actions that specify periodic performance mandate periodic verification of a status or condition are functionally equivalent to SRs.

MORE RESTRICTIVE

M.1 ITS LCO 3.0.4 and SR 3.0.4 are added to establish limitations on changes in Modes or other conditions specified in LCO applicability if the LCO or SR is not met. ITS LCO 3.0.4 and SR 3.0.4 preclude placing the unit in a mode or other specified condition stated in the applicability when conditions are such that the requirements of the LCO would not be met if the LCO were entered. Although these restrictions are generally consistent with current practice, there are no equivalent requirements in the CTS. This more restrictive change is acceptable because it does not introduce any operation unanalyzed while requiring a more conservative approach to satisfying LCO Applicability restrictions and SR requirements than is currently required. Therefore, this change has no adverse impact on safety.

M.2 CTS 4.1 and CTS Definition 1.12, Surveillance Interval, specify that each SR must be performed within the surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

ITS SR 3.0.2 maintains the same requirement; however, ITS SR 3.0.2 is limited so that the extension does not apply to either of the following: SRs that have a Frequency of "once"; and, the initial performance of Required Actions with Completion Times that require subsequent periodic performance.

This change is needed because Required Actions with Completion Times that require periodic performance usually involve the following: verification that no loss of function has occurred by checking the status of redundant or diverse components; or, implementing the safety function of the inoperable equipment in an alternative manner. Therefore, this change has no significant adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 ITS LCO 3.0.5 is added to provide an exception to ITS LCO 3.0.2 to permit restoration of inoperable equipment to an operable status to

demonstrate the Operability of the equipment being returned to service or to demonstrate that other equipment or variables are within limits.

ITS LCO 3.0.5 is needed because many Technical Specification Actions require an inoperable component to be removed from service, (e.g., maintain an isolation valve closed or trip an inoperable instrument channel) and the testing required for restoration cannot be performed while continuing to comply with Required Actions. To allow the performance of Surveillance Requirements, the exception to ITS 3.0.2 provided in ITS 3.0.5 allows returning the inoperable equipment to service before demonstrating its Operability. Without this allowance certain components could not be restored to Operable status and a plant shutdown could be required. The addition of ITS LCO 3.0.5 is acceptable because it is not the intent that the Technical Specifications preclude the return to service of a component believed to be operable. This less restrictive change will have no adverse impact on safety because this allowance is safer than requiring a plant shutdown to complete the restoration and confirmatory testing of inoperable equipment.

L.2 CTS 4.1 provides a grace period if an SR is missed so that associated Actions may be delayed for up to 24 hours (to permit completion of the missed SR). This allowance only applies if the Required Action for the missed SR has a Completion Time of less than 24 hours. Otherwise, no grace period is provided.

ITS SR 3.0.3 also provides a grace period if an SR is missed; however, the grace period is based on the SR Frequency instead of the Completion Time for the Required Action for the missed SR. Specifically, ITS SR 3.0.3 allows a grace period of the lessor of 24 hours or the specified SR frequency to perform a missed surveillance.

The existing requirement in ITS SR 3.0.3 differs from the existing requirement in CTS 4.1 in that the ITS SR 3.0.3 allowance is determined based on the SR Frequency of the missed SR while the CTS 4.1 allowance is based on the LCO allowable out of service time for the component made inoperable by the missed SR. Therefore, comparison of the differences in the length of the allowance in CTS versus ITS is not relevant. The difference between the CTS and the ITS is that in the CTS an LCO with an SR not satisfied is considered not met while in ITS an LCO with an SR

not satisfied is considered met for up to 24 hours while the SR is performed. This change is based on NRC Generic Letter 87-09 that states, "It is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed. The opposite is in fact the case, the vast majority of surveillances demonstrate that systems or components in fact are Operable. When a Surveillance is missed, it is primarily a question of Operability that has not been verified by the performance of the required surveillance."

The NRC concluded in the Generic Letter 87-09 that 24 hours is an acceptable time limit for completing a missed Surveillance when the allowable outage times of the Actions are less than the 24 hour limit or a shutdown is required to comply with actions. This conclusion was based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance and the safety significance of the delay in completing the Surveillance. Because 24 hours has been determined to be an acceptable time limit for completing a missed SR, this 24-hour deferral applies to all systems or components, even if the Actions Completion Time is 24 hours or less. However, the ITS does not apply the limit of 24 hours to complete a missed SR if the specified Frequency of the missed Surveillance is less than 24 hours. If the SR frequency is less than 24 hours, ITS permits a delay period to complete a missed SR equal to the normal SR frequency. Based on the above discussion, this less restrictive change will have no adverse impact on safety.

REMOVED DETAIL

NONE

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.0:
"LCO APPLICABILITY and SR APPLICABILITY"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.0 - LCO AND SR APPLICABILITY

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   ITS LCO 3.0.5 is added to provide an exception to ITS LCO 3.0.2 to permit restoration of inoperable equipment to an operable status to demonstrate the Operability of the equipment being returned to service or to demonstrate that other equipment or variables are within limits.

   ITS LCO 3.0.5 is needed because some Technical Specification Actions require an inoperable component to be removed from service, (e.g., maintain an isolation valve closed or trip an inoperable instrument channel) and the testing required for restoration cannot be performed while continuing to comply with Required Actions. To allow the performance of Surveillance Requirements, the exception to ITS 3.0.2 provided in ITS 3.0.5 allows returning the inoperable equipment to service before demonstrating it Operability. Without this allowance certain components could not be restored to Operable status and a plant shutdown could be required.

   This change does not result in a significant increase in the probability of an accident previously evaluated because the equipment will be restored only when it is in a condition expected to provide the required safety function. As stated in Generic Letter 87-09, "The vast majority of surveillances do in fact demonstrate that systems or components are operable. Also, returning the equipment to service under administrative controls will promote timely restoration of the Operability of the equipment and reduce the probability of any events that may have been prevented by such operable equipment."

This change does not result in a significant increase in the consequences of an accident previously evaluated because the equipment to be restored is already out of service. Therefore, the unavailability of the equipment has been previously considered in the evaluation of consequences of an accident. Temporarily returning the equipment to service in a state expected to function as required to mitigate the consequences of a previously analyzed accident will promote timely restoration of the Operability of the equipment and restore the capabilities of the equipment to mitigate the consequences of any events as previously analyzed.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because performance of the surveillance is a confirmatory check of that capability that demonstrates that the equipment is operable. For those times when equipment temporarily returned to service under administrative controls is subsequently determined to be inoperable, the resulting condition is comparable to the equipment having been determined to be inoperable during operation, with continued operation for a specified time allowed to complete required actions.

   Temporarily returning inoperable equipment to service for confirming Operability will place the plant in a condition that has been previously evaluated and determined to be acceptable for short periods. Additionally, the equipment has been determined to be in a condition that provides the previously determined margin of safety. The performance of the surveillance simply confirms the expected result and capability of the equipment.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.0 - LCO AND SR APPLICABILITY

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   This change adds ITS SR 3.0.3 differs from the existing requirement in CTS 4.1 in that the ITS SR 3.0.3 allowance is determined based on the Frequency of the missed SR while the CTS 4.1 allowance is based on the LCO allowable out of service time for the component made inoperable by the missed SR. Specifically, ITS SR 3.0.3 allows a grace period of the lessor of 24 hours or the specified SR frequency to perform a missed surveillance.

   This change will not result in a significant increase in the probability of an accident previously evaluated based on NRC Generic Letter 87-09 that states that "It is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed. The opposite is in fact the case, the vast majority of surveillances demonstrate that systems or components in fact are Operable. When a Surveillance is missed, it is primarily a question of Operability that has not been verified by the performance of the required surveillance."

   The NRC concluded in Generic Letter 87-09 that 24 hours is an acceptable time limit for completing a missed Surveillance when the allowable outage times of the Actions are less than the 24-hour limit or a shutdown is required to comply with actions. This conclusion was based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance and the safety significance of the delay in completing the Surveillance. Because 24 hours has been determined to be an acceptable time limit for completing a missed SR, this 24-hour deferral applies to all systems or components,

even if the Actions Completion Time is 24 hours or less. However, the ITS does not apply the limit of 24 hours to complete a missed SR if the specified Frequency of the missed Surveillance is less than 24 hours. If the SR frequency is less than 24 hours, ITS permits a delay period to complete a missed SR equal to the normal SR frequency.

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

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because when a Surveillance is missed, it is primarily a question of Operability that has not been verified by the performance of the required surveillance.

   The NRC concluded in Generic Letter 87-09 that 24 hours is an acceptable time limit for completing a missed Surveillance when the allowable outage times of the Actions are less than the 24-hour limit or a shutdown is required to comply with actions. This conclusion was based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance and the safety significance of the delay in completing the Surveillance. Because 24 hours has been determined to be an acceptable time limit for completing a missed SR, this 24-hour deferral applies to all systems or components, even if the Actions Completion Time is 24 hours or less. However, the ITS does not apply the limit of 24 hours to complete a missed SR if the specified Frequency of the missed Surveillance is less than 24 hours. If the SR frequency is less than 24 hours, ITS permits a delay period to complete a missed SR equal to the normal SR frequency.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.0:
"LCO APPLICABILITY and SR APPLICABILITY"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.0

This ITS Specification is based on NUREG-1431 Specification No. **3.0**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
BWOG-007		ADD AN ADDITIONAL EXAMPLE TO THE SR 3.0.2 BASES	Rejected by TSTF	Not Incorporated	**N/A**
BWOG-022	**122 R0**	REVISE LCO 3.0.2 BASES TO REMOVE POSSIBLE CONFUSION	Approved by NRC	Incorporated	**T.5**
BWROG-002	**001 R1**	MAKE LCO 3.0.5 APPLICABLE TO VARIABLES IN ADDITION TO SYSTEMS AND EQUIPMENT.	Rejected by NRC	N/A	**N/A**
CEOG-024 R1	**071 R1**	ADD EXAMPLE OF SFDP TO THE 3.0.6 BASES	Approved by NRC	Not Incorporated	**N/A**
WOG-003.1 R1	**006 R1**	ADD EXCEPTION FOR LCO 3.0.7 TO LCO 3.0.1	Approved by NRC	Incorporated	**T.1**
WOG-003.2 R1	**007 R1**	DELETE THE 1 HOUR TIME LIMIT TO BEGIN REDUCING POWER FROM LCO 3.0.3	Rejected by NRC	Not Incorporated	**N/A**

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WOG-003.3 R2	**008 R2**	REVISE THE SR 3.0.1 BASES TO ALLOW CREDIT FOR UNPLANNED EVENTS TO MEET ANY SURVEILLANCE	Approved by NRC	Incorporated	**T.2**
WOG-004.4 R1	**012 R1**	DELETE LCO 3.1.9 AND 3.1.11 (PHYSICS TESTS EXCEPTIONS)	Approved by NRC	Incorporated	**T.3**
WOG-035	**103 R1**	ADD BRACKETED INFORMATION TO LCO 3.0.4 INAPPROPRIATELY DELETED BY REV. 0 CHANGE BWR-26	Rejected by NRC	Not Incorporated	**N/A**
WOG-036	**104 R0**	RELOCATES DISCUSSION OF EXCEPTIONS FROM LCO 3.0.4 TO THE BASES	Approved by NRC	Incorporated	**T.4**
WOG-042	**052**	IMPLEMENT 10 CFR 50, APPENDIX J, OPTION B	TSTF to Rewrite	Not Incorporated	**N/A**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.6**
WOG-077	**165 R0**	REVISE THE LCO 3.0.5 BASES TO REFER TO TESTING AND NOT SRS	Approved by NRC	Incorporated	**T.7**
WOG-078	**166 R0**	CORRECT INCONSISTENCY BETWEEN LCO 3.0.6 AND THE SFDP REGARDING PERFORMANCE OF AN EVALUATION	Approved by NRC	Incorporated	**T.8**
WOG-107		SFDP CLARIFICATIONS	TSTF Review	Not Incorporated	**N/A**

<CTS> 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

<DOC A.3> LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2. and LCO 3.0.7 (T.1)

<1.13> <DOC A.4> <DOC A.5> LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

<3.0> <DOC A.6> <DOC A.7> LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

<DOC M.1> LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

~~Reviewers's Note: LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

X.1

<Doc. L.1>

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

(continued)

3.0 LCO APPLICABILITY (continued)

<DOC A.8> LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, ~~additional~~ an evaluation~~s~~ ~~and limitations may be required~~ shall be performed in accordance with Specification 5.5.~~15~~ 14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. T.8

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

<DOC A.9> LCO 3.0.7

Test Exception LCOs ~~3.1.9~~ such as 3.1.~~10~~ 8, ~~3.4.11~~, ~~and 3.4.19~~ allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications. T.3 T.6

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

<4.1>
<DOC A.10>

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

<4.1>
<DOC A.11>
<1.12>
<DOC M.2>

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

<4.1>
<DOC L2>

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

(continued)

3.0 SR APPLICABILITY

SR 3.0.3 (continued)

declared not met, and the applicable Condition(s) must be entered.

<DOC M.D>

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.

~~Reviewer's Note: SR 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, SR 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in SR 3.0.4 were previously applicable in all MODES. Before this version of SR 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

Insert: B 3.0-2-01

alternatives

may

T.5

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

(continued)

INSERT: B 3.0-2-01

T.5

Additionally, if intentional entry into ACTIONS

BASES (continued)

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "~~Fuel Storage Pool~~ Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel

14 14 Spent Fuel Pit

(continued)

BASES

LCO 3.0.3 (continued)

assemblies in the ~~fuel storage pool~~ spent fuel pit." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the ~~fuel storage pool~~ spent fuel pit" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability

(continued)

BASES

LCO 3.0.4 (continued)

that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Insert: B3.0-6-01

T.4

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. ~~[In some cases (e.g., ...) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]~~

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to

(continued)

INSERT: B 3.0-6-01

T.4

The exceptions allow entry in MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for a continuous period of time.

BASES

LCO 3.0.5 (continued)

provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate: T.7

Required testing

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance. T.7

required testing to demonstrate OPERABILITY

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system. T.7

required testing

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the

(continued)

BASES

LCO 3.0.6 (continued)

supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

14

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support

(continued)

BASES

LCO 3.0.6 (continued)

system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs (~~3.1.9, 3.1.10, 3.1.11, and 3.4.19~~) allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

such as LCO 3.1.8

T.3

T.6

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Insert: B3.0-10-01

T.2

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

(continued)

INSERT: B 3.0-10-01

T.2

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

(continued)

BASES

SR 3.0.2 (continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most

(continued)

BASES

SR 3.0.3 (continued)

probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

(continued)

BASES

SR 3.0.4 (continued)

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event,

(continued)

BASES

SR 3.0.4 (continued)

condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.0:
"LCO APPLICABILITY and SR APPLICABILITY"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.0 - LCO AND SR APPLICABILITY

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

None

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-06, Rev.1 (WOG 3.1), which clarifies that LCO 3.0.7, governing Test Exception LCOs, permits a relaxation in the requirements of LCO 3.0.1, that LCOs shall be met during the MODES or other specified conditions in the Applicability. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates Generic Change TSTF-08, Rev.2 (WOG 3.3), which revises the Bases for SR 3.0.1 to clarify that credit may be taken for unplanned events to satisfy any SR, not just those in Section 3.8, "Electrical Power Systems". This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.3 This change incorporates Generic Change TSTF-12, (WOG 4.4), which deletes LCO 3.1.9, "Physics Tests Exceptions - Mode 1" and LCO 3.1.11, "Shutdown Margin (SDM) Test Exceptions" and the associated Bases. The physics tests that LCO 3.1.9 required are RCCA Pseudo Ejection Test, RCCA Pseudo Drop and Misalignment Test, and Xenon Stability Measurements. These physics tests were only contained in initial plant startup testing programs. This physics test exception can be deleted since these physics tests are never performed during post-refueling

startup testing. The physics test that LCO 3.1.11 requires was the Rod Worth Measurement condition. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement.

T.4 This change incorporates Generic Change TSTF-104, Rev.0 (WOG 36), which eliminates the discussion provided in LCO 3.0.4 with respect to the use of exceptions and provides the necessary discussion in the Bases. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.5 This change incorporates Generic Change TSTF-122, Rev.0 (BWROG 22), which revises the LCO 3.0.2 Bases to remove possible confusion. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.6 This change incorporates Generic Change TSTF-136, Rev.0 (WOG 059), which makes administrative changes to support the combination of LCO 3.1.1 and LCO 3.1.2. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.7 This change incorporates Generic Change TSTF-165, Rev.0 (WOG 077), which revises the Bases for LCO 3.0.5 to use the acronym "SR" instead of the word "testing" when referring to testing required by Technical Specifications. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.8 This change incorporates Generic Change TSTF-166, Rev.0 (WOG 078), which corrects an inconsistency between LCO 3.0.6, the Safety Function Determination Program (SFDP), and the LCO 3.0.6 Bases. As currently written, LCO 3.0.6 does not explicitly require an evaluation in accordance with the SFDP, rather it states that additional evaluations may be required. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

X.1 NUREG-1431 LCO 3.0.4 differs from previous version of LCO 3.0.4 in that restrictions on Mode changes apply only for Modes 1, 2, 3 and 4. NUREG-1431 specifies that before this relaxed version of LCO 3.0.4 can be implemented, the licensee must review the existing technical specifications to determine where specific restrictions on Mode changes or Required Actions should be included in individual LCOs to justify this change. Additionally, this evaluation must be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.

IP3 Current Technical Specifications do not include any requirement similar to LCO 3.0.4; therefore, there are no restrictions on Mode changes when an LCO is not met. As a result, the relaxation to LCO 3.0.4 provided in Revision 1 to NUREG-1431 does not result in any relaxation to IP3 Current Technical Specifications.

Indian Point 3

Improved

Technical Specifications

Conversion Submittal

Volume 3

New York Power Authority

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.1:
"SHUTDOWN MARGIN (SDM)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

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

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

ACTIONS

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 the limits specified in the COLR.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks within the limits of LCO 3.1.5, "Shutdown Bank Insertion Limits" 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.

BASES

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. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis.

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 energy deposition of ≤ 250 cal/gm for non-irradiated fuel and ≤ 200 cal/gm for irradiated fuel to satisfy requirements 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 is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment

BASES

APPLICABLE SAFETY ANALYSES (continued)

initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

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

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high neutron flux level trip or a overtemperature ΔT trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists

BASES

APPLICABLE SAFETY ANALYSES (continued)

between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36. 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.

The MSLB (Ref. 2) and the boron dilution (Ref. 2) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). 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

BASES

APPLICABILITY (continued)

are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, Control Bank Insertion Limits.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average loop temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

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.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Chapter 14.

3. 10 CFR 100.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.1:
"SHUTDOWN MARGIN (SDM)"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-1	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-15	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-16	181	181	No TSCRs	No TSCRs for this Page	N/A

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

A.1 A.2

~~Applicability:~~

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

~~Objectives:~~

~~To ensure:~~

~~1. Core subcriticality after reactor trip.~~

~~2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and~~

~~3. Limit potential reactivity insertions caused by hypothetical control rod ejection.~~

Specifications: M.1

LCO 3.1.1 ~~3.10.1~~ Shutdown Reactivity

LCO 3.1.1, Applicability ~~3.10.1.1~~ Whenever T_{avg} > 200°F (Mode 2 with keff < 1.0, Modes 3, 4 and 5) the shutdown margin shall be ≥ ~~1.3%~~ (Specified in COLR) ΔK/k. LA.1

Req. Act A.1 ~~3.10.1.2~~ When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit. (within 15 minutes) A.3

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq (F_Q^{RTP}/p) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \leq 0.5$$

$$F_{\Delta H}^N \leq F_{\Delta H}^{RTP}(1 + PF_{\Delta H}(1-P))$$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

SEE ITS 3.2.1 ITS 3.2.2

Amendment No. ~~13~~, ~~49~~, ~~48~~, ~~61~~, ~~73~~, ~~86~~, ~~103~~, 112

Add SR 3.1.1.1 M.2

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3 % $\Delta k/k$ is required to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

The action to be taken when shutdown margin in out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

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

A.1

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequency over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The assumed control rod drop time in the safety analysis is 2.7 seconds, consisting of 1.80 seconds for normal rod drop time plus additional margin which includes a seismic allowance. The required control rod drop time in Section 3.10.8 is therefore consistent with that assumed in the safety analysis.

REFERENCE

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program," August 1975
2. FSAR Appendix 14C
3. Letter from J.P. Bayne to S.A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."
4. WCAP-14668, "Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3," October 1996 (Proprietary).

A.1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.1:
"SHUTDOWN MARGIN (SDM)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.1 - SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.1.2 specifies that if the shutdown margin (SDM) requirements in CTS 3.10.1.1 are not met, then boration must be initiated. No completion time for the required action is specified, so a completion time of zero is assumed in accordance with CTS 3.0.

Under the same conditions, ITS 3.1.1, Required Action A.1, specifies that boration must be initiated within 15 minutes. This change is

needed because 15 minutes provides a reasonable time for an operator to align systems and initiate injection of boron. This is an administrative change because the new requirement is consistent with a reasonable interpretation of the existing requirement and still ensures that the appropriate action is pursued without delay and in a controlled manner. Therefore, this change has no significant adverse impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.10.1.1 requires SDM to be ≥ 1.3% Δk/k "whenever Tavg is > 200°F" (i.e., Modes 1, 2, 3 and 4). ITS LCO 3.1.1 requires SDM to be within the limits specified in the Core Operating Limits Report(COLR)(See ITS 3.1.1, DOC LA.1) whenever the plant is in Mode 2 with keff < 1.0 or in Modes 3, 4, and 5.

The first change, requiring that SDM limits are met in Mode 5 (i.e., Tavg is ≤ 200°F) is needed because minimum required SDM is assumed as an initial condition in the safety analysis for a boron dilution event that could occur in Mode 5. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring that SDM limits are met and periodically verified when in Mode 5. Therefore, this change has no adverse impact on safety.

The second change, excluding Mode 1 or in Mode 2 with keff ≥ 1.0, is needed because SDM is maintained within the same limits by complying with ITS LCO 3.1.5, Shutdown Bank Insertion Limits, and ITS LCO 3.1.6, Control Bank Insertion Limits, which also reference the limits specified in the COLR. This part of the change to the Applicability is an administrative change with no impact on safety because ITS LCO 3.1.5 and ITS LCO 3.1.6 maintain the same limits as an intrinsic part of meeting the shutdown and control bank insertion limits.

M.2 CTS 3.10.1.1 requires SDM to be within specified limits whenever Tavg is > 200°F (See ITS 3.1.1, DOC M.1); however, CTS does not include any specific requirements for the periodic verification that SDM limits are met (CTS Table 4.1-1, Item 1, does require an analysis of boron

concentration 2 days/week; however, no explicit acceptance criteria to verify SDM requirement is met.)

ITS SR 3.1.1.1 adds a new requirement to verify that SDM is within the limits specified in the COLR (See ITS 3.1.1, DOC LA.1) every 24 hours. This change is needed to require periodic verification that SDM limits are met. The SR Frequency of 24 hours is acceptable because SDM changes slowly due to the slow change in required RCS boron concentration and because RCS makeup sources are maintained within boron concentration limits that minimize the potential for dilution. Therefore, verification of SDM every 24 hours provides a high degree of assurance that SDM will be within required limits if an event occurs.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that SDM is within limits specified in the LCO. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 3.10.1.1 requires that SDM be "≥ 1.3% Δk/k." ITS LCO 3.1.1 requires SDM to be within the limits specified in the Core Operating Limits Report (COLR).

This change allows the specific limits for shutdown margin to be removed from the ITS and relocated to the Core Operating Limits Report (COLR). This change is needed because the specific value for SDM is a cycle-specific variable. Therefore, by maintaining the SDM value in the COLR, the core reload design can be completed after shutdown when the actual end of cycle burnup is known. This saves redesign efforts that occur if actual burnup differs from the projected value.

This change is acceptable because ITS LCO 3.1.1 maintains the

requirement to meet SDM requirements and ITS 5.6.5, Core Operating Limits Report (COLR), includes detailed requirements that ensure SDM limits will be properly established and maintained. Requirements established by ITS 5.6.5 include the following:

a. The analytical methods used to determine the core operating limits (including the SDM) must be those previously reviewed and approved by the NRC. The approved documents that document this approved methodology must be listed in ITS 5.6.5 and can be changed only with a TS change.

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

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications. Additionally, an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.1:
"SHUTDOWN MARGIN (SDM)"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.1 - SHUTDOWN MARGIN (SDM)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.1:
"SHUTDOWN MARGIN (SDM)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.1

This ITS Specification is based on NUREG-1431 Specification No. **3.1.1**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.1 R1	**009 R1**	RELOCATE VALUE FOR SHUTDOWN MARGIN TO COLR	Approved by NRC	Incorporated	**T.1**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**

<CTS>

3.1 REACTIVITY CONTROL SYSTEMS

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

<3.10.1.1>
<DOC LA.1>

LCO 3.1.1 SDM shall be ~~≥ [1.6]% Δk/k~~ Insert: 3.1-1-01 T.1

<3.10.1.1>
<DOC M.1>

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

<3.10.1.2>
<DOC A.3>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is ~~≥ [1.6]% Δk/k~~	24 hours

<DOC M.2>

Insert: 3.1-1-02 T.1

INSERT: 3.1-1-01

within the limits specified in the COLR.

INSERT: 3.1-1-02

within the limits specified in the COLR.

B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

Insert: B3.1-1-01

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.~~5~~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. DB.1

(continued)

INSERT: B 3.1-1-01

within the limits of LCO 3.1.5, "Shutdown Bank Insertion Limits"

BASES (continued)

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. DB.1

Insert: B3.1-2-01

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 DB.3

Insert: B3.1-2-02

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

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to

(continued)

INSERT: B 3.1-2-01

DB.1

For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis.

INSERT: B 3.1-2-02

of ≤ 225 cal/gm for nonirradiated fuel and ≤ 200 cal/gm for irradiated fuel to satisfy requirements

DB.3

BASES

APPLICABLE SAFETY ANALYSES (continued)

power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

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

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

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.

neutron flux

overtemperature ΔT

DB2

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.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

10 CFR 50.36

SDM satisfies Criterion 2 of the NRC Policy Statement. 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. PA.1

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 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. 2 3

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3 and 4 (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 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, SDM is ensured by complying with LCO 3.1.8, "Shutdown Bank Insertion Limits," and LCO 3.1.7. T.2

(Control Bank Insertion limits.) 5 6

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that

(continued)

BASES

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~~ refueling water storage tank. The operator should borate with the best source available for the plant conditions.

~~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.~~ DB.74

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.6 and LCO 3.1.7 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 MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

c. RCS average temperature; loop

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

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.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

PA.1

REFERENCES

1. 10 CFR 50, Appendix A, ~~GDC 26~~.
2. FSAR, Chapter ~~[15]~~. 14
3. ~~FSAR, Chapter [15].~~
4. 10 CFR 100.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.1:
"SHUTDOWN MARGIN (SDM)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.1 - SHUTDOWN MARGIN (SDM)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described below, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DB.2 For IP3, the accident analysis assumes that an uncontrolled rod withdrawal transient from a subcritical condition is terminated by the power range high neutron flux trip (low setting); an uncontrolled rod withdrawal transient at power is terminated by either the Overtemperature ΔT trip or the power range high neutron flux trip.

DB.3 IP3 Updated FSAR 14.2.6 establishes the requirement that a rod ejection will maintain average fuel pellet enthalpy at the hot spot below 225 cal/gm for non-irradiated fuel and 200 cal/gm for irradiated fuel. This limit is based on a review of experimental data and is intended to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves following a rod

ejection.

DB.4 Bases statements providing guidance for determining the boration flow rate are deleted because the IP3 design does not adjust the boration flow rate to compensate for core depletion.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-09 (WOG-04.1), Rev.1 , which relocated values for shutdown margin (SDM) to the COLR. SDM is a cycle-specific variable similar to moderator temperature coefficient, the rod insertion limits, axial flux difference, heat flux hot channel factor, and nuclear rise hot channel factor, which are currently contained in the COLR. In addition, there is an NRC approved methodology for determining SDM.

T.2 This change incorporates Generic Change TSTF-136, R.1 (WOG-59), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, and ISTS 3.1.2, SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F, into ISTS 3.1.1, SHUTDOWN MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.2:
"Core Reactivity"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

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

APPLICABILITY: MODE 1

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
	AND		
	A.2	Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation,a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations

BASES

BACKGROUND (continued)

and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever 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 reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Accident evaluations (Ref. 2) are, 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

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 at beginning of cycle (BOL) do not agree to within specified limits, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL, 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 BOL, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10 CFR 50.36.

BASES

LCO

This LCO requires that measured core reactivity is within ± 1% Δk/k of predicted values. During steady state power operation, this comparison includes reactor coolant system boron concentration, control rod position, reactor coolant system average loop temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration.

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODE 1 because a reactivity balance must exist when the reactor is critical and 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 MODE 2 because enough operating

BASES

APPLICABILITY (continued)

margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤5% RTP) such that reactivity anomalies are unlikely to occur. 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. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to

BASES

ACTIONS

A.1 and A.2 (continued)

provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

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 2 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made during steady state operation because other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is also performed during physics testing following refueling as an initial check on core conditions and design calculations at BOL. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value, if performed, 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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued).

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 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. As specified in a Note to the FREQUENCY, the inital performance of the SR in MODE 1 after refueling is not required until 60 EFPDs after entering MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A.

2. FSAR, Chapter 14.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.2:
"Core Reactivity"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-8	181	181	No TSCRs	No TSCRs for this Page	N/A

A.1 A.2

3.10.9 Rod Position Monitor

SEE ITS 3.1.4

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per 8 hours and after a load change greater than 10 percent of rated power.

~~3.10.10~~ Core Reactivity ~~Balance~~

LCO 3.1.2
SR 3.1.2.1

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). ~~This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration.~~ The predicted reactivity values ~~shall~~ may be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

LA.1

A.4

Note to SR 3.1.2.1

3.10.11 Notification

SEE ITS 3.2.1, 3.2.2, 3.2.3, 3.2.4

Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

~~Basis~~

~~Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR in normal operation or in short term transients.~~

~~In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant~~

A.1

Add Note to SR 3.1.2.1 Frequency — A.4

Add LCO 3.1.2, Applicability — L.1

Add Conditions A and B and associated Req Act. — L.2

Add SR 3.1.2.1, Frequency (Prior to entering Mode 1 after refuel) — A.3

Amendment No. ~~28, 61, 86, 103,~~ TSCR 97-118

TSCR 97-118 incorporated as Amendment 181. SEE ITS 3.1.4

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per 8 hours and after a load change greater than 10 percent of rated power.

3.10.10 Reactivity Balance

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3.10.11 Notification

Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR in normal operation or in short term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.2: "Core Reactivity"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.2 - Core Reactivity

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.10 requires verification that measured core reactivity is within ± 1% $\Delta k/k$ of predicted values at least once per 31 Effective Full Power Days (EFPD). The CTS Bases include the clarification that physics measurements are performed after each refueling to establish the acceptance criteria for CTS 3.10.10.

ITS SR 3.1.2.1 maintains this requirement; however, ITS SR 3.1.2.1 includes a specific requirement to verify analytic predictions "once

prior to entering MODE 1 after each refueling." The ITS 3.1.2 Bases explain that this requirement is satisfied by the successful completion of low power physics testing. This is an administrative change with no impact on safety because it is a reasonable interpretation of the existing requirement.

A.4 CTS 3.10.10 requires verification that measured core reactivity is within ± 1% $\Delta k/k$ of predicted values at least once per 31 EFPD. CTS 3.10.10 also specifies that predicted reactivity values must be adjusted (normalized) to correspond to the actual core condition before exceeding a fuel burnup of 60 EFPD after each fuel loading.

ITS SR 3.1.2.1 maintains this requirement including the normalization before 60 EFPD; however, ITS SR 3.1.2.1 includes a Note permitting the first performance of this SR to be deferred until 60 EFPD after the most recent refueling and specifies that normalization is optional.

This change is needed because the required normalization of predicted RCS boron concentration to the measured value is typically performed after reaching rated thermal power with the control rods in their normal positions for power operation. After a refueling outage, achieving the optimum conditions could take longer than the 31 EFPD Frequency specified for this SR. This change is acceptable because ITS SR 3.1.2.1 includes a specific requirement to verify measured versus predicted core reactivity before entering Mode 1 after each refueling to verify analytic predictions. Additionally, ITS LCO 3.1.1, Shutdown Margin, ensures that SDM limits are met throughout the initial 60 EFPDs of the cycle. Finally, the probability of an unrelated accident or transient occurring in the additional 30 EFPD permitted to perform this test is low.

This is an administrative change with no significant adverse impact on safety because it is a reasonable interpretation of the intent of the existing requirement.

MORE RESTRICTIVE

None

LESS RESTRICTIVE

L.1 CTS 3.10.10 requires verification that measured core reactivity is within ± 1% Δk/k of predicted values at least once per 31 EFPD. No applicability is specified and the requirement is interpreted to be applicable whenever the reactor is critical.

ITS 3.1.2 maintains the same requirement; however, the Applicability (as modified by TSTF 141 (CEOG-056)) is limited to Mode 1 (i.e., greater than 5% RTP).

This change is needed because it allows testing needed to determine the cause of a reactivity anomaly discovered during performance of this SR or discovered during a normal startup. Prohibiting operation in Mode 2 if LCO 3.1.2 is not met will eliminate the ability to investigate the cause of the reactivity anomaly further.

This change is acceptable because a substantial margin exists between operating conditions and the design limits in Mode 2 (i.e., less than 5% RTP). Additionally, the probability of an unrelated accident or transient occurring in Mode 2 while investigating a reactivity anomaly is low. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.10.10 requires verification that measured core reactivity is within ± 1% Δk/k of predicted values at least once per 31 EFPD; however, no actions are specified if this requirement is not met. Therefore, failure to meet CTS 3.10.10 is interpreted to require an immediate shutdown.

ITS LCO 3.1.2 maintains the requirement for a limit between measured and predicted core reactivity; however, Required Actions A.1, A.2, and B.1 and associated Completion Times are added to address the conditions where the measured core reactivity is not within specified limits. Specifically, Required Actions A.1, A.2, and B.1 (as modified by TSTF 142 (CEOG-058)) allow 7 days to either: Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation; or, establish appropriate operating restrictions and SRs. Otherwise, reactor power reduction to Mode 2 (See ITS 3.1.2, DOC L.1) is required.

This change is needed because a reactivity anomaly is typically indicative of incorrect analysis or assumptions and a determination and explanation of the cause of the anomaly will involve an offsite fuel analysis department and the fuel vendor. Contacting and obtaining the necessary information may require several days.

This change is acceptable because ITS LCO 3.1.1, Shutdown Margin, ensures that SDM limits are met while investigating the reactivity anomaly. Additionally, the probability of an unrelated accident or transient occurring in the 7 day period while investigating a reactivity anomaly is low. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.10.10 requires that measured core reactivity be within ± 1% $\Delta k/k$ of predicted values. CTS 3.10.10 also specifies that this comparison must include reactor coolant system boron concentration, control rod position, reactor coolant system temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. ITS SR 3.1.2.1 maintains this requirement; however, the details about what must be considered in the reactivity comparison are relocated to the LCO Section of the ITS 3.1.2 Bases.

This change is acceptable because ITS 3.1.2 maintains the requirement that measured core reactivity must be within ± 1% $\Delta k/k$ of predicted values. This change is acceptable because these details are consistent with a reasonable interpretation of the requirement to perform a reactivity balance. Additionally, ITS LCO 3.1.1 maintains the requirement to meet SDM requirements and ITS 5.6.5, Core Operating Limits Report (COLR), includes detailed requirements that ensure SDM limits will be properly established and maintained (See ITS 3.3.1, DOC LA.1).

Finally, the ITS Bases are controlled by ITS 5.5.13, Technical Specifications (TS) Bases Control Program. This program is designed to assure that changes to the ITS Bases do not result in changes to the Specification and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do

not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.2:
"Core Reactivity"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.2 - Core Reactivity

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change requires that measured core reactivity be within ± 1% $\Delta k/k$ of predicted values whenever the reactor is in Mode 1 (i.e., greater than 5% RTP) instead of the implied requirement of whenever the reactor is critical.

   This change is needed because it allows testing that might be needed to determine the cause of a reactivity anomaly discovered during performance of this SR or discovered during a normal startup. Prohibiting operation in Mode 2 if LCO 3.1.2 is not met will eliminate the ability to investigate the cause of the reactivity anomaly further.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because a substantial margin exists between operating conditions and the design limits in Mode 2 (i.e., less than 5% RTP). Additionally, the probability of an unrelated accident or transient occurring in Mode 2 while investigating a reactivity anomaly is low.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because a substantial margin exists between operating conditions and the design limits in Mode 2 (i.e., less than 5% RTP). Additionally, the probability of an unrelated accident or transient occurring in Mode 2 while investigating a reactivity anomaly is low.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

This change eliminates the implied requirement to perform an immediate plant shutdown if measured core reactivity is within ± 1% $\Delta k/k$ of predicted values. If measured core reactivity is not within specified limits, Required Actions A.1, A.2, and B.1 (as modified by TSTF 142 (CEOG-058)) allow 7 days to either: Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation; or, establish appropriate operating restrictions and SRs. Otherwise, reactor power reduction to Mode 2 (See ITS 3.1.2, DOC L.1) is required.

This change is needed because a reactivity anomaly is typically indicative of incorrect analysis or assumptions and a determination and explanation of the cause of the anomaly will involve an offsite fuel analysis department and the fuel vendor. Contacting and obtaining the necessary information may require several days.

This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because ITS LCO 3.1.1, Shutdown Margin, ensures that SDM limits are met while investigating the reactivity anomaly. Additionally, the probability of an unrelated accident or transient occurring in the 7 day period while investigating a reactivity anomaly is low.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because ITS LCO 3.1.1, Shutdown Margin, ensures that SDM limits are met while investigating the reactivity anomaly. Additionally, the probability of an unrelated accident or transient occurring in the 7 day period while investigating a reactivity anomaly is low.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.2:
"Core Reactivity"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.2

This ITS Specification is based on NUREG-1431 Specification No. **3.1.3**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
CEOG-056	**141 R0**	DELETE THE MODE 2 APPLICABILITY FOR REACTIVITY BALANCE	Approved by NRC	Incorporated	**T.2**
CEOG-058	**142 R0**	INCREASE THE COMPLETION TIME WHEN THE CORE REACTIVITY BALANCE IS NOT WITHIN LIMIT	Approved by NRC	Incorporated	**T.3**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.1**

<CTS> 3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~3~~ 2 Core Reactivity

LCO 3.1.~~3~~ 2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.
<3.10.10>

APPLICABILITY: MODE~~S~~ 1 ~~and 2~~. T.2
<DOC L.1>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit. <DOC L.2>	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	~~72 hours~~ 7 days T.3
	AND	
	A.2 Establish appropriate operating restrictions and SRs.	~~72 hours~~ 7 days T.3
B. Required Action and associated Completion Time not met. <DOC L.2>	B.1 Be in MODE ~~3~~ 2.	6 hours T.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.~~3~~ 2.1 <3.10.10>	----------NOTE---------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ----------	
<3.10.10> <DOC A.3> <DOC A.4>	Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND -----NOTE------ Only required after 60 EFPD ---------- 31 EFPD thereafter

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~3~~ 2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~$T_{avg} \geq 200°F$~~") in ensuring the reactor can be brought safely to cold, subcritical conditions. T.1

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the

(continued)

BASES

BACKGROUND (continued)

calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever 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 reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

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

(continued)

to within specified limits

PA.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of ~~cycle~~ life (BOL) (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL (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 BOL (BOC), or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL (BOC) conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% Δk/k has been established based on engineering judgment. A 1% deviation in reactivity from

Insert: B3.1-14-01

(continued)

INSERT: B 3.1-14-01

PA-1

This LCO requires that measured core reactivity is within ± 1% $\Delta k/k$ of predicted values. During steady state power operation, this comparison includes reactor coolant system boron concentration, control rod position, reactor coolant system average loop temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration.

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.

Insert: B 3.1-15-01

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of

(continued)

INSERT: B 3.1-15-01 T.2

This Specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤5% RTP) such that reactivity anomalies are unlikely to occur.

BASES

ACTIONS

A.1 and A.2 (continued)

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 ~~72 hours~~ 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. T.3

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of ~~72 hours~~ 7 days is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation. T.3

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. ~~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. 2 T.2 2

(continued)

BASES (continued)

PA.1

SURVEILLANCE REQUIREMENTS

SR 3.1.~~3~~ 2.1

during physics testing following refueling

during steady state operation because

also

BOL

if performed,

Insert: B 3.1-17-01

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, 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 SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, ~~following the initial 60 EFPD after entering MODE 1, is acceptable,~~ based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. 10 CFR 50, Appendix A, ~~GDC 26, GDC 28, and GDC 29~~.
2. FSAR, Chapter ~~[15]~~. 14

INSERT: B 3.1-17-01

As specified in a Note to the FREQUENCY, the initial performance of the SR in MODE 1 after a refueling is not required until 60 EFPDs after entering MODE 1.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.2:
"Core Reactivity"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.2 - Core Reactivity

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-136, Rev. 0 (WOG-59), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, and ISTS 3.1.2, SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F, into ISTS 3.1.1, SHUTDOWN MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR.

T.2 This change incorporates Generic Change TSTF-141 (WOG-56), Rev. 0, which deletes the required limits for Reactivity balance in Mode 2. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.3 This change incorporates Generic Change TSTF-141 (WOG-56), Rev. 0, which which increases the Completion Time When the core reactivity balance is not within limits. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.3:
"Moderator Temperature Coefficient (MTC)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ≤ 0.0 Δk/k°F at hot zero power.

APPLICABILITY: MODE 1 and MODE 2 with k_{eff} ≥ 1.0 for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. MTC not within upper limit.	A.1	Establish administrative withdrawal limits for control banks to maintain MTC 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
C. MTC not within lower limit.	C.1	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.3.2	----------NOTES---------- 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. 2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. 3. SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. ---------- Verify MTC is within lower limit.	Once each cycle

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of life (BOL) 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 BOL within the range analyzed in the plant accident analysis. The end of life (EOL) 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 EOL limit.

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

BASES

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.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

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 FSAR, Chapter 14 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 2) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be

BASES

APPLICABLE SAFETY ANALYSES (continued)

evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

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 BOL or EOL. 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 BOL and EOL. An EOL measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOL value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of 10 CFR 50.36. 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 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOL; this upper bound must not be exceeded. This maximum upper limit occurs at BOL, all rods out (ARO), hot zero power conditions. At EOL 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.

BASES

LCO (continued)

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOL and EOL on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOL positive limit and the EOL 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.

APPLICABILITY

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

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

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

BASES

ACTIONS

A.1 (continued)

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOL 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 EOL MTC limit means that the safety analysis assumptions for the EOL accidents that use a bounding negative MTC value may be invalid. If the EOL MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

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 3.1.3.1

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

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

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOL 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 EOL LCO limit. The 300 ppm SR value is sufficiently less negative than the EOL LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that include the following requirements:

1. This SR is not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

2. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOL limit on MTC could be reached before the planned EOL. Because the MTC changes slowly with core

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOL limit.

3. 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 EOL limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. 10 CFR 50, Appendix A.

2. FSAR, Chapter 14.

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

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.3:
"Moderator Temperature Coefficient (MTC)"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.1-25	149	149	No TSCRs	No TSCRs for this Page	N/A

Add LCO 3.1.3, Lower limit and associated Applicability and Required Action C.1 — A.1 A.2

Add Conditions A and B and associated Req. Act. — M.1

SEE ITS 3.1.8 — L.1

e. ~~MINIMUM CONDITIONS FOR CRITICALITY~~

LCO 3.1.3 Applicability

1. Except during low power physics test the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive. — A.3

~~2. This section intentionally deleted.~~

SEE ITS 3.1.8 ITS 3.4.2

3. At all times during critical operation, the lowest loop T_{avg} shall be no lower than 540 °F.
   a. If T_{avg} is less than 540°F when the reactor is critical, restore T_{avg} to $\geq$ 540 °F within 15 minutes or be in hot shutdown within the following 15 minutes.

SEE ITS 3.4.9

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. [1] [2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range. [1] [2] Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions. — A.1

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of an increase in moderator temperature. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical except when T_{avg} is $\geq$ 540 °F provides assurance that an overpressure event will not occur whenever the reactor vessel is in the nil-ductility temperature range and that the reactor is operated within the bounds of the safety analyses. The safety analyses, which assume a critical temperature of 547 °F, are applicable for critical temperatures as low as 540 °F. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. The Surveillance requirement to support this specification is provided in Table 4.1-1 item no. 4.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the reactor coolant not be solid when criticality is achieved.

References:

~~1. FSAR Table 3.2-1~~
~~2. FSAR Figure 3.2-9~~

A.1

Add SR 3.1.3.1 — M.2

Add SR 3.1.3.2 — M.3

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.3:
"Moderator Temperature Coefficient (MTC)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.3 - Moderator Temperature Coefficient (MTC)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.1.C.1 requires that the reactor not be made critical "at any temperature above which the moderator temperature coefficient is positive."

ITS LCO 3.1.3 maintains this requirement with the requirement that the Moderator Temperature Coefficient (MTC) maximum upper limit must be $\leq 0.0\ \Delta k/k°F$ at hot zero power. This is an administrative change with

no adverse impact on safety because there is no change to the existing requirement.

MORE RESTRICTIVE

M.1 CTS 3.1.C.1 requires that the reactor not be made critical at any temperature above which the moderator temperature coefficient is positive; however, there is no limit specified for the MTC lower limit.

ITS LCO 3.1.3 maintains this requirement for a maximum upper limit on MTC (See ITS 3.1.3, DOC A.3); however, ITS LCO 3.1.3 includes a new requirement for a lower limit for MTC with the limit specified in the Core Operating Limits Report (COLR). In conjunction with this change, ITS LCO 3.1.3 specifies that the lower limit for MTC is Applicable in Modes 1, 2 and 3. Additionally, ITS LCO 3.1.3, Condition C and associated Required Actions, will require that the plant be in Mode 4 within 12 hours if the MTC lower limit is not met.

These changes are needed because the analyses of accidents that cause core overcooling (e.g., feedwater injection event or loss of feedwater heating) include assumptions regarding the lower (more negative) limit for MTC.

This change is acceptable because the MTC lower limit specified in the COLR and the Applicability for the MTC lower limit (Modes 1, 2 and 3) are consistent with assumptions regarding initial conditions in the analysis of accidents that cause core overcooling. Additionally, Required Actions that promptly place the plant outside of the applicable Modes if limits are not met is appropriate when analysis assumptions are not met. The 12-hour 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.

These more restrictive changes are acceptable because they do not introduce any operation that is un-analyzed while establishing formal requirements for meeting analysis assumptions regarding MTC lower limits. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.1.C.1 requires that the reactor not be made critical at any temperature above which the MTC is positive; however, there is no explicit requirement for verification of the MTC maximum upper limit following refueling that limits are met. CTS 3.1.C.1 Bases state that suitable physics measurements of MTC will be made as part of the startup program to verify analytic predictions.

ITS SR 3.1.3.1 is added to require verification that MTC is within the upper limit once before entering Mode 1 after each refueling.

This change is needed because it establishes a formal requirement to verify that MTC is within the upper limit immediately following refueling consistent with the existing CTS Bases assumption that the startup program will verify analytic predictions regarding MTC.

This change is acceptable because MTC is most positive at the beginning of a core cycle because of the relative abundance of fuel. Therefore, verification that MTC is within required limits once before entering Mode 1 after each refueling ensures that MTC will be within the required upper limit throughout that core cycle.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring verification that MTC is within the upper limit once before entering Mode 1 after each refueling. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.1.C.1 does not specify any requirements for the MTC lower limit. Therefore, ITS LCO 3.1.3 includes a new requirement for a lower limit for MTC with the limit specified in the Core Operating Limits Report (COLR)(See ITS 3.1.3, DOC M.1). In conjunction with this change, ITS SR 3.1.3.2 (as modified by TSTF 13, Rev. 1 (WOG-04.5)) adds a new requirement to verify MTC is within the lower limit once each cycle after exceeding the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. Periodic re-verification of the MTC lower limit is required if the initial verification indicates that the MTC lower limit could be exceeded before the end of core life.

This change is needed because it establishes a formal requirement to

verify that MTC is within the lower limit.

This change is acceptable because MTC is most negative at the end of a core cycle because of fuel depletion. Therefore, verification that MTC is within required limits once at a point relatively late in the core cycle ensures that MTC will be above the required lower limit during the final portion of the core cycle. The criteria used to trigger this verification is core life reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm. If the verification performed at the 300 ppm equilibrium RTP ARO boron concentration equivalent indicates that MTC is more negative than predicted, then there is a potential that MTC may become more negative than permitted by the LCO before the end of core life and the SR must be repeated periodically until EOL. If MTC is within limits at 60 ppm, SR does not have to be performed again because there is a high degree of assurance that the EOL limit will not be exceeded prior to EOL.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring verification that MTC is within the lower limit at a point in core life that ensures the limit will be maintained until the next refueling.

LESS RESTRICTIVE

L.1 CTS 3.1.C.1 requires that the reactor not be made critical at any temperature above which the MTC is positive; however, no actions or Completion Times are specified if this requirement is not met. Therefore, failure to meet CTS 3.1.C.1 is construed to require an immediate shutdown.

ITS LCO 3.1.3 maintains the requirement for an upper limit for MTC; however, Required Action A.1 and B.1 and associated Completion Times are added to address the condition where the upper limit for MTC is not within the specified limit. Specifically, Required Action A.1 requires establishing administrative withdrawal limits for control banks within 24 hours. Otherwise, Required Action B.1 requires that the plant be placed outside the LCO Applicability within the following 6 hours.

DISCUSSION OF CHANGES
ITS SECTION 3.1.3 - Moderator Temperature Coefficient (MTC)

This change is needed because MTC becomes more negative with control bank insertion because of the resulting decreased boron concentration. Therefore, this action allows the LCO to be met with minimum impact on plant operations.

This change is acceptable because MTC becomes more negative as control banks are inserted causing the MTC upper limit to be met. The 24 hour Completion Time provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits. Limiting the amount of time that MTC may not be within the upper limit minimizes the probability that an unrelated accident or transient dependent upon the MTC upper limit will occur while MTC is not within required limits. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.3:
"Moderator Temperature Coefficient (MTC)"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.3 - Moderator Temperature Coefficient (MTC)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   ITS LCO 3.1.3 maintains the requirement for an upper limit for MTC; however, Required Action A.1 and B.1 and associated Completion Times are added to address the condition where the upper limit for MTC is not within the specified limit. Specifically, Required Action A.1 requires establishing administrative withdrawal limits for control banks within 24 hours. Otherwise, Required Action B.1 requires that the plant be placed outside the LCO Applicability within the following 6 hours. CTS does not specify any actions or completion times for failure to meet the requirement for an upper limit for MTC.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because MTC becomes more negative as control banks are inserted causing the MTC upper limit to be met. The 24 hour Completion Time the upper provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits the upper. Limiting the amount of time that MTC may not be within the upper limit minimizes the probability that an unrelated accident or transient dependent upon the MTC upper limit will occur while MTC is not within required limits. Therefore, this change has no significant adverse impact on safety.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant

operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because MTC becomes more negative as control banks are inserted causing the MTC upper limit to be met. The 24 hour Completion Time the upper provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits the upper. Limiting the amount of time that MTC may not be within the upper limit minimizes the probability that an unrelated accident or transient dependent upon the MTC upper limit will occur while MTC is not within required limits.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.3:
"Moderator Temperature Coefficient (MTC)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.3

This ITS Specification is based on NUREG-1431 Specification No. **3.1.4**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.5 R1	**013 R1**	MOVE SR FOR 300 PPM MTC MEASUREMENT TO FREQUENCY NOTE OF SR 3.1.4.3	Approved by NRC	Incorporated	**T.2**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.1**

3.1 REACTIVITY CONTROL SYSTEMS

<CTS>

3.1.~~4~~ 3 Moderator Temperature Coefficient (MTC)

<3.1.C.1>

LCO 3.1.~~4~~ 3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ~~1~~ ≤ [ ] 0.0 Δk/k°F at hot zero power~~]~~ ~~[that specified in Figure 3.1.4-1]~~. DB.1

<DOC A.3>
<DOC M.1>

<3.1C.1> APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

<DOC M.1>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC 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
C. MTC not within lower limit.	C.1 Be in MODE 4.	12 hours

<DOC L.1>
<DOC L.1>
<DOC M.1>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.~~4~~ 3.1 <DOC M.2>	Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
~~SR 3.1.4.2~~ 3	~~Verify MTC is within 300 ppm Surveillance limit specified in the COLR.~~	~~-----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 each cycle~~ T.2 T.2
Insert: 3.1-6-01		
SR 3.1.~~4~~ 3.~~3~~ 2 3 <DOC M.3>	----------NOTES---------- ~~1.~~ 2 If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.~~4~~ 4.~~2~~ shall be repeated once per 14 EFPD during the remainder of the fuel cycle. 3 2 ~~2.~~ 3 SR 3.1.~~4~~ 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. ----------	~~-----NOTE----- Not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm ----------~~ T.2
	Verify MTC is within lower limit.	Once each cycle

INSERT: 3.1-6-01:

1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

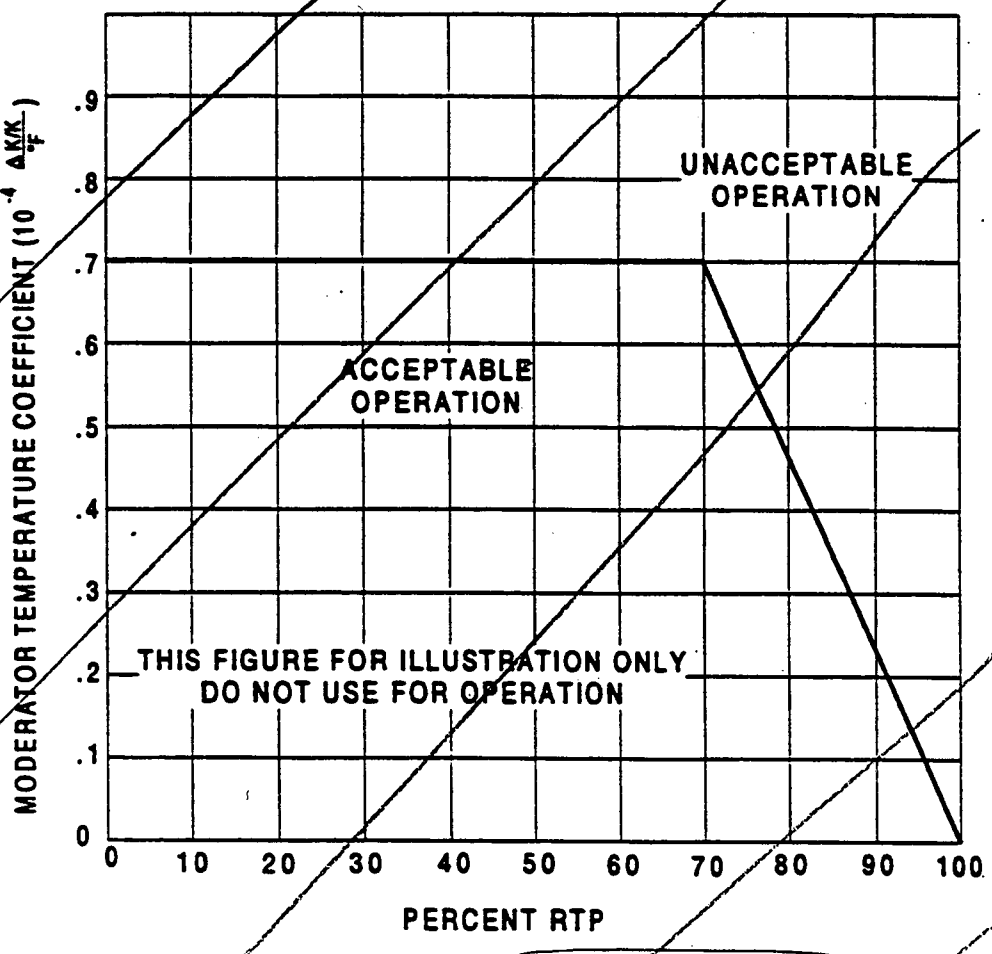

Figure 3.1.4-1 (page 1 of 1) 3
Moderator Temperature Coefficient vs. Power Level

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~1~~3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than 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 limit.

BOL

life

EOL

EOL

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

B 3.1.3-1

(continued)

BASES

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.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

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.

14 The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

2

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,

(continued)

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). BOL or EOL BOL and EOL EOL

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. EOL

10 CFR 50.36

MTC satisfies Criterion 2 of 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 3.1.4 3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded. BOL EOL

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met. BOL

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

(continued)

BASES

LCO (continued)

advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

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

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

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

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

(continued)

BASES

ACTIONS (continued)

B.1

BOL

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

EOL

Exceeding the ~~EOC~~ MTC limit means that the safety analysis assumptions for the ~~EOC~~ accidents that use a bounding negative MTC value may be invalid. If the ~~EOC~~ MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

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.~~4~~ 3.1

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

BOL

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.~~4~~3.2 ~~and SR 3.1.4.3~~

In similar fashion, the LCO demands that the MTC be less negative than the specified value for ~~EOC~~ EOL 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~~ EOL LCO limit. The 300 ppm SR value is sufficiently less negative than the ~~EOC~~ EOL LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.~~4~~3.~~3~~2 is modified by ~~a Note~~ three Notes that includes the following requirements:

Insert: B3.1-23-01

~~a~~. If the 300 ppm Surveillance limit is exceeded, it is possible that the ~~EOC~~ EOL limit on MTC could be reached before the planned ~~EOC~~ EOL. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the ~~EOC~~ EOL 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~~ EOL limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. 10 CFR 50, Appendix A, ~~GDC 1~~.
2. FSAR, Chapter ~~[18]~~ 14.
3. WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
~~4. FSAR, Chapter [15].~~

INSERT: B 3.1-23-01:

a ~~1~~. This SR is 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.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.3:
"Moderator Temperature Coefficient (MTC)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.3 - Moderator Temperature Coefficient (MTC)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-136, Rev.1 (WOG-59), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, and ISTS 3.1.2, SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F, into ISTS 3.1.1, SHUTDOWN MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR.

T.2 This change incorporates Generic Change TSTF-013, Rev.1 (WOG-04.5), which deleted ITS SR 3.1.3.2 (MTC measurement at 300 ppm), and replaced it with a note in ITS SR 3.1.3.3 (now ISTS SR 3.1.3.2). The Bases for ISTS 3.1.4 was revised to reflect the change.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.4:
"Rod Group Alignment Limits"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with rod group alignment limits as follows:

a. When THERMAL POWER is > 85% RTP,

1. Groups with step counter demand position ≤ 212 steps shall have all individual indicated rod positions ≤ ± 12 steps of their group step counter demand position; and

2. Groups with step counter demand position > 212 steps shall have all individual indicated rod positions ≤ +17 steps and -12 steps of their group step counter demand position

b. When THERMAL POWER is ≤ 85% RTP, all individual indicated rod positions shall be ≤ ± 18 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One rod not within alignment limits.	B.1	Restore rod to within alignment limits.	1 hour
	OR		
	B.2.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
		OR	
	B.2.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND		
	B.2.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	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		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B (continued)	B.2.6	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 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 specified in the COLR.	1 hour
	OR		
	D.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	AND		
	D.2	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	---------- NOTE ---------- Not required to be performed for individual control rods until 1 hour after completion of control rod movement. ---------- 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 one 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 loss of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

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 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).

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 have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more

BASES

BACKGROUND (continued)

RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs may consist of two groups that are moved in a staggered fashion, but always within one step of each other. IP3 has four control banks and four 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 at the desired position. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Individual Rod Position Indication (IRPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is 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 IRPI System provides an indication of actual control rod position, but at a lower precision than the step counters. This

BASES

BACKGROUND (continued)

system is based on inductive analog signals from a coil stack located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained. The rod position maximum uncertainty is ± 12 steps (± 7.5 inches). Misalignment limit of 12 steps precludes a rod misalignment of > 15 inches when instrument error is considered. An indicated misalignment limit of 18 steps precludes a rod misalignment of > 18.75 inches when instrument error is considered.

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 are that:

a. There 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

BASES

APPLICABLE SAFETY ANALYSES (Continued)

meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment. With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 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 also fully withdrawn.

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

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

BASES

APPLICABLE SAFETY ANALYSES (Continued)

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36.

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

To ensure that individual rods are properly aligned with its associated group step counter demand position, the following limits are placed on individual rod positions:

When THERMAL POWER is > 85% RTP,

1. Groups with step counter demand position ≤ 212 steps shall have all individual indicated rod positions within 12 steps of their group step counter demand position; and

2. Groups with step counter demand position > 212 steps shall have all individual indicated rod positions ≤ +17 steps and -12 steps of their group step counter demand position

When THERMAL POWER is ≤ 85% RTP, all individual indicated rod positions shall be ≤ ± 18 steps of their group step counter demand position.

These limits ensure analysis assumptions for SDM and peaking factors are met because an indicated misalignment of 12 steps precludes a rod misalignment of > 15 inches when instrument error is considered. An indicated misalignment limit of 18 steps precludes a rod misalignment of > 18.75 inches when instrument error is considered.

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 (Ref. 4).

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 typically 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

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Required Actions A.1.1 and A.1.2 apply if either SR 3.1.4.2 or SR 3.1.4.3 are not met. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

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 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 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

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. If all individual indicated rod positions are ≤ ± 18 steps of their group step counter demand position, the LCO may be met by reducing reactor power ≤ 85% RTP.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.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 20 steps from the top of the core would require a significant power reduction, since

BASES

ACTIONS

B.2.1.1 and B.2.1.2 (continued)

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

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

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. 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.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to

BASES

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

The analysis specified by Required Action B.2.6 must address the potential ejected rod worth, non-uniform fuel depletion, associated transient power distribution peaking factors and accidents. The following issues must also be addressed:

a. Rod cluster control assembly insertion characteristics;

b. Rod Cluster Control Assembly Misalignment;

c. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates the emergency core cooling system;

d. Single rod cluster control assembly withdrawal at full power;

e. Major reactor coolant system pipe ruptures (loss of coolant accident);

f. Major Secondary system pipe rupture; and

g. Rupture of a control rod drive mechanism housing.

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 obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

BASES

ACTIONS (continued)

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 initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 (continued)

the operator to detect a rod that is beginning to deviate from its expected position. Rod position may be verified using normal indication, direct readings using a digital voltmeter, or the plant computer. 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. This SR is not required to be performed for a control rod until 1 hour after completion of movement of that rod. This allowance is needed because it provides time for thermal stabilization of rod position instrumentation. This allowance is acceptable because individual rod position indicators may not accurately reflect control rod position prior to thermal stabilization and there is a presumption that individual control rods will move with their group.

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 in a single direction will not cause radial or axial power tilts, or oscillations, to occur. This SR requires that control rods be inserted or withdrawn by at least 10 steps which is sufficient to ensure that rod movement can be confirmed by individual rod position indicators. Administrative controls and Technical Specification limits ensure that control rod insertion limits are met. 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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.2 (continued)

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) 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 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.

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 was performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A.

2. 10 CFR 50.46.

3. FSAR, Chapter 14.

4. WCAP-14668, Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3, October 1996 (Proprietary).

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.4:
"Rod Group Alignment Limits"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-5	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-6	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-7	160	160	No TSCRs	No TSCRs for this Page	N/A
3.10-8	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-10	180	180			
3.10-16	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-17	103	103	No TSCRs	No TSCRs for this Page	N/A
T 4.1-1(2)	169	169	No TSCRs	No TSCRs for this Page	N/A
T 4.1-3(1)	178 TSCR 97-156, 98-043	178 TSCR 97-156, 98-043	IPN 98-043	Instrument Channel Surveillance Intervals Extended to 24 Months	Incorporated
T 4.1-3(1)	178 TSCR 97-156, 98-043	178 TSCR 97-156, 98-043	IPN 97-156	SR Freq for Main Turbine Stop and Control Valves	Incorporated

SR 3.1.4.1 ~~3.10.3.3~~	~~The rod position indicators shall be monitored and logged~~ once each shift (12 hours) to verify rod position within each bank assignment.	A.10 A.9
SEE ITS 3.2.4	3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.	
SEE ITS 3.1.5	3.10.4 Rod Insertion Limits 3.10.4.1 The shutdown rods shall be fully withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of Specification 3.10.1).	
SEE ITS 3.1.6	3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.	
3.1.4 Req. Act. A.1.1, A.1.2 B.2.1.1, B.2.1.2, B.2.3 D.1.1, D.1.2	3.10.4.3 ~~Control bank insertion shall be further restricted if:~~ ~~a) The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown.~~ ~~b) A rod is inoperable (Specification 3.10.7).~~	A.6
SEE ITS 3.1.6	3.10.4.4 Control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin required by Specification 3.10.1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one control rod inserted.	
SEE ITS 3.1.8		

Add Req. Act B.2.3 — M.2

Amendment No. ~~74~~, ~~88~~, ~~103~~, 112

Applicability: Mode 1 and 2 — A.4

ITS 3.1.4

3.10.5 Rod Misalignment Limitations — 12 hours — A.9 — A.1 A.2

3.10.5.1 At least once ~~per shift~~ (allowing one hour for thermal soak after rod motion) the position of each control or shutdown rod shall be determined: — A.14

SR 3.1.4.1
SR 3.1.4.1, Note

a. For operation less than or equal to 85% of rated thermal power, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be less than or equal to 18 steps. A control or shutdown rod indicting a misalignment greater than 18 steps shall be realigned within one hour or the core peaking factors shall be determined within two hours and the requirements of Specification 3.10.2 applied. — A.7 — 72 — M.3

LCO 3.1.4.b
Req Act B.1
Req Act B.2.4, B.2.5

b. For operation greater than 85% of rated thermal power, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be ±12 steps for less than or equal to 212 steps and +17, -12 steps for greater than 212 steps. A control or shutdown rod indicating a misalignment greater than the above mentioned steps shall be realigned within one hour or the core peaking factors shall be determined within two hours and the requirements of Specification 3.10.2 applied. — A.7 — 72 — M.3

LCO 3.1.4.a
Req Act B.1
Req Act B.2.4, B.2.5

Req Act B.2.1.1, B.2.1.2, B.2.3 — Verify SDM in 1 hr / 12 hours thereafter — M.2

3.10.5.2 If the requirements of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value. — A.8 — L.1 — within 2 hr — reactor power — 75% — M.4

Req Act B.2.2

3.10.5.3 If the misaligned control rod is not realigned within 8 hours the rod shall be declared inoperable. — M.3 — M.5

Req Act A.2, C.1, D.2

SEE ITS 3.1.7

3.10.6 Inoperable Rod Position Indicator Channels

3.10.6.1 If a rod position indicator channel is out of service, then:

a. For operation between 50 percent and 100 percent of rating, the position of the control rod shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) once per 8 hours, or subsequent to rod motion exceeding 24 steps, whichever occurs first.

b. During operation below 50 percent of rating, no special monitoring is required.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3.10.6.3 If a control rod having a rod position indicator channel out of service, is found to be misaligned from 3.10.6.1a above, then Specification 3.10.5 will be applied.

Amendment No. 29, 103, 176, 181

A.1 A.2

No — M.1

3.10.7 Inoperable Rod Limitations — A.3

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.8 or fails to meet the requirements of 3.10.8.

LCO 3.1.4 Applicability — Mode 1 and 2 — A.4

~~3.10.7.2~~ Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the ~~hot shutdown condition~~ Mode 3 in 6 hours. — A.12 — A.5

Req Act A.2
Req Act D.2

3.10.7.3 If any rod has been declared ~~inoperable~~ misaligned, then the potential ejected rod worth, associated transient power distribution peaking factors and the accident listed in Table 3.10-1 shall be analyzed within 5 days, or the reactor brought to the hot shutdown condition using normal operating procedures. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis. — LA.1 — LA.1

Req. Act B.2.6

Req. Act C.1

Mode 3 in 6 hours — A.5

3.10.8 Rod Drop Time — ≥ 500 °F — all RCPs running — A.13

SR 3.1.4.3 At ~~operating temperature~~ and ~~full flow~~, the drop time to each control rod shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. — M.6

3.10.9 Rod Position Monitor

LA.2

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per 8 hours and after a load change greater than 10 percent of rated power.

3.10.10 Reactivity Balance

SEE ITS 3.1.2

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

SEE ITS 3.2.1, 3.2.2 3.2.3, 3.2.4

3.10.11 Notification

Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR in normal operation or in short term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant

A.1

(e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^N$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available. When a measurement of $F_{\Delta H}^N$ is taken, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the group step counter demand position (operating at greater than 85% of rated thermal power with no accounting for peaking factor margin), or 18.75 inches (operating at less than or equal to 85% of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment greater than 15 inches with consideration of instrumentation error and 18 steps indicated misalignment corresponds to 18.75 inches with instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

A.1

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequency over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The assumed control rod drop time in the safety analysis is 2.7 seconds, consisting of 1.80 seconds for normal rod drop time plus additional margin which includes a seismic allowance. The required control rod drop time in Section 3.10.8 is therefore consistent with that assumed in the safety analysis.

REFERENCE

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program," August 1975
2. FSAR Appendix 14C
3. Letter from J.P. Bayne to S.A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."
4. WCAP-14668, "Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3," October 1996 (Proprietary).

A.1

LA.1

TABLE 3.10-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

12 hours — A.9

SEE ITS 3.1.7

LA.2

TABLE 4.1-1 (Sheet 2 of 6)

	Channel Description	Check	Calibrate	Test	Remarks
	8. 6.9 KV Voltage	N.A.	18M	Q	Reactor protection circuits only
	6.9 KV Frequency	N.A.	24M	Q	Reactor protection circuits only
SR 3.1.4.1	9. Analog Rod Position	S	24M	M	
	10. Steam Generator Level	S	24M	Q	
	11. Residual Heat Removal Pump Flow	N.A.	24M	N.A.	
	12. Boric Acid Tank Level	S	24M	N.A.	Bubbler tube rodded during calibration
	13. Refueling Water Storage Tank Level				
	a. Transmitter	W	18M	N.A.	Low level alarm
	b. Indicating Switch	W	6M	N.A.	Low level alarm
	14a. Containment Pressure - narrow range	S	24M	Q	High and High-High
	14b. Containment Pressure - wide range	M	18M	N.A.	
	15. Process and Area Radiation Monitoring:				
	a. Fuel Storage Building Area Radiation Monitor (R-5)	D	24M	Q	
	b. Vapor Containment Process Radiation Monitors (R-11 and R-12)	D	24M	Q	
	c. Vapor Containment High Radiation Monitors (R-25 and R-26)	D	24M	Q	
	d. Wide Range Plant Vent Gas Process Radiation Monitor (R-27)	D	24M	Q	

SEE CTS MASTER MARKUP

Amendment No. 8, 38, 65, 68, 74, 93, 107, 125, 137, 140, 144, 148, 150, 154, 169

TABLE 4.1-3 (Sheet 1 of 2)

	FREQUENCIES FOR EQUIPMENT TESTS		
		Check	Frequency
SR3.1.4.3	~~1.~~ Control Rods	Rod drop times of all control rods	~~24M~~ Prior to critical after head removed (M6)
SR 3.1.4.2	2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every ~~31~~ 92 days ~~during reactor critical operations~~ (L.2) (A.4) except fully inserted rods (A.11)
	3. Pressurizer Safety Valves	Set Point	24M*
	4. Main Steam Safety Valves	Set Point	24M
	5. Containment Isolation System	Automatic actuation	24M
	6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
	7. Primary System Leakage	Evaluate	5 days/week
	8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
	9. Turbine Steam Stop Control Valves	Closure	Yearly
	10. L.P. Steam Dump System (6 lines)	Closure	Monthly
	11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Quarterly
	12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

* ~~Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.~~ Deleted by TSCR 97-156

Insert from TSCR 98-043.

TSCR 97-156

TSCR 98-043

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M*
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop And Control Valves	Closure	Not to exceed 6 months**
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Monthly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

* ~~Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.~~ TSCR 97-156

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.

Amendment No. ~~10~~, ~~14~~, ~~47~~, ~~65~~, ~~97~~, ~~99~~, ~~125~~, ~~126~~, ~~127~~, ~~129~~, ~~137~~, ~~144~~, ~~165~~, TSCR 98-043, TSCR 97-156

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.4:
"Rod Group Alignment Limits"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.4 - Rod Group Alignment Limits

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.7 defines an inoperable rod as any of the following: a rod that does not trip; or, a rod declared inoperable under CTS 3.10.5 (i.e., rod group alignment not within specified limits); or a rod that fails to meet CTS 3.10.8 (i.e., rod drop time not within limits).

ITS LCO 3.1.4 requires that rods are Operable and within specified alignment limits. A rod is Operable if it has freedom of movement

(meets ITS SR 3.1.4.2) and the rod drop time is within specified limits (meets ITS SR 3.1.4.3). A rod is within specified alignment limits if it is within 12 steps of the group step counter demand position when > 85% RTP (with an additional allowance for rods in groups withdrawn >212 steps) and within 18 steps when ≤ 85% RTP (meets ITS LCO 3.1.4 and SR 3.1.4.1).

This change is needed because it establishes consistent terminology between the ITS LCO 3.1.4 requirements (rod Operability and alignment), the ITS SRs that verify these LCO requirements are met, and the Conditions and Required Actions that are applicable if these requirements are not met. This is an administrative change with no significant adverse impact on safety because it clarifies existing requirements. Any differences between CTS and ITS are identified and justified elsewhere.

A.4 CTS 3.10.7 specifies that requirements for control rod Operability (and alignment) apply whenever the reactor is critical. Additionally, CTS Table 4.1-3, Item 2, specifies that verification of control rod movement is required to be performed only when the reactor is critical.

ITS LCO 3.1.4, Applicability, specifies that requirements for control rod Operability and alignment apply in Modes 1 and 2 (i.e., Keff >0.99).

This change is needed because control rod Operability and alignment are implicit assumptions during any approach to criticality as well as when the reactor is critical. Under CTS, control rod Operability requirements are imposed as soon as action is initiated to make the reactor critical (i.e., entry into Mode 2). Otherwise, the CTS LCO is not met as soon as the reactor is critical. Therefore, this is an administrative change with no significant adverse impact on safety.

A.5 CTS 3.10.7.2 specifies that if there is more than one inoperable rod (See ITS 3.1.4, DOC M.1), then the reactor must be placed in hot shutdown (i.e., Mode 3) with no completion time specified. Similarly, CTS 3.10.7.3 specifies that if there is an inoperable (i.e., misaligned) rod and specified actions are not completed, then the reactor must be

brought to the hot shutdown condition (i.e., Mode 3) using normal operating procedures.

Under the same conditions, ITS LCO 3.1.4, Required Actions C.1 and D.2, require that the plant be in Mode 3 within 6 hours. This is an administrative change with no significant adverse impact on safety because the 6-hour 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.

A.6 CTS 3.10.7.2 specify Actions for control rods that are not Operable (i.e., not trippable or slow). Additionally, if rods are not Operable, then the Actions in CTS 3.10.4.3 (additional restrictions on insertion limits) are also applicable because control rod insertion limits are invalidated by the inoperable or misaligned rod.

Under the same conditions (rods not within alignment limits and/or not trippable or slow), ITS LCO 3.1.4, Required Actions A.1.1, A.1.2, require verification that SDM requirements are met or the initiation of boration to restore SDM to within limits.

This is an administrative change because insertion limits are established to ensure that SDM requirements are met and rod insertion is changed by the initiation of boration. Therefore, CTS 3.10.4.3 and the ITS LCO 3.1.4, Required Actions, both require that rods are withdrawn more than required by the insertion limits specified in the COLR to account for the scram reactivity insertion lost due to an untrippable, slow or misaligned rod. Therefore, this is an administrative change with no significant adverse impact on safety.

A.7 CTS 3.10.5.1.a and CTS 3.10.5.1.b require that core peaking factors be determined when a rod is determined not to be within required alignment limits. CTS 3.10.2, Power Distribution Limits, defines the peaking factors as $F_Q(Z)$ and $F^N_{\Delta H}$. Under the same conditions (rod not within alignment limits), ITS 3.1.4, Required Actions B.2.4 and B.2.5, require performance of SR 3.2.1.1 and SR 3.2.2.1.

ITS SR 3.2.1.1 requires verification that Heat Flux Hot Channel Factor ($F_Q(Z)$) is within required limits and ITS SR 3.2.2.1 requires verification that Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)is within required limits. Therefore, this is an administrative change with no significant adverse impact on safety.

A.8 CTS 3.10.5.2 specifies that requirements to reduce the reactor flux trip setpoint (See ITS 3.1.4, DOC L.1) as compensatory action for a misaligned rod do not apply if Quadrant Power Tilt Limits in CTS 3.10.3 are not met and Actions are being taken in accordance with CTS 3.10.3. CTS 3.10.3 requires restricting reactor power and reducing the high flux trip setpoint 3% RTP for every 1% that indicated power tilt exceeds 1.0.

ITS LCO 3.1.4 does not retain this allowance. This change is needed because the requirements of ITS LCO 3.1.4, Rod Group Alignment Limits, and ITS LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR), are both applicable. These LCOs are established for different reasons and, therefore, have different although similar Required Actions. Specifically, a reduction in power level is appropriate compensatory action for failure to meet rod group alignment limits (See ITS LCO 3.1.4, DOC L.1); whereas, a reduction in the high neutron flux trip setpoint is appropriate compensatory action for exceeding QPTR limits (See ITS 3.2.4).

This change is acceptable because under CTS the requirements of CTS 3.10.3 for exceeding QPTR are more restrictive than the CTS 3.10.5.2 requirements for a misaligned rod. This is an administrative change with no significant adverse impact on safety because any differences between CTS and ITS requirements for exceeding rod alignment limits or QPTR limits are identified and justified elsewhere in this conversion package.

A.9 CTS 3.10.3.3, CTS 3.10.5.1, and CTS Table 4.1-1, Item 9, each require verification individual rod positions are within alignment limit every shift.

ITS SR 3.1.4.1 maintains this requirement with a specified Frequency of every 12 hours. This is an administrative change with no significant

adverse impact on safety because the notes to CTS Table 4.1-1 (as modified by TSCR IPN 97-118) specify that an SR frequency specified as once per shift must be performed at least once per 12 hours.

A.10 CTS 3.10.3.3 requires that the rod position indicators are monitored and logged as part of verification of rod position required every shift.

ITS SR 3.1.4.1 maintains the requirement to verify individual rod positions are within alignment limits; however, the requirement to log the results as part of verification is deleted.

This change is needed because ITS does not include requirements for documenting the results of any other required SRs. This change is acceptable because documenting SR results is not an essential element for ensuring LCO requirements are met. This is an administrative change with no significant adverse impact on safety because requirements for documenting the results of required SRs are governed quality assurance and other administrative programs.

A.11 CTS Table 4.1-3, Item 2, requires periodic verification that rods can be moved at least 10 steps in either direction. ITS SR 3.1.4.2 maintains this requirement; however, ITS SR 3.1.4.2 excludes any control rod is fully inserted into the core. This change is acceptable because the purpose of the SR is to verify that rods will fully insert into the core when tripped. This is an administrative change with no significant adverse impact on safety because there is no need to verify that rods already fully inserted are capable of being inserted.

A.12 CTS 3.10.7.2 specifies that not more than one inoperable (i.e., misaligned) control rod is allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. ITS LCO 3.1.4 does not include this exception to rod alignment limits for physics testing because the allowance is provided by ITS LCO 3.1.8, Physics Test Exceptions.

A.13 CTS 3.10.8 specifies that rod drop time testing must be performed at operating temperature and full flow. ITS SR 3.1.4.3 specifies that rod drop time testing must be performed with all RCPs operating and the average moderator temperature ≥ 500°F. Both CTS and ITS intend that the SR is performed under conditions designed to simulate a reactor trip under actual conditions.

This change is needed because it more precisely defines the plant conditions that simulate a reactor trip under actual conditions. This is an administrative change with no significant adverse impact on safety because ITS SR 3.1.4.3 is a reasonable interpretation of the existing requirement.

A.14 CTS 3.10.5.1 specifies that the required verification individual rod positions should allow "one hour for thermal soak after rod motion." ITS SR 3.1.4.1 maintains the requirement to verify individual rod positions are within alignment limits. Additionally, a Note to ITS SR 3.1.4.1 specifies that this SR is not required to be performed for individual control rods until 1 hour after completion of control rod movement. This allowance is needed because it provides time for thermal stabilization of rod position instrumentation. This allowance is acceptable because individual rod position indicators may not accurately reflect control rod position prior to thermal stabilization and there is a presumption that individual control rods will move with their group. This is an administrative change with no adverse impact on safety because there is no change to the existing requirement.

MORE RESTRICTIVE

M.1 CTS 3.10.7.2 specifies that not more than one inoperable rod shall be allowed. Therefore, CTS 3.10.7.2 permits continued operation with one rod that is slow or not trippable and/or not within alignment limits.

ITS 3.1.4 allows continued plant operation with one misaligned rod (ITS LCO 3.1.4, Condition B) if specified requirements are met (See ITS 3.1.4, DOCs M.2 and M.3); however, ITS 3.1.4, Required Action A.2, requires the plant be shutdown within 6 hours if one or more rods are

not trippable or slow.

This change is needed because there is a possibility that the required scram reactivity insertion rate and/or SDM assumed in the accident analysis may not be met. This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring a more conservative response than is currently required when the potential exists that required SDM and/or the scram reactivity insertion rate assumed in the accident analysis may not be met. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.10.5.1.a and CTS 3.10.5.1.b specify the Required Actions for a misaligned control rod which include verification that peaking factors are within required limits and performing other Actions in CTS 3.10.2 associated with maintaining core peaking factors. However, CTS 3.10.5 does not explicitly require verification that the misaligned rod has not resulted in insufficient SDM resulting from the misaligned rod violating insertion limits.

Under the same conditions (misaligned rod), ITS 3.1.4, Required Actions B.2.1.1 and B.2.1.2, and B.2.3, explicitly require verification and periodic re-verification that the misaligned rod has not resulted in insufficient SDM resulting from the misaligned rod violating insertion limits. Additionally, ITS 3.1.4, Required Action B.2.3, requires re-verification of SDM every 12 hours as long as any rod is not within alignment limits.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while explicitly requiring verification that required SDM is maintained when operating with a misaligned rod. Therefore, this change has no significant adverse impact on safety.

M.3 CTS 3.10.5.1.a and CTS 3.10.5.1.b require that core peaking factors be determined within 2 hours when a rod is not within required alignment limits. If core peaking factors are not determined, then CTS 3.10.5.2 requires a reduction in reactor power (See ITS 3.1.4, DOC M.4).

Under the same conditions (rod not within alignment limits), ITS 3.1.4, Required Action B.2.2, requires a reduction in reactor power within 2 hours regardless of the status or results of the peaking factor verification; but, ITS 3.1.4, Required Actions B.2.4 and B.2.5, allow 72 hours (versus 2 hours) to verify core peaking factors (See ITS 3.1.4, DOC A.7).

This change is needed because the reduction of power ensures that local linear heat rate 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. Additionally, the Completion Time of 72 hours for verification of peaking factors results in the peaking factors being determined at the reduced power level and allows more time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring a reduction in power when operating with a misaligned rod. Therefore, this change has no significant adverse impact on safety.

M.4 CTS 3.10.5.2 requires that the high flux trip setpoint (See ITS 3.1.4, DOC L.1) be reduced to < 85% of its rated value when a rod is not within required alignment limits (See ITS 3.1.4, DOC M.3). This action restricts reactor power to some value less than 85% in order to ensure that local linear heat rate increases due to a misaligned RCCA will not cause the core design criteria to be exceeded.

ITS 3.1.4, Required Action B.2.2, requires a reduction in reactor power (See ITS 3.1.4, DOC L.1) to ≤ 75% of its rated value within 2 hours when a rod is not within required alignment limits.

This change is more restrictive because ITS 3.1.4 always requires the power reduction to 75% RTP for a misaligned rod, even if the misaligned rod results in entry into requirements for exceeding flux tilt limits (CTS 3.10.3 and ITS 3.2.4) which could allow a smaller power reduction.

This change is needed because ITS 3.1.4, Required Actions B.2.4 and B.2.5, defers verification of core peaking factors to 72 hours (versus 2 hours in CTS)

This more restrictive change is acceptable because it does not introduce any operation that is unanalyzed while requiring a more conservative response than is currently required when a rod is not within alignment limits. This change is consistent with the ISTS, and has no adverse impact on safety.

M.5 CTS 3.10.5.3 requires that misaligned control rods must be declared inoperable if not realigned within 8 hours. CTS 3.10.7.2 specifies that not more than one inoperable rod shall be allowed. Together, CTS 3.10.5.3 and CTS 3.10.7.2 prohibit operation with more than one rod not within alignment limits and require initiation of a plant shutdown within 8 hours of the determination that this condition exists. This requires that the plant be in Mode 3 within 14 hours assuming 6 hours is allowed for performing a normal reactor shutdown.

ITS LCO 3.1.4, Condition D and associated Required Action D.2, maintains the prohibition against operation with more than one rod not within alignment limits; however, the Completion Time for placing the reactor in Mode 3 is reduced from 14 hours to 6 hours.

This change is needed because if more than one rod is not within alignment limits, the plant is outside of the accident analysis assumptions. Therefore, prompt initiation of a reactor shutdown is warranted. 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.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring a more conservative response than is currently required when the plant is outside analysis assumptions because of more than one misaligned control rod. Therefore, this change has no significant adverse impact on safety.

M.6 CTS 3.10.8 and CTS Table 4.1-3 require verification every 24 months that rod drop times are within specified limits. ITS SR 3.1.4.3 maintains this requirement except that the SR Frequency is changed to "prior to reactor criticality after each removal of the reactor head." This change may require more or less frequent performance of this SR depending on adherence to the nominal 24 month refueling cycle.

This change is needed and is acceptable because it ties performance of the SR with the activity that is most likely to affect rod motion or rod drop time adversely. Additionally, it is expected that the SR will in almost all cases be performed within the existing required SR Frequency. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.5.2 requires that the high flux trip setpoint be reduced when a rod is not within required alignment limits.

ITS 3.1.4, Required Action B.2.2, requires a reduction in reactor power (but not a reduction in the high flux trip setpoint) when a rod is not within required alignment limits (See ITS 3.1.4, DOCs M.3 and M.4).

This change is acceptable because both the CTS and ITS Actions are intended to reduce power to ensure that local linear heat rate increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. Additionally, ITS 3.1.4, Required Actions B.2.4 and B.2.5, require performance of SR 3.2.1.1 and SR 3.2.2.1. If SR 3.2.1.1 and SR 3.2.2.1 identify that Heat Flux Hot Channel Factor ($F_Q(Z)$) and/or Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) are not within required limits, then Required Actions associated with ITS LCO 3.2.1 and/or ITS LCO 3.2.2 will require appropriate reductions in the high flux trip setpoint to prevent exceeding thermal limits during a transient. Finally, ITS LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR), is applicable and will require appropriate reduction in flux trip setpoints if the misaligned rod results in exceeding QPTR limits. Therefore, this change has no significant adverse impact on safety.

L.2 CTS Table 4.1-3, Item 2, requires verification every 31 days that rods can be moved at least 10 steps in any one direction. ITS SR 3.1.4.2 maintains this requirement; however, the required SR Frequency is extended to 92 days.

This change is needed because experience performing this SR indicates that the identification of a failure is rare. Therefore, performance of this SR consumes considerable resources without a commensurate improvement in plant safety. This change is acceptable because extending the SR Frequency will not significantly increase the probability that the plant will be operated for an extended period of time without identifying a rod that is not capable of being tripped. This is true because operating experience indicates that stuck rods are rare and the occurrence of multiple stuck rod is less probable. Additionally, SR 3.1.4.1 verifies every 12 hours that individual rods are within alignment limits that will quickly identify any rod not moving with its bank. Finally, verification of rod movement for each withdrawn rod will continue to be performed every 92 days. Therefore, considering that the accident analysis assumes the highest worth rod will fail to insert due to a random failure, there is a high degree of assurance that extending this SR Frequency will not result in a failure to meet accident analysis assumptions during a reactor trip. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.10.7.3 specifies that special analyses must be completed within 5 days of the start of operation with a misaligned rod as a condition of continued operation. CTS 3.10.7.3 further specifies that these analyses must address the potential ejected rod worth, non-uniform fuel depletion, associated transient power distribution peaking factors, and the accidents in CTS Table 3.10-1. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level that is consistent with the safety analysis.

Under the same conditions (operation with a misaligned rod for more than 5 days), ITS LCO 3.1.4, Required Action B.2.6, requires a reevaluation

of safety analyses to confirm results remain valid for duration of operation with the misaligned rods. The details regarding required aspects of this analysis found in CTS 3.10.7.3 are relocated to the Bases for ITS LCO 3.1.4, Required Action B.2.6.

This change is acceptable because ITS LCO 3.1.4, Required Action B.2.6, requires a reevaluation of safety analyses to confirm results remain valid for duration of operation with the misaligned rods and the associated Bases defines the required scope of the analysis. Maintaining this information in the Bases is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated to the Technical Specification Bases.

LA.2 CTS 3.10.9 requires that if the rod position deviation monitor is inoperable, then individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power. Additionally, CTS Table 4.1-1, Item 9, requires that the deviation monitor be tested every 31 days.

ITS LCO 3.1.4 (ISTS 3.1.5 modified by TSTF-110 (WOG-49), Rev 1) does not establish any requirements for the rod position deviation monitor; and, details contained in CTS 3.10.9 and CTS Table 4.1-1, Item 9, are relocated to the final Safety Analysis Report (FSAR) and will be implemented by plant procedures. Note that TSTF-110 (WOG-49), Rev 1, a

requirement in NUREG-1431 that was similar to CTS 3.10.9.

This change is acceptable because ITS SR 3.1.4.1 maintains the requirement that rod position be verified every 12 hours regardless of the status of the deviation monitor. The ITS SR 3.1.4.1 Frequency of 12 hours for the verification of rod position recognizes that rod position information is continuously available to the operator in the control room, so that deviations can immediately be detected. Additionally, the requirement for accelerated verification will be maintained in the FSAR and plant procedures.

Changes to the FSAR can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the FSAR and future changes to the FSAR will be controlled in accordance with 10 CFR 50.59.

This change is a less restrictive administrative change with no impact on safety because ITS 3.1.4 maintains the requirements to verify rod position every 12 hours. Therefore, a requirement to test the deviation monitor and accelerate monitoring when the monitor is not functional can be maintained in the FSAR with no significant adverse impact on safety.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.4:
"Rod Group Alignment Limits"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.4 - Rod Group Alignment Limits

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   CTS 3.10.5.2 requires that the high flux trip setpoint be reduced when a rod is not within required alignment limits. ITS 3.1.4, Required Action B.2.2, requires a reduction in reactor power (but not a reduction in the high flux trip setpoint) when a rod is not within required alignment limits.

   This change does not involve a significant increase in the probability of an accident previously evaluated because the status of the high flux trip setpoint has no effect of the initiators of any analyzed event. This change does not involve a significant increase in the consequences of an accident previously evaluated because both the CTS and ITS Actions are intended to reduce power to ensure that local linear heat rate increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. Additionally, ITS 3.1.4, Required Actions B.2.4 and B.2.5, require performance of SR 3.2.1.1 and SR 3.2.2.1. If SR 3.2.1.1 and SR 3.2.2.1 identify that Heat Flux Hot Channel Factor ($F_Q(Z)$) and/or Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) are not within required limits, then Required Actions associated with ITS LCO 3.2.1 and/or ITS LCO 3.2.2 will require appropriate reductions in the high flux trip setpoint to prevent exceeding thermal limits during a transient. Finally, ITS LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR), is applicable and will require appropriate reduction in flux trip setpoints if the misaligned rod results in exceeding QPTR limits.

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

The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC), or involve changes in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because both the CTS and ITS Actions are intended to reduce power to ensure that local linear heat rate increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. Additionally, ITS 3.1.4, Required Actions B.2.4 and B.2.5, require performance of SR 3.2.1.1 and SR 3.2.2.1. If SR 3.2.1.1 and SR 3.2.2.1 identify that Heat Flux Hot Channel Factor ($F_Q(Z)$) and/or Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) are not within required limits, then Required Actions associated with ITS LCO 3.2.1 and/or ITS LCO 3.2.2 will require appropriate reductions in the high flux trip setpoint to prevent exceeding thermal limits during a transient. Finally, ITS LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR), is applicable and will require appropriate reduction in flux trip setpoints if the misaligned rod results in exceeding QPTR limits.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

CTS Table 4.1-3, Item 2, requires verification every 31 days that rods

can be moved at least 10 steps in either direction. ITS SR 3.1.4.2 maintains this requirement; however, the required SR Frequency is extended to 92 days.

This change is needed because experience performing this SR indicates that the identification of a failure is rare. Therefore, performance of this SR consumes considerable resources without a commensurate improvement in plant safety.

This change does not involve a significant increase in the probability of an accident previously evaluated because the SR Frequency for rod movement verification has no effect of the initiators of any analyzed event. This change does not involve a significant increase in the probability of an accident previously evaluated because extending the SR Frequency will not significantly increase the probability that the plant will be operated for an extended period of time without identifying a rod that is not capable of being tripped. This is true because operating experience indicates that stuck rods are rare and the occurrence of multiple stuck rod is less probable. Additionally, SR 3.1.4.1 verifies every 12 hours that individual rods are within alignment limits that will quickly identify any rod not moving with its bank. Finally, verification of rod movement for each withdrawn rod will continue to be performed every 92 days. Therefore, considering that the accident analysis assumes the highest worth rod will fail to insert due to a random failure, there is a high degree of assurance that extending this SR Frequency will not result in a failure to meet accident analysis assumptions during a reactor trip.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC), or involve changes in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of

safety because extending the SR Frequency will not significantly increase the probability that the plant will be operated for an extended period of time without identifying a rod that is not capable of being tripped. This is true because operating experience indicates that stuck rods are rare and the occurrence of multiple stuck rod is less probable. Additionally, SR 3.1.4.1 verifies every 12 hours that individual rods are within alignment limits that will quickly identify any rod not moving with its bank. Finally, verification of rod movement for each withdrawn rod will continue to be performed every 92 days. Therefore, considering that the accident analysis assumes the highest worth rod will fail to insert due to a random failure, there is a high degree of assurance that extending this SR Frequency will not result in a failure to meet accident analysis assumptions during a reactor trip.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.4:
"Rod Group Alignment Limits"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.4

This ITS Specification is based on NUREG-1431 Specification No. **3.1.5**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.1 R1	**009 R1**	RELOCATE VALUE FOR SHUTDOWN MARGIN TO COLR	Approved by NRC	Incorporated	**T.1**
WOG-004.2 R1	**010 R1**	REVISE THE CONTROL ROD LCOS APPLICABILITY FROM MODE 2 TO MODE 2 WITH KEFF >= 1.0	Rejected by NRC	Not Incorporated	**N/A**
WOG-004.3 R1	**011 R1**	DELETE "ALL" FROM LCO 3.1.5, "ROD GROUP ALIGNMENT LIMITS"	Rejected by NRC	Not Incorporated	**N/A**
WOG-004.7	**015 R0**	CORRECT ERROR IN BASES FOR LCO 3.1.5	See Next Rev.	See Rev 1	**N/A**
WOG-004.7 R1	**015 R1**	CORRECT ERROR IN BASES FOR LCO 3.1.5	Approved by NRC	Incorporated	**T.3**
WOG-043	**107 R0**	SEPARATE CONTROL RODS THAT ARE UNTRIPPABLE VERSUS INOPERABLE	TSTF to Rewrite	Not Incorporated	**N/A**

Technical Specification 3.1.4: "Rod Group Alignment Limits"

WOG-049 R2	**110 R2**	DELETE SR FREQUENCIES BASED ON INOPERABLE ALARMS	Approved by NRC	Incorporated	**T.4**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**
WOG-094		ELIMINATE UNNECESSARY ACTIONS TO RESTORE COMPLIANCE WITH THE LCO	TSTF Review	Not Incorporated	**N/A**
WOG-105		REQUIRE STATIC AND TRANSIENT FQ MEASUREMENT	TSTF Review	Not Incorporated	**N/A**

T.1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~5~~ 4 Rod Group Alignment Limits

LCO 3.1.~~5~~ 4 All shutdown and control rods shall be OPERABLE, ~~with all individual indicated rod positions within 12 steps of their group step counter demand position.~~

CLB.1

Insert: 3.1-8-01

<3.10.7.2>
<DOC A.3>
<DOC M.1>

APPLICABILITY: MODES 1 and 2.

<3.10.7.2>
<DOC A.4>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.10.7.2> <3.10.4.3> A. One or more rod(s) untrippable. <DOC A.6>	A.1.1 Verify SDM is ≥ ~~[1.6]%~~ Δk/k.	1 hour
	OR	
<DOC A.6>	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
<DOC M.1>	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits. <3.10.5.1.a> <3.10.5.1.b>	B.1 Restore rod to within alignment limits.	1 hour
	OR	
<DOC M.2>	B.2.1.1 Verify SDM is ≥ ~~[1.6]%~~ Δk/k.	1 hour
	OR	
		(continued)

T.1

Insert: 3.1-8-02

Insert: 3.1-8-03

T.1

Page Break

INSERT: 3.1-8-01

CLB.1

with rod group alignment limits as follows:

<3.10.5.1.b> a. When THERMAL POWER is > 85% RTP,

1. Groups with step counter demand position ≤ 212 steps shall have all individual indicated rod positions ≤ ± 12 steps of their group step counter demand position; and

<3.10.5.1.a> 2. Groups with step counter demand position > 212 steps shall have all individual indicated rod positions ≤ +17 steps and ≥ -12 steps of their group step counter demand position

b. When THERMAL POWER is ≤ 85% RTP, all individual indicated rod positions shall be ≤ ± 18 steps of their group step counter demand position.

INSERT: 3.1-8-02

within the limits specified in the COLR.

T.1

INSERT: 3.1-8-03

within the limits specified in the COLR.

T.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.10.5.1.a, b> B. (continued) <3.10.7.2> <3.10.4.3> <DOC M.2>	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
<3.10.5.2> <DOC L.1> <DOC M.4> <DOC M.3>	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND	
<DOC M.2> <DOC A.6> Insert: 3.1-9-01	B.2.3 Verify SDM is ~~≥ 1.6% Δk/k.~~	Once per 12 hours (T.1)
	AND	
<3.10.5.1.a> <3.10.5.1.b> <DOC A.7> <DOC M.3>	B.2.4 Perform SR 3.2.1.1.	72 hours
	AND	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	AND	
<3.10.7.3>	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
<3.10.5.3> <3.10.7.3> <DOC A.5> C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)

INSERT: 3.1-9-01

within the limits specified in the COLR.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is ≥ [1.6]% Δk/k	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

T.1

⟨3.10.7.2⟩
⟨3.10.4.3⟩
⟨DOC A.6⟩
⟨DOC A.6⟩

Insert: 3.1-10-01

⟨3.10.7.2⟩
⟨DOC A.5⟩
⟨3.10.5.3⟩
⟨DOC M.5⟩

CLB.2

Insert: 3.1-10-02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.~~5~~.1 4	Verify individual rod positions within alignment limit.	12 hours AND ~~Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable~~

T.4

⟨DOC A.14⟩
⟨3.10.5.D⟩
⟨3.10.3.3⟩
⟨DOC A.9⟩
⟨DOC A.10⟩
⟨T 4.1-1, #9⟩
⟨DOC M.6⟩

(continued)

INSERT: 3.1-10-01

within the limits specified in the COLR.

INSERT: 3.1-10-02

CLB.2

<DOC A.14>

----------NOTE----------
Not required to be performed for individual control rods until 1 hour after completion of control rod movement.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.~~5~~4.2 <T4.1-3, #2> <DOC A.4> <DOC L.2> <DOC A.11>	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in ~~either~~ one direction.	92 days
SR 3.1.~~5~~4.3 <3.10.8> <T.4.1-3, #1> <DOC M.6> <DOC A.13>	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.2 (1.8) seconds from the ~~beginning of decay~~ loss of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

PA.1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~5~~ 4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., 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 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).

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 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 5/8 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 may consist~~s~~ of two groups DB.1

(continued)

BASES

BACKGROUND (continued)

that are moved in a staggered fashion, but always within one step of each other. ~~All units have~~ four control banks and ~~at least two~~ shutdown banks. DB.1

IP3 has

four

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. DB.1

at the desired position

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the ~~Digital~~ Rod Position Indication (~~D~~RPI) System.

Individual

I

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is ~~considered~~ highly precise (± 1 step or ± 5/8 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.

I

an

The ~~D~~RPI System provides ~~a highly accurate~~ indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from ~~a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six 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~~ DB.1

Insert: B3.1-25-01

(continued)

INSERT: 3.1-25-01

DB.1

coil stack located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained.

BASES

BACKGROUND (continued)

The rod position

Insert: B 3.1-26-01

~~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 indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.~~

DB.1

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 are that:

a. There 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.

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

(continued)

INSERT: B 3.1-26-01

misalignment limit of 12 steps precludes a rod misalignment of > 15 inches when instrument error is considered. An indicated misalignment limit of 18 steps precludes a rod misalignment of > 18.75 inches when instrument error is considered.

BASES

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. 5).

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

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

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 ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

Insert: B3.1-27-01

~~The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.~~

CLB.1

(continued)

INSERT: B 3.1-27-01

To ensure that individual rods are properly aligned with its associated group step counter demand position, the following limits are placed on individual rod positions:

When THERMAL POWER is > 85% RTP,

1. Groups with step counter demand position ≤ 212 steps shall have all individual indicated rod positions within 12 steps of their group step counter demand position; and

2. Groups with step counter demand position > 212 steps shall have all individual indicated rod positions ≤ +17 steps and ≥ -12 steps of their group step counter demand position

When THERMAL POWER is ≤ 85% RTP, all individual indicated rod positions shall be ≤ ± 18 steps of their group step counter demand position.

These limits ensure analysis assumptions for SDM and peaking factors ae met because an indicated misalignment of 12 steps precludes a rod misalignment of > 15 inches when instrument error is considered. An indicated misalignment limit of 18 steps precludes a rod misalignment of > 18.75 inches when instrument error is considered.

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. (Ref. 4)

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 typically 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. DB.1

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM. PA.1

Insert: B 3.1-28-01

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

A.2

If the 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

(continued)

INSERT: B 3.1-28-01

Required Actions A.1.1 and A.1.2 apply if either SR 3.1.4.2 or SR 3.1.4.3 are not met.

BASES

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.

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.

CLB.1

Insert: B3.1-29-01

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.~~6~~ 5, "Shutdown Bank Insertion Limits," and LCO 3.1.~~7~~ 6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.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.

(continued)

INSERT: B 3.1-29-01

If all individual indicated rod positions are ≤ ± 18 steps of their group step counter demand position, the LCO may be met by reducing reactor power to less than 85% RTP.

BASES

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.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

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.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

Insert: B 3.1-30-01

(continued)

INSERT: B 3.1-30-01

The analysis specified by Required Action B.2.6 must address the potential ejected rod worth, non-uniform fuel depletion, associated transient power distribution peaking factors and accidents. The following issues must also be addressed:

Rod cluster control assembly insertion characteristics;

Rod Cluster Control Assembly Misalignment;

Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates the emergency core cooling system;

Single rod cluster control assembly withdrawal at full power;

Major reactor coolant system pipe ruptures (loss of coolant accident);

Major Secondary system pipe rupture; and

Rupture of a control rod drive mechanism housing.

BASES

ACTIONS (continued)

C.1.1 and C.1.2 D

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 or LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

C.2 D

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 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 C

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

T.3

T.3

(continued)

BASES

ACTIONS

~~B.1 (continued)~~

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

SURVEILLANCE REQUIREMENTS

SR 3.1.~~5~~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. ~~If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal.~~ 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.

T.4

Insert: B3.1-32-01

Insert: B3.1-32-02

CLB.2

SR 3.1.~~5~~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 in a single direction will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.~~5~~4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.~~5~~4.2 (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) of the control rod(s) must be made, and appropriate action taken.

PA.1

Insert: B3.1-32-03

4

4

(continued)

INSERT: B 3.1-32-01

Rod position may be verified using normal indication, direct readings using a digital volt meter, or the plant computer.

INSERT: B 3.1-32-02

CLB.2

This SR is not required to be performed for an individual control rod until 1 hour after completion of movement of that rod. This allowance is needed because it provides time for thermal stabilization of rod position instrumentation. This allowance is acceptable because individual rod position indicators may not accurately reflect control rod position prior to thermal stabilization and there is a presumption that individual control rods will move with their group.

INSERT: B 3.1-32-03

PA.1

This SR requires that control rods be inserted or withdrawn by at least 10 steps which is sufficient to ensure that rod movement can be confirmed by individual rod position indicators. Administrative controls and Technical Specification limits ensure that control rod insertion limits are met.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.~~5~~ 4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

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~~ was performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC ~~10 and GDC 26~~.
2. 10 CFR 50.46.
3. FSAR, Chapter ~~[15]~~ 14.
4. ~~FSAR, Chapter [15].~~
5. ~~FSAR, Chapter [15].~~
6. ~~FSAR, Chapter [15].~~
7. ~~FSAR, Chapter [15].~~

Insert: B 3.1-33-01

INSERT: B 3.1-33-01

4. WCAP-14668, Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3, October 1996 (Proprietary).

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.4:
"Rod Group Alignment Limits"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.4 - Rod Group Alignment Limits

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG-1431, Rev1, LCO 3.1.4 requires that all shutdown and control rods have indicated rod positions within 12 steps of their group step counter demand position. IP3 ITS differs from NUREG-1431, Rev1, in that the limits are relaxed when Thermal Power is > 85% RTP. This deviation from NUREG-1431, Rev.1, maintains the current licensing basis as approved in Technical Specifications through Amendment 181.

CLB.2 CTS 3.10.5.1 specifies that the required verification individual rod positions should allow "one hour for thermal soak after rod motion." IP3 ITS differs from NUREG-1431 in that a Note to ITS SR 3.1.4.1 specifies that this SR is not required to be performed for individual control rods until 1 hour after completion of control rod movement. This allowance is needed because it provides time for thermal stabilization of rod position instrumentation. This allowance is acceptable because individual rod position indicators may not accurately reflect control rod position prior to thermal stabilization and there is a presumption that individual control rods will move with their group. This deviation from NUREG-1431, Rev.1, maintains the current licensing basis as approved in Technical Specifications through Amendment 181.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-

explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-09 (WOG-04.1), Rev.1, which relocated values for shutdown margin (SDM) to the COLR. SDM is a cycle-specific variable similar to moderator temperature coefficient, the rod insertion limits, axial flux difference, heat flux hot channel factor, and nuclear rise hot channel factor, which are currently contained in the COLR.

T.2 This change incorporates Generic Change TSTF-136 (WOG-59), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM)-T_{avg} > 200°F, and ISTS 3.1.2, SHUTDOWN MARGIN (SDM)-T_{avg} ≤ 200°F, into ISTS 3.1.1, SHUTDOWN MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR.

T.3 This change incorporates Generic Change TSTF-015 (WOG-04.7), which corrected an error in the Bases for ISTS LCO 3.1.5 caused by incomplete incorporation of WOG-17, C.1, Revision 0 change.

T.4 This change incorporates Generic Change TSTF-110, Rev. 2 (WOG-49), which relocated actions (in the form of an increased surveillance frequency) related to several surveillances (rod position deviation monitor, rod insertion limit monitor, AFD monitor, and QPTR alarm) from the Technical Specifications to other licensee controlled documents. The monitors or alarms themselves do not directly relate to the LCO requirements.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.5:
"Shutdown Bank Insertion Limits"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

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.

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

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 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.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs may consist of two groups that are moved in a staggered fashion, but always within one step of each other. IP3 has four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

BASES

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, power, and fuel depletion. 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 power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank

BASES

APPLICABLE SAFETY ANALYSES (Continued)

insertion limit also limits the reactivity worth of an ejected shutdown rod when at power.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

1. Specified acceptable fuel design limits, or

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 shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36.

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

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

BASES

APPLICABILITY (continued)

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 4, 5, or 6, the shutdown banks are normally fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

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 the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a

BASES

ACTIONS

B.1 (continued)

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

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A.

2. 10 CFR 50.46.

3. FSAR, Chapter 14.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.5:
"Shutdown Bank Insertion Limits"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-5	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-15	112	112	No TSCRs	No TSCRs for this Page	N/A

SEE ITS 3.1.4	3.10.3.3	The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.
SEE ITS 3.2.4	3.10.3.4	The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.
LCO 3.1.5	~~3.10.4~~	Rod Insertion Limits (Shutdown Bank)
	~~3.10.4.1~~	The shutdown rods shall be ~~fully withdrawn~~ (A.3) as specified in the COLR ~~when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of Specification 3.10.1)~~ (A.4) (Mode 1, Mode 2 with any control Bank not inserted)
SEE ITS 3.1.6	3.10.4.2	When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.
	3.10.4.3	Control bank insertion shall be further restricted if: a) The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown. b) A rod is inoperable (Specification 3.10.7).
Note to LCO 3.1.5, Applicability	~~3.10.4.4~~	Control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. ~~However, the shutdown margin required by Specification 3.10.1 must be maintained~~ (L.1) except for the low power physics test to measure control rod worth and shutdown margin. For this ~~test~~, the reactor may be critical with all but one control rod inserted.
SEE ITS 3.1.8		

(Add Condition A and B and associated Req. Actions) — (M.1)

(Add SR 3.1.5.1) — (M.2)

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3 % $\Delta k/k$ is required to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

The action to be taken when shutdown margin in out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

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

A.1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.5:
"Shutdown Bank Insertion Limits"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.5 - Shutdown Bank Insertion Limits

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.4.1 requires that the shutdown rods be fully withdrawn as specified in the Core Operating Limits Report (COLR). ITS LCO 3.1.5 requires that each shutdown bank be within the insertion limits specified in the COLR. This is an administrative change with no significant adverse impact on safety because the COLR specifies that the shutdown bank position for criticality is fully withdrawn.

A.4 CTS 3.10.4.1, Shutdown Bank Insertion Limits, is Applicable when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of CTS 3.10.1). ITS LCO 3.1.5, Shutdown Bank Insertion Limits, is Applicable in Mode 1 and in Mode 2 with any control bank not fully inserted.

The difference between the CTS Applicability and the ITS Applicability is the core reactivity level that defines the transition between Mode 2 (CTS 1.2.3 definition of Reactor Critical) and Mode 3 (CTS 1.2.2 definition of Hot Shutdown Condition). CTS define the transition as the point when the reactor is subcritical by less than the required SDM of 1.3% $\Delta k/k$. ITS defines the transition as the point when the reactor is subcritical by less than 1.0% $\Delta k/k$. However, LCO 3.1.1, Shutdown Margin, maintains the requirement to have SDM $\geq$ 1.3% $\Delta k/k$ when in Modes 3 and 4 and in Mode 2 when Keff is < 1.0. (See ITS Section 1.0, DOC A.4)

Under CTS the reactor is considered approaching criticality with initiation of withdrawal of the first control bank. Under ITS, the reactor is in Mode 2 prior to the initiation of withdrawal of the first control bank. Therefore, this is an administrative change with no significant adverse impact on safety because the CTS 3.10.4.1 Applicability and the ITS LCO 3.1.5 Applicability are functionally equivalent.

MORE RESTRICTIVE

M.1 CTS 3.10.4.1 requires that the shutdown rods be fully withdrawn as specified in the COLR; however, no Required Actions or Completion Times are specified if this requirement is not met. ITS LCO 3.1.5, Required Actions A.1.1, A.1.2, A.2, and B.1, are added to address the condition of one or more shutdown banks not within the insertion limits, or the Required Actions and Completion Times associated with restoring the shutdown banks to within limits are not met. If one or more shutdown banks are not within the insertion limits, these Required Actions specify that within one hour SDM must be verified within limits in the COLR or boration is initiated to restore the SDM within limits.

Additionally, the shutdown banks must be restored to within the insertion limits within 2 hours. Otherwise, the reactor must be in Mode 3 within the following 6 hours.

This change is needed because the available SDM may be significantly reduced if any shutdown bank is not within the insertion limit and sufficient SDM is not otherwise met. The allowed Completion Times provide an acceptable time for evaluating and correcting minor problems without allowing the plant to remain in an unacceptable condition for an extended time. This more restrictive change is acceptable because it does not introduce any operation un-analyzed while requiring appropriate actions be completed within a reasonable time when requirements are not met. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.10.4.1 requires that the shutdown rods be fully withdrawn as specified in the COLR; however, CTS does not establish any requirement for periodic verification that this requirement is met. ITS SR 3.1.5.1 is added to verify each shutdown bank is within the limits specified in the COLR before the control banks are withdrawn during a unit startup and every 12 hours thereafter.

The Frequency of 12 hours is adequate to ensure requirements are being met because the shutdown banks are positioned manually and control room operators will be aware of any changes. Also, the 12 hour Frequency takes into account other information available in the control room for monitoring the status of shutdown rods. This more restrictive change is acceptable because it does not introduce any operation un-analyzed while requiring verification that requirements are met. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.4.4 specifies that control rod insertion limits do not apply during periodic exercise of individual rods; however, the shutdown margin required by CTS 3.10.1 must be maintained. ITS LCO 3.1.5 maintains this allowance with a Note that specifies that the LCO is not

applicable while performing SR 3.1.4.2. SR 3.1.4.2 moves each control rod by 10 steps to verify freedom of movement. However, ITS does not explicitly state that SDM requirements must be met when using this allowance.

This change is acceptable because the impact on SDM of moving individual or small groups of control rods by 10 steps is small, the SR is performed infrequently (92 days), and the duration of the test is short. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.5:
"Shutdown Bank Insertion Limits"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.5 - Shutdown Bank Insertion Limits

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   CTS 3.10.4.4 specifies that control rod insertion limits do not apply during periodic exercise of individual rods; however, the shutdown margin required by CTS 3.10.1 must be maintained. ITS LCO 3.1.5 maintains this allowance; however, the ITS does not explicitly state that SDM requirements must be met when using this allowance.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the impact on SDM of moving individual or small groups of control rods by 10 steps is small, the SR is performed infrequently (92 days), and the duration of the test is short.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the impact on SDM of moving individual or small groups of control rods by 10 steps is small, the SR is performed infrequently (92 days), and the duration of the test is short.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.5:
"Shutdown Bank Insertion Limits"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.5

This ITS Specification is based on NUREG-1431 Specification No. **3.1.6**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.1 R1	**009 R1**	RELOCATE VALUE FOR SHUTDOWN MARGIN TO COLR	Approved by NRC	Incorporated	**T.1**
WOG-004.2 R1	**010 R1**	REVISE THE CONTROL ROD LCOS APPLICABILITY FROM MODE 2 TO MODE 2 WITH KEFF >= 1.0	Rejected by NRC	Not Incorporated	**N/A**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**
WOG-093		CORRECT SHUTDOWN BANK INSERTION LIMITS APPLICABILITY	TSTF Review	Not Incorporated	**N/A**

T.2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~6~~ 5 Shutdown Bank Insertion Limits

LCO 3.1.~~6~~ 5 Each shutdown bank shall be within insertion limits specified in the COLR.

<3.10.4.1>
<DOC A.3>

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

<3.10.4.1>
<DOC A.4>

----------NOTE----------
This LCO is not applicable while performing SR 3.1.~~5~~ 4.2.

<3.10.4.4>
<DOC L.1>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is ≥ ~~[1.6]% Δk/k~~	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

<DOC M.1>

Insert: 3.1-12-01

T.1

<DOC M.1>

INSERT: 3.1-12-01

within the limits specified in the COLR.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.~~6~~ 5.1	Verify each shutdown bank is within the limits specified in the COLR.	12 hours

<Doc M.2>

T.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~6~~ 5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), 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.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~plants have~~ four control banks and ~~at least two~~ shutdown banks. See LCO 3.1.~~3~~ 4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.~~3~~ 7, "Rod Position Indication," for position indication requirements.

may

IP3 has

AY1

four

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.

(continued)

BASES

power, and fuel depletion

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.

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.~~7~~ 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) ~~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, 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.

T.2

when at power

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. There 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 shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

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

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.~~5~~4.2. This SR verifies the freedom of the rods to

(continued)

BASES

APPLICABILITY (continued)

move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

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 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.~~6~~ 5.1 (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.

REFERENCES

1. 10 CFR 50, Appendix A, ~~GDC 10, GDC 26, and GDC 28~~.
2. 10 CFR 50.46.
3. FSAR, Chapter ~~15~~ 14.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.5:
"Shutdown Bank Insertion Limits"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.5 - Shutdown Bank Insertion Limits

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-09, Rev.0 (WOG-04.1), which relocated values for shutdown margin (SDM) to the COLR. SDM is a cycle-specific variable similar to moderator temperature coefficient, the rod insertion limits, axial flux difference, heat flux hot channel factor, and nuclear rise hot channel factor, which are currently contained in the COLR. In addition, there is an NRC approved methodology for determining SDM.

T.2 This change incorporates Generic Change TSTF-136, Rev.0 (WOG-59), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, and ISTS 3.1.2,

SHUTDOWN MARGIN (SDM) – $T_{avg} \leq 200°F$, into ISTS 3.1.1, SHUTDOWN MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.6:
"Control Bank Insertion Limits"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

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.

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

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

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
	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

(continued)

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 specified 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 3.	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.	Within 4 hours prior to achieving criticality

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 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.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs may consist of two groups that are moved in a staggered fashion, but always within one step of each other. IP3 has four control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit. The COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The fully withdrawn position is defined in the COLR.

BASES

BACKGROUND (continued)

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 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 will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Protection System (RPS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

a. There be no violations of:

1. specified acceptable fuel design limits, or

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).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36 because they are initial conditions assumed in the safety analysis.

BASES

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the 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 violate the LCO.

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

BASES

ACTIONS

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

normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

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

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps 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 must be brought to MODE 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated for a time different from when criticality occurs, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Verifying the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

BASES

REFERENCES

1. 10 CFR 50, Appendix A.

2. 10 CFR 50.46.

3. FSAR, Chapter 14.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.6:
"Control Bank Insertion Limits"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-5	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-15	112	112	No TSCRs	No TSCRs for this Page	N/A
F 3.10-4	143	143	No TSCRs	No TSCRs for this Page	N/A
F 3.10-5	48	48	No TSCRs	No TSCRs for this Page	N/A

SEE ITS 3.1.4 3.10.3.3	The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.
SEE ITS 3.2.4 3.10.3.4	The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.
~~3.10.4~~	~~Rod~~ Insertion Limits (Control Bank)
SEE ITS 3.1.5 3.10.4.1	The shutdown rods shall be fully withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of Specification 3.10.1).
~~3.10.4.2~~	~~When the reactor is critical~~ (Mode 1, Mode 2 with keff ≥ 1.0), the control banks shall be limited ~~in physical insertion~~ to the insertion (sequence and overlap) limits specified in the COLR. (A.1) (A.3)
~~3.10.4.3~~	~~Control bank insertion shall be further restricted if:~~ ~~a) The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown.~~ ~~b) A rod is inoperable (Specification 3.10.1).~~ (A.4)
Note to ~~3.10.4.4~~ LCO 3.1.6, Applicability	Control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. ~~However, the shutdown margin required by Specification 3.10.1 must be maintained~~ (L.1)
SEE ITS 3.1.8	except for the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one control rod inserted.

Add Conditions A, B and C and associated Req. Actions (M.1)

Add SR 3.1.6.1
SR 3.1.6.2
SR 3.1.6.3 (M.2)

Amendment No. ~~14~~, ~~88~~, ~~103~~, 112

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3 % $\Delta k/k$ is required to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

The action to be taken when shutdown margin in out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

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

A.1

DELETED

A.1

Figure 3.10-4 Insertion Limits 100 Step Overlap - Four Loop Operation

A.1

D E L E T E D

Figure 3.10-5 Insertion Limits 100 Step Overlap 3 Loop Operation

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.6:
"Control Bank Insertion Limits"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.6 - Control Bank Insertion Limits

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.4.2 requires that the control banks be limited in physical insertion to the insertion limits in the COLR. ITS LCO 3.1.6 requires that control banks be within the insertion, sequence, and overlap limits specified in the COLR. The insertion limits specified in the COLR include insertion, sequence, and overlap limits. Therefore, this is an administrative change with no significant adverse impact on safety because there is no change to the existing requirements.

A.4 CTS 3.10.4.3 specifies that control bank insertion must be restricted because of the following: a) the measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown margin; or, b) A rod is inoperable (i.e., untrippable, slow or misaligned per CTS 3.10.7.1). CTS 3.10.4.3 is guidance regarding conditions that may invalidate or modify the normal control bank insertion limits specified in the COLR.

The guidance provided in CTS 3.10.4.3 is not explicitly included in the ITS. This change is acceptable because ITS LCO 3.1.4, Rod Group Alignment Limits, and ITS LCO 3.1.5, Shutdown Bank Insertion Limits, establish Limiting Conditions for Operation that ensure appropriate Required Actions are initiated if conditions exist that could invalidate the normal control bank insertion limits specified in the COLR. Additionally, ITS LCO 3.1.4, Required Actions A.1.1, A.1.2, B.2.1.1, B.2.1.2, D.1.1 and D.1.2, require verification within one hour that SDM requirements are met if an inoperable (slow or untrippable rod) or a misaligned rod invalidate assumptions regarding insertion limits (See ITS 3.1.4). Therefore, this is an administrative change with no significant adverse impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.10.4.2 requires that control banks must be limited in physical insertion to the insertion limits specified in the COLR; however, no Required Actions or Completion Times are specified if this requirement is not met.

ITS LCO 3.1.6, Required Actions A.1.1, A.1.2, A.2, and B.1.1, B.1.2, B.2, and C.1, are added to address the condition of one or more control banks not within the insertion, sequence or overlap limits, or the Required Actions and Completion Times associated with restoring the control banks to within limits are not met. If one or more control banks are not within the insertion, sequence or overlap limits, these Required Actions specify that within one hour SDM must be verified within limits in the COLR or boration is initiated to restore the SDM within limits. Additionally, the control banks must be restored to within the insertion limits within 2 hours. Otherwise, the reactor must

be in Mode 3 within the following 6 hours.

This change is needed because the available SDM may be significantly reduced or peaking factors violated if any control bank is not within limits specified in the COLR. The allowed Completion Times provide an acceptable time for evaluating and correcting minor problems without allowing the plant to remain in an unacceptable condition for an extended time. This more restrictive change is acceptable because it does not introduce any operation un-analyzed while requiring appropriate actions be completed within a reasonable time when requirements are not met. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.10.4.2 requires that control banks must be limited in physical insertion to the insertion limits specified in the COLR; however, CTS does not establish any requirement for periodic verification that this requirement is met.

ITS SR 3.1.6.1 is added to require verification within 4 hours prior to achieving criticality that the estimated critical control bank position is within limits specified in the COLR. SR 3.1.6.1 ensures that the reactor does not achieve criticality with the control banks below their insertion limits. The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated for a time different than when criticality occurs, 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 operating attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes to xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

ITS SR 3.1.6.2 is added to require verification every 12 hours that each control bank insertion is within the limits specified in the COLR. SR 3.1.6.2 is sufficient to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. When control banks are maintained within their insertion limits, it is unlikely that their sequence and overlap will not be in

accordance with the requirements specified in the COLR.

ITS SR 3.1.6.3 is added to require verification every 12 hours that sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn. Performance of SR 3.1.6.3 every 12 hours is consistent with the insertion limit check of SR 3.1.6.2. This more restrictive change is acceptable because it does not introduce any operation un-analyzed while requiring verification that requirements are met. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.4.4 specifies that control rod insertion limits do not apply during periodic exercise of individual rods; however, the shutdown margin required by CTS 3.10.1 must be maintained. ITS LCO 3.1.6 maintains this allowance with a Note that specifies that the LCO is not applicable while performing SR 3.1.4.2. SR 3.1.4.2 moves each control rod by 10 steps to verify freedom of movement. However, ITS does not explicitly state that SDM requirements must be met when using this allowance.

This change is acceptable because the impact on SDM of moving individual or small groups of control rods by 10 steps is small, the SR is performed infrequently (92 days), and the duration of the test is short. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.6:
"Control Bank Insertion Limits"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.6 - Control Bank Insertion Limits

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   CTS 3.10.4.4 specifies that control rod insertion limits do not apply during periodic exercise of individual rods; however, the shutdown margin required by CTS 3.10.1 must be maintained. ITS LCO 3.1.6 maintains this allowance; however, the ITS does not explicitly state that SDM requirements must be met when using this allowance.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the impact on SDM of moving individual or small groups of control rods by 10 steps is small, the SR is performed infrequently (92 days), and the duration of the test is short.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the impact on SDM of moving individual or small groups of control rods by 10 steps is small, the SR is performed infrequently (92 days), and the duration of the test is short.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.6:
"Control Bank Insertion Limits"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.6

This ITS Specification is based on NUREG-1431 Specification No. **3.1.7**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.1 R1	**009 R1**	RELOCATE VALUE FOR SHUTDOWN MARGIN TO COLR	Approved by NRC	Incorporated	**T.1**
WOG-004.2 R1	**010 R1**	REVISE THE CONTROL ROD LCOS APPLICABILITY FROM MODE 2 TO MODE 2 WITH KEFF >= 1.0	Rejected by NRC	Not Incorporated	**N/A**
WOG-049 R2	**110 R2**	DELETE SR FREQUENCIES BASED ON INOPERABLE ALARMS	Approved by NRC	Incorporated	**T.3**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**
WOG-092		CORRECT CONTROL BANK INSERTION LIMITS ACTION FOR APPLICABLE MODE	TSTF Review	Not Incorporated	**N/A**

T.2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~7~~ 6 Control Bank Insertion Limits

LCO 3.1.~~7~~ 6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

<3.10.4.2>
<DOC A.3>
<DOC A.4>

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

<3.10.4.2>

----------NOTE----------
This LCO is not applicable while performing SR 3.1.~~5~~ 4.2.

<3.10.4.4>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1 Verify SDM is ≥ ~~1.6% Δk/k~~	1 hour
	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

T.1

<DOC M.1>

Insert: 3.1-14-01

(continued)

INSERT: 3.1-14-01

within the limits specified in the COLR.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met. <DOC M.1>	B.1.1 Verify SDM is ≥ ~~[1.6]% Δk/k~~ Insert 3.1-15-01	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. <DOC M.1>	C.1 Be in MODE 3.	6 hours

T.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.~~7~~ 6.1 <DOC M.2>	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality

(continued)

INSERT: 3.1-15-01

within the limits specified in the COLR.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.~~7~~ 6.2 <DOC M.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~~ T.3
SR 3.1.~~7~~ 6.3 <DOC M.2>	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours

T.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~7~~ 6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consist~~s~~ of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All~~ plants have four control banks and ~~at least two~~ shutdown banks. See LCO 3.1.~~5~~ 4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.~~8~~ 7, "Rod Position Indication," for position indication requirements.

may

IP3 has

four

The control bank insertion limits are specified 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~~.

no H

The COLR

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~~

PA.2

(continued)

BASES

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.

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 power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.5, LCO 3.1.6, "Shutdown Bank Insertion Limits," LCO 3.1.7, 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.

"Rod Group Alignment Limits" 4

6 "Control Bank Insertion Limits"

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

a. There be no violations of:

Space

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. 4). 3

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5). 3

10 CFR 50.36 because

The insertion limits satisfy Criterion 2 of the NRC Policy Statement, 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 maintained, and ensuring adequate

(continued)

BASES

LCO (continued)

negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.9.2. 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.

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 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) — $T_{avg} \geq 200°F$") 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.

(continued)

BASES

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 a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps 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 must be brought to MODE 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.

SURVEILLANCE REQUIREMENTS

SR 3.1.~~7~~ 6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

for a time different from when

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

occurs

PA.2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.~~7~~ 6.1 (continued)

that point. ~~Performing~~ Verifying 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.

PA.1

SR 3.1.~~7~~ 6.2

~~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~~ 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.~~

T.3

SR 3.1.~~7~~ 6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.~~0~~ 6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.~~7~~ 6.2.

REFERENCES

1. 10 CFR 50, Appendix A, ~~GDC 10, GDC 26, GDC 28~~.
2. 10 CFR 50.46.
3. FSAR, Chapter ~~15~~ 14.
4. ~~FSAR, Chapter [15].~~
5. ~~FSAR, Chapter [15].~~

PA.2

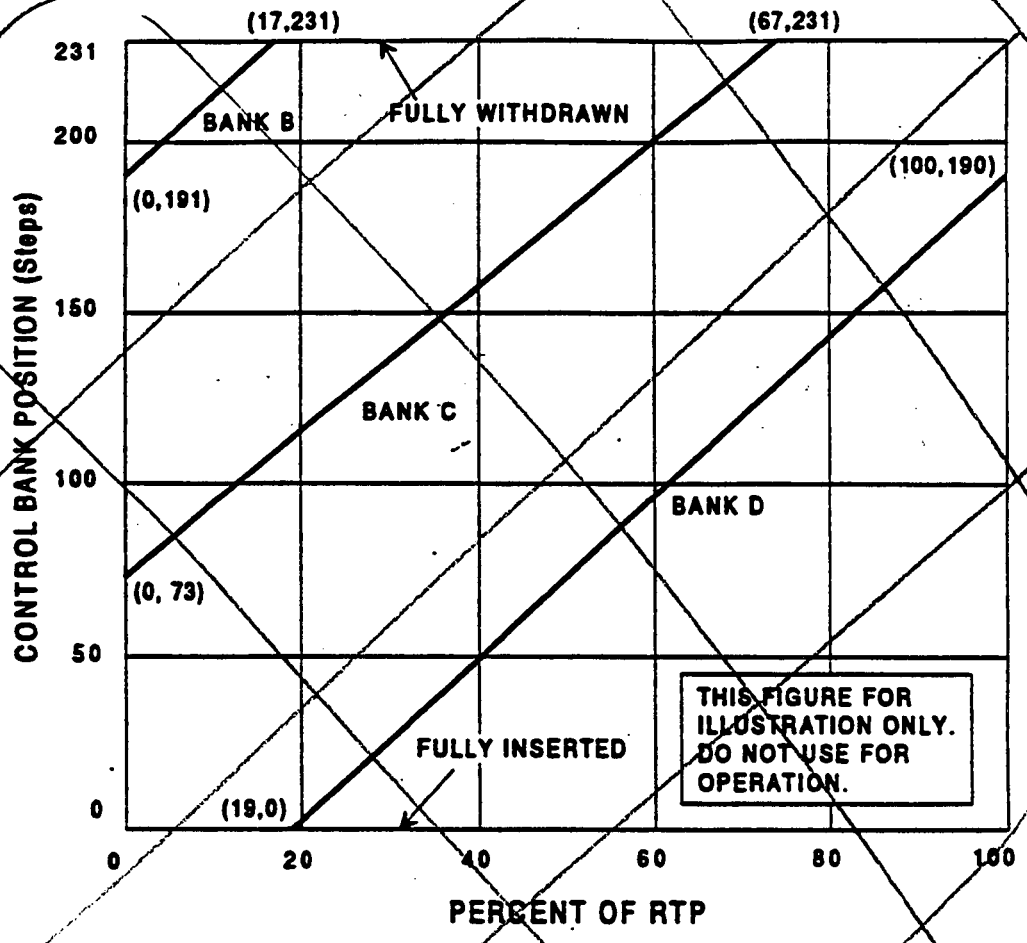

Figure B 3.1.7-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

Indian Point 3
Improved Technical Specifications (ITS)
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Technical Specification 3.1.6:
"Control Bank Insertion Limits"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.6 - Control Bank Insertion Limits

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PA.2 ISTS Basis 3.1.7 states that the control bank insertion limits are specified in the COLR, and provide a figure showing an example of control bank insertion limits for information. The ISTS Basis 3.1.7 figure also indicates how the control banks are moved in an overlap pattern. The ITS basis deletes the figure and references to it since the COLR is readily available to control room operators and other users of the Technical Specifications. Therefore, the inclusion of the figure provides no benefit, and is a potential source of error.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-09, Rev.1 (WOG-04.1), which relocated values for shutdown margin (SDM) to the COLR. SDM is a cycle-

specific variable similar to moderator temperature coefficient, the rod insertion limits, axial flux difference, heat flux hot channel factor, and nuclear rise hot channel factor, which are currently contained in the COLR. In addition, there is an NRC approved methodology for determining SDM.

T.2 This change incorporates Generic Change TSTF-136, Rev. 1 (WOG-59), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, and ISTS 3.1.2, SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F, into ISTS 3.1.1, SHUTDOWN MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR.

T.3 This change incorporates Generic Change TSTF-110, Rev. 2 (WOG-49), which relocated actions (in the form of an increased surveillance frequency) related to several surveillances (rod position deviation monitor, rod insertion limit monitor, AFD monitor, and QPTR alarm) from the Technical Specifications to other licensee controlled documents. The monitors or alarms themselves do not directly relate to the LCO requirements.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.7:
"Rod Position Indication"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Individual Rod Position Indication (IRPI) System and the Demand Position Indication System shall be OPERABLE.

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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One IRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.	8 hours AND Once per 24 hours thereafter
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (condtinued)

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.	8 hours
	OR		
	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1	Verify by administrative means all IRPIs for the affected banks are OPERABLE.	Once per 8 hours
	AND		
	C.1.2	Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart when > 85% RTP and ≤ 18 steps apart when ≤ 85% RTP.	Once per 8 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (condtinued)

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 3.1.7.1	Verify each IRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Prior to reactor crititcality after each removal of the reactor vessel head

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required for rod cluster control assemblies (RCCAs), or rods, to ensure OPERABILITY of position indicators to determine control rod positions and thereby ensure compliance with the 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 SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a 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 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.

BASES

BACKGROUND (Continued)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Individual Rod Position Indication (IRPI) System.

The Bank Demand Position Indication 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 Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The IRPI System provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a coil stack located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained. An indicated misalignment limit of 12 steps precludes a rod misalignment of > 15 inches when instrument error is considered. An indicated misalignment limit of 18 steps precludes a rod misalignment of > 18.75 inches when instrument error is considered.

BASES

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"). 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 rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36. The control rod position indicators monitor rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that one IRPI System and one Bank Demand Position Indication System be OPERABLE for each rod. For the rod position indicators to be OPERABLE, the SR of the LCO and the following must be met:

a. The IRPI System indicates within the required number of steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits";

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

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the IRPI System.

The agreement limit between the Bank Demand Position Indication System and the IRPI System indicates that the Bank Demand

BASES

LCO (continued)

Position Indication 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 in LCO 3.1.4, in position indication for a single rod, ensures high confidence that the position uncertainty of the corresponding rod group is within the assumed values used in the analysis (that specified rod group insertion limits).

These requirements ensure that 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 rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the IRPI 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.

ACTIONS

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

BASES

ACTIONS (continued)

A.1

When one IRPI channel per group fails, the position of the rod can still be determined by use of the 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.

Re-verification every 24 hours thereafter is acceptable because operating experience indicates that significant drift of an individual rod during this interval is not likely and the requirement in required Action B.1 to re-verify within 8 hours if the associated control rod bank is moved significantly during this interval.

Note that an IRPI channel is not inoperable if rod position can be determined using a digital volmeter in lieu of the installed indicators.

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 2).

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.

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

BASES

ACTIONS

B.1 and B.2 (continued)

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 8 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 8 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

With one demand position indicator per bank inoperable (i.e., bank demand position cannot be determined), the rod positions can be determined by the IRPI 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 most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart when > 85% RTP and ≤ 18 steps apart when ≤ 85% RTP within the allowed Completion Time of once every 8 hours is adequate.

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.

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

BASES

ACTIONS

D.1 (continued)

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 IRPI agrees with the demand position within the required number of steps ensures that the IRPI is operating correctly. Only points within the indicated ranges are required in comparison.

This surveillance is performed prior to reactor criticality after each removal of the reactor vessel head because there is a potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A.

2. FSAR, Chapter 14.

3. WCAP-14668, Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3, October 1996 (Proprietary).

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.7:
"Rod Position Indication"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-6	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-10	180	180			
3.10-16	181	181	No TSCRs	No TSCRs for this Page	N/A
T 4.1-1(2)	169	169	No TSCRs	No TSCRs for this Page	N/A

3.10.5 Rod Misalignment Limitations

3.10.5.1 At least once per shift (allowing one hour for thermal soak after rod motion) the position of each control or shutdown rod shall be determined:

a. For operation less than or equal to 85% of rated thermal power, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be less than or equal to 18 steps. A control or shutdown rod indicting a misalignment greater than 18 steps shall be realigned within one hour or the core peaking factors shall be determined within two hours and the requirements of Specification 3.10.2 applied.

SEE ITS 3.1.4

b. For operation greater than 85% of rated thermal power, the indicated misalignment between the group step counter demand position and the analog rod position indicator shall be ±12 steps for less than or equal to 212 steps and +17, -12 steps for greater than 212 steps. A control or shutdown rod indicating a misalignment greater than the above mentioned steps shall be realigned within one hour or the core peaking factors shall be determined within two hours and the requirements of Specification 3.10.2 applied.

3.10.5.2 If the requirements of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned control rod is not realigned within 8 hours, the rod shall be declared inoperable.

LCO 3.1.7

~~3.10.6~~ Inoperable Rod Position Indicator Channels

Cond A

3~~.10.6.1~~ If a rod position indicator channel is out of service, then:

Req. Act A.1

a. For operation between 50 percent and 100 percent of rating, the position of the control rod shall be checked indirectly by core instrumentation (~~excore detectors~~ ~~and/or~~ movable incore detectors) once ~~per 8~~ hours, or subsequent to rod motion exceeding 24 steps, whichever occurs first. (within 8 hours) M.2 within 8 hrs and every 24 hrs after M.2

Req. Act B.1

Req. Act A.2, B.2

b. During operation below 50 percent of rating, no special monitoring is required.

Cond A, Action Note

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time. L.1 A.3

3.10.6.3 ~~If a control rod having a rod position indicator channel out of service, is found to be misaligned from 3.10.6.1a above, then Specification 3.10.5 will be applied.~~ A.4

Add Actions Note A.5

Add LCO 3.1.7 for Demand Position Ind System
Add Condition C and associated Req. Act L.2

Add Condition D and assoc. Req. Act A.6

(e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^N$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available. When a measurement of $F_{\Delta H}^N$ is taken, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the group step counter demand position (operating at greater than 85% of rated thermal power with no accounting for peaking factor margin), or 18.75 inches (operating at less than or equal to 85% of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment greater than 15 inches with consideration of instrumentation error and 18 steps indicated misalignment corresponds to 18.75 inches with instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

A.1

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequency over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or moveable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The assumed control rod drop time in the safety analysis is 2.7 seconds, consisting of 1.80 seconds for normal rod drop time plus additional margin which includes a seismic allowance. The required control rod drop time in Section 3.10.8 is therefore consistent with that assumed in the safety analysis.

REFERENCE

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program," August 1975
2. FSAR Appendix 14C
3. Letter from J.P. Bayne to S.A. Varga dated April 23, 1985, entitled "Proposed Technical Specifications Regarding the Cycle 4/5 Refueling."
4. WCAP-14668, "Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3," October 1996 (Proprietary).

A.1

SEE ITS 3.1.4

A.7

M.1

Prior to reactor critical after head removal

TABLE 4.1-1 (Sheet 2 of 6)

	Channel Description	Check	Calibrate	Test	Remarks
	8. 6.9 KV Voltage	N.A.	18M	Q	Reactor protection circuits only
	6.9 KV Frequency	N.A.	24M	Q	Reactor protection circuits only
SR 3.1.7.1	9. Analog Rod Position	S	24M	M	
	10. Steam Generator Level	S	24M	Q	
	11. Residual Heat Removal Pump Flow	N.A.	24M	N.A.	
	12. Boric Acid Tank Level	S	24M	N.A.	Bubbler tube rodded during calibration
	13. Refueling Water Storage Tank Level				
	a. Transmitter	W	18M	N.A.	Low level alarm
	b. Indicating Switch	W	6M	N.A.	Low level alarm
	14a. Containment Pressure - narrow range	S	24M	Q	High and High-High
	14b. Containment Pressure - wide range	M	18M	N.A.	
	15. Process and Area Radiation Monitoring:				
	a. Fuel Storage Building Area Radiation Monitor (R-5)	D	24M	Q	
	b. Vapor Containment Process Radiation Monitors (R-11 and R-12)	D	24M	Q	
	c. Vapor Containment High Radiation Monitors (R-25 and R-26)	D	24M	Q	
	d. Wide Range Plant Vent Gas Process Radiation Monitor (R-27)	D	24M	Q	

SEE CTS MASTER MARKUP

Amendment No. 8, 38, 68, 68, 74, 93, 107, 128, 137, 140, 144, 148, 150, 154, 169

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.7:
"Rod Position Indication"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.7 - Rod Position Indication

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety because neither are required by 10 CFR 50.36, and neither define nor impose any specific requirements.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.6.2 specifies that not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time. ITS LCO 3.1.7, maintains the same requirement by a combination of Condition A which applies when one IRPI per group is inoperable for one or more groups and by defaulting to ITS LCO 3.0.3 when more than one IRPI per group is inoperable because of the absence of any Condition that applies. This is an administrative

change with no significant adverse impact on safety because there is no change to the existing requirements.

A.4 CTS 3.10.6.3 specifies that CTS 3.10.5, Actions for misaligned rod, applies if a control rod having a rod position indicator channel out of service is found to be misaligned.

ITS LCO 3.1.4, Rod Group Alignment Limits, and ITS LCO 3.1.7, Rod Position Indiction, are both Applicable whenever the plant is in the Modes in which these LCOs apply. Additionally, rod position and rod alignment are not support and/or supported systems governed by ITS LCO 3.0.6. Therefore, there is no need in the ITS for a statement that rod group alignment limits are applicable even if associated rod position indication is not Operable. This is an administrative change with no significant adverse impact on safety because there is no change to the existing requirement.

A.5 ITS 3.1.7 Conditions and Required Actions are preceded by the Note "Separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank." In conjunction with the ITS Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the CTS for inoperable rod position indication. Specifically, this note allows separate entry into an LCO 3.1.7 Condition for each individual rod position indication and separate tracking of Completion Times based on a particular indicator's time of entry into the Condition. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each degraded or inoperable rod position indicator (See ITS 3.1.7, DOC A.3). Complying with the Required Actions for one rod position indicator may allow continued operation, and subsequent degraded or inoperable rod position indicators are governed by separate Condition entry and application of associated Required Actions. This is an administrative change with no impact on safety because any differences between the existing requirements and ITS 3.1.7 are described and justified elsewhere in this discussion of changes.

A.6 CTS 3.10.6 does not specify a Condition or Required Actions if Action or

Completion Times are not met for inoperable rod position indication; therefore, a requirement for immediate reactor shutdown is assumed because rod alignment verifications cannot be performed in accordance with CTS 3.10.5.1. ITS LCO 3.1.7, Condition D and associated Required Actions, is added to require the reactor is in Mode 3 within 6 hours when Conditions or Required Actions for inoperable rod position indication cannot be met. This is an administrative change with no significant adverse impact on safety because there is no change to the existing requirements.

A.7 CTS Table 4.1-1 requires calibrations of analog rod position indication every 24 months. ITS SR 3.1.7.1 maintains this requirement except that ITS SR 3.1.7.1 states that this calibration must be performed by verification that the IRPI agrees with the demand position within 12 steps. This is an administrative change with no significant adverse impact on safety because it is a more explicit statement of the existing requirement.

MORE RESTRICTIVE

M.1 CTS Table 4.1-1 requires calibrations of analog rod position indication every 24 months. ITS SR 3.1.7.1 (as modified by TSTF-89, Rev. 1 (WOG-048)) maintains this requirement except that the SR Frequency is changed to "prior to reactor criticality after each removal of the reactor head." This change may require more or less frequent performance of this SR depending on adherence to the nominal 24 month refueling cycle.

This change is needed and is acceptable because it ties performance of the SR with the activity that is most likely to affect rod position indication adversely. Additionally, it is expected that the SR will in almost all cases be performed within the existing required SR Frequency. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.10.6.1 requires that if a rod position (IRPI) channel is out of service then the position of the control rod shall be checked indirectly by core instrumentation (excore detectors and/or movable incore

detectors). ITS LCO 3.1.7, Actions A.1 and B.1, maintain this requirement except that only the movable incore detectors may be used to verify individual rod position.

In conjunction with this change, the Frequency for re-verification (after the initial check at 8 hours) is extended to every 24 hours. Re-verification every 24 hours after the initial verification is acceptable because operating experience indicates that significant drift of an individual rod during this interval is not likely and the requirement in Required Action B.1 to re-verify within 8 hours if the associated control rod bank is moved significantly during this interval. Additionally, significant deviation is rod position will be evident from excore detector indication as is permitted by the CTS.

This change is needed and is acceptable because the movable incore detectors provide a more precise indication of rod position. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring use of the more precise movable incore detectors to verify the position of a control rod with inoperable individual position indication. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.6.1 specifies that if a rod position indicator channel is out of service, then the position of the control rod shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first. However, no completion time is specified for this action.

ITS LCO 3.1.7, Required Actions A.1 and B.1, maintain this requirement except that Required Action B.1 specifies that verification required after rod motion exceeding 24 steps must be completed within 8 hours (versus an implied requirement in the CTS to initiate action immediately). This change is needed because it eliminates ambiguity and ensures the Required Action is completed within an appropriate time. This change is acceptable because of the low probability that any rod is significantly misaligned as the result of routine rod motion and the low

probability of an event in which a misaligned rod would be significant in the 8 hours allowed to verify rod position. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.10 does not include an explicit requirement for Operability of the rod Demand Position Indication System; however, an implied requirement exists in CTS 3.10.5 to compare the group step counter position to individual rod position indication. CTS 3.10 does not identify any Conditions or Required Actions if the rod demand position indication system for one or more groups is inoperable; therefore, a requirement for immediate reactor shutdown is assumed because rod alignment verifications cannot be performed in accordance with CTS 3.10.5.1.

ITS LCO 3.1.7 requires the Operability of the Demand Position Indication System in Modes 1 and 2. In conjunction with this change, ITS LCO 3.1.7, Condition C and associated Required Actions, specifies the requirements if one demand position indicator per bank is inoperable for one or more banks. Specifically, if one demand position indicator per bank is inoperable for one or more banks, then Required Actions C.1.1 and C.1.2 allow plant operation to continue if every 8 hours it is verified that all IRPIs for the affected banks are Operable and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart when > 85% RTP and within 18 steps of the group step counter demand position when < 85% RTP.

This change is needed and is acceptable because rod group alignment limits and rod insertion limits can be verified to meet the requirements of ITS LCO 3.1.4, 3.1.5 and 3.1.6 with a very high degree of confidence if all IRPIs for the affected banks are Operable and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart when > 85% RTP and within 18 steps of the group step counter demand position when < 85% RTP. Therefore, this change has no significant adverse impact on safety.

<u>REMOVED DETAIL</u>

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.7:
"Rod Position Indication"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.7 - Rod Position Indication

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   ITS LCO 3.1.7 maintains the CTS requirement that if a rod position indicator is out of service, then the position of the rod must be checked indirectly by core instrumentation except that the verification must be completed within 8 hours (versus an implied requirement in the CTS to initiate action immediately).

   This change does not involve a significant increase in the probability of an accident previously evaluated because verification of rod position has no effect of the initiators of any analyzed event. This change does not involve a significant increase in the consequences of an accident previously evaluated because of the low probability that any rod is significantly misaligned as the result of routine rod motion and the low probability of an event in which a misaligned rod would be significant in the 8 hours allowed to verify rod position.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC), or involve changes in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because of the low probability that any rod is significantly misaligned as the result of routine rod motion and the low probability of an event in which a misaligned rod would be significant in the 8 hours allowed to verify rod position.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

CTS 3.10 does not include an explicit requirement for Operability of the rod Demand Position Indication System; however, an implied requirement exists in CTS 3.10.5 comparison of the group step counter position to individual rod position indication is required. CTS 3.10 does not identify any Conditions or Required Actions if the rod demand position indication system for one or more groups is inoperable; therefore, a requirement for immediate reactor shutdown is assumed because rod alignment verifications cannot be performed in accordance with CTS 3.10.5.1.

ITS LCO 3.1.7 requires the Operability of the Demand Position Indication System in Modes 1 and 2. In conjunction with this change, ITS LCO 3.1.7, Condition C and associated Required Actions, specifies the requirements if one demand position indicator per bank is inoperable for one or more banks. Specifically, if one demand position indicator per bank is inoperable for one or more banks, then Required Actions C.1.1 and C.1.2 allow plant operation to continue if every 8 hours it is

verified that all IRPIs for the affected banks are Operable and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart when > 85% RTP and within 18 steps of the group step counter demand position when < 85% RTP.

This change does not involve a significant increase in the probability or consequences of an accident previously evaluated because rod group alignment limits and rod insertion limits can be verified to meet the requirements of ITS LCO 3.1.4, 3.1.5 and 3.1.6 with a very high degree of confidence if all IRPIs for the affected banks are Operable and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart when > 85% RTP and within 18 steps of the group step counter demand position when < 85% RTP.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC), or involve changes in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because rod group alignment limits and rod insertion limits can be verified to meet the requirements of ITS LCO 3.1.4, 3.1.5 and 3.1.6 with a very high degree of confidence if all IRPIs for the affected banks are Operable and the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart when > 85% RTP and within 18 steps of the group step counter demand position when < 85% RTP.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.7:
"Rod Position Indication"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.7

This ITS Specification is based on NUREG-1431 Specification No. **3.1.8**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.2 R1	**010 R1**	REVISE THE CONTROL ROD LCOS APPLICABILITY FROM MODE 2 TO MODE 2 WITH KEFF >= 1.0	Rejected by NRC	Not Incorporated	**N/A**
WOG-048	**089 R0**	CHANGE FREQUENCY OF SR 3.1.8.1	Approved by NRC	Incorporated	**T.2**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.1**
WOG-073 R1		ADD ACTION FOR MORE THAN ONE DRPI INOPERABLE	TSTF Review	Not Incorporated	**N/A**

T.1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~8~~ 7 Rod Position Indication

Individual

<3.10.6>
<DOC L.2>

LCO 3.1.~~8~~ 7 The ~~Digital~~ Rod Position Indication (~~D~~ I RPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

<3.10.6.2>
<DOC A.5>

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.10.6.2> <3.10.6.1a> <DOC A.3> <DOC M.2> A. One ~~D~~ I RPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.	~~Once per~~ 8 hours Insert: 3.1-17-01
	OR	
<3.10.6.1.b>	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
<3.10.6.1.a> <DOC L.1> <DOC M.2> 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.	~~1~~ 8 hours
	OR	
		(continued)

INSERT: 3.1-17-01:

AND

Once per 24 hours thereafter

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.10.6.1.b>	B. (continued)	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
<DOC L.2>	C. One demand position indicator per bank inoperable for one or more banks.	C.1.1	Verify by administrative means all I D RPIs for the affected banks are OPERABLE.	Once per 8 hours
		AND		
	when > 85% RTP and ≤ 18 steps apart when ≤ 85% RTP.	C.1.2	Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours (CLB.1)
		OR		
		C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
<DOC A.6>	D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.~~8~~ 7.1	Verify each ~~D~~ I RPI agrees within ~~[~~12~~]~~ steps of the group demand position for the ~~[~~full indicated range~~]~~ of rod travel.	~~[18 months]~~

<Table 4.1-1 # 9 <Doc M.1>

T.2

Insert: 3.1-19-01

T.2

INSERT: 3.1-19-01:

T.2

Prior to reactor criticality after each removal of the reactor vessel head

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.~~8~~ 7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 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. 7

Insert: B3.1-46-01

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a ~~control~~ rod to become inoperable or to become misaligned from its group. ~~Control~~ Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, ~~control~~ rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on ~~control~~ rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

(continued)

INSERT: B 3.1-46-01

for rod cluster control assemblies (RCCAs), or rods,

BASES

BACKGROUND (continued)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System. Individual

The Bank Demand Position Indication 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 Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 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 DRPI System provides ~~a highly~~ an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

Insert: B3.1-47-01

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

(continued)

INSERT: 3.1-47-01

coil stack located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained.

An indicated misalignment limit of 12 steps precludes a rod misalignment of > 15 inches when instrument error is considered. An indicated misalignment limit of 18 steps precludes a rod misalignment of > 18.75 inches when instrument error is considered.

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.~~7~~ 6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.~~5~~ 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.

10 CFR 50.36

The ~~control~~ rod position indicator channels satisfy Criterion 2 of ~~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.~~8~~ 7 specifies that one ~~D~~ I RPI System and one Bank Demand Position Indication System be OPERABLE for each ~~control~~ rod. For the ~~control~~ rod position indicators to be OPERABLE, ~~requires meeting~~ the SR of the LCO and the following:

must be met

a. ~~I~~ The ~~D~~ I RPI System indicates within ~~12~~ 12 steps of the group step counter demand position as required by LCO 3.1.~~5~~ 4, "Rod Group Alignment Limits";

b. For the ~~D~~ I RPI System there are no failed coils; and

The specified number of

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the ~~D~~ I RPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the ~~D~~ I RPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.~~5~~ 4, in position indication for a single ~~control~~ rod, ensures high confidence that the position uncertainty of the corresponding ~~control~~ rod group is within the assumed values used in the analysis (that specified ~~control~~ rod group insertion limits).

These requirements ensure that ~~control~~ rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

(continued)

BASES

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 DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.~~5~~ 4, LCO 3.1.~~6~~ 5, and LCO 3.1.~~7~~ 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.

ACTIONS

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

A.1

When one DRPI channel per group fails, the position of the rod can still be determined by use of the 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.

Insert: B3.1-49-01

(continued)

INSERT: B 3.1-49-01

Re-verification every 24 hours thereafter is acceptable because operating experience indicates that significant drift of an individual rod during this interval is not likely and the requirement in Required Action B.1 to re-verify within 8 hours if the associated control rod bank is moved significantly during this interval.

Note that an IRPI channel is not inoperable if rod position an be determined using a digital voltmeter in lieu of the installed indicators.

BASES

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~~ 2).

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.

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 ~~[#]~~ 8 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 ~~[#]~~ 8 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

(i.e., bank demand position cannot be determined)

With one demand position indicator per I bank inoperable, the rod positions can be determined by the DRPI 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 most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart (when > 85% RTP and ≤ 18 steps apart when ≤ 85% RTP) within the allowed Completion Time of once every 8 hours is adequate.

(continued)

BASES

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.

SURVEILLANCE REQUIREMENTS

SR 3.1.~~8~~ 7.1

Verification that the DRPI agrees with the demand position within [12] steps ensures that the DRPI 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.

the required number of

Insert: B3.1-51-01

~~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.~~

T.2

REFERENCES

1. 10 CFR 50, Appendix A, ~~GDC 13~~.
2. FSAR, Chapter [15]. 14
3. ~~FSAR, Chapter [15]~~

Insert: B3.1-51-02

INSERT: B 3.1-51-01:

T.2

This surveillance is performed prior to reactor criticality after each removal of the reactor vessel head because there is a potential for unnecessary plant transients if the SR were performed with the reactor at power.

INSERT: B 3.1-51-02:

3. WCAP-14668, Conditional Extension of the Rod Misalignment Technical Specification for Indian Point Unit 3, October 1996 (Proprietary).

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.7:
"Rod Position Indication"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.7 - Rod Position Indication

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG-1431, Rev 1, Section 3.1.7, was modified as needed to reflect the IP3 design and current licensing basis. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-059, Rev. 1 (WOG-136), which combines ISTS 3.1.1, SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F, and ISTS 3.1.2, SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F, into ISTS 3.1.1, SHUTDOWN

MARGIN (SDM). This change is necessary because ISTS 3.1.1 and ISTS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin to the COLR. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates Generic Change TSTF-89, Rev.1 (WOG-048), which changes frequency of SR 3.1.8.1 from 18 months to "Once prior to criticality after each removal of the reactor vessel head. " This SR verifies that each IRPI agrees within specified limits of the group demand position for the full indicated range of rod travel. This surveillance is performed during a plant outage or plant startup since there is potential for unnecessary plant transients if the SR is performed with the reactor at power. By not specifying a fixed frequency for this SR, any unit shutdown and reactor vessel head removal would require that the SR be performed again to verify that the operability of the rod position indicator systems has not been affected.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.8:
"PHYSICS TESTS Exceptions MODE 2"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
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, provided:

a. RCS lowest loop average temperature is ≥ 540°F; and

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

c. THERMAL POWER IS ≤ 5% RTP.

APPLICABILITY: MODE 2 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
B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 ≥ 540°F.	30 minutes
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions – MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program (Ref. 3) 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; and

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

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 are listed in Reference. 4.

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

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). These PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature

BASES

APPLICABLE SAFETY ANALYSES (continued)

"Coefficient (MTC)," LCO 3.1.4, "Group Rod Alignments", 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", 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 ≥ 540°F, and SDM is kept within the limits specified in the COLR for low power physics tests.

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 Rod Cluster Control Assemblies (RCCAs) or control rods (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. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36.

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 MTC to be outside its specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

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 during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 540 °F;

BASES

LCO (continued)

b. SDM is within the limit specified in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

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.

B.1

When THERMAL POWER is > 5% RTP, as indicated on power range instruments, 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 < 540°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

BASES

ACTIONS

C.1 (continued)

with the reactor critical and with temperature below 540°F could violate the assumptions for accidents analyzed in the safety analyses.

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 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 Protection System (RPS) Instrumentation." The frequency is specified in LCO 3.3.1. A CHANNEL OPERATIONAL TEST is normally performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RPS 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 ≥ 540°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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

Verification that THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analysis. Verification of 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

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

BASES

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.

6. WCAP-11618, including Addendum 1, April 1989.

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Technical Specification 3.1.8:
"PHYSICS TESTS Exceptions MODE 2"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.1-25	149	149	No TSCRs	No TSCRs for this Page	N/A
T 3.5-2(1)	93	93	No TSCRs	No TSCRs for this Page	N/A
3.10-4	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-5	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-7	160	160	No TSCRs	No TSCRs for this Page	N/A

SEE ITS 3.1.3

C. MINIMUM CONDITIONS FOR CRITICALITY

LCO 3.1.8

~~1.~~ Except during low power physics test, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

~~2. This section intentionally deleted.~~

SEE ITS 3.4.2

LCO 3.1.8.a
Cond C & Req. Act C.1
Req. Act D.1

3. At all times during critical operation, the lowest loop T_{avg} shall be no lower than 540 °F.
   a. If T_{avg} is less than 540°F when the reactor is critical, restore T_{avg} to $\geq$ 540 °F within 15 minutes or be in hot shutdown within the following 15 minutes.

SEE ITS 3.4.9

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. [1] [2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range. [1] [2] Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions. A.1

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of an increase in moderator temperature. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical except when T_{avg} is $\geq$ 540 °F provides assurance that an overpressure event will not occur whenever the reactor vessel is in the nil-ductility temperature range and that the reactor is operated within the bounds of the safety analyses. The safety analyses, which assume a critical temperature of 547 °F, are applicable for critical temperatures as low as 540 °F. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. The Surveillance requirement to support this specification is provided in Table 4.1-1 item no. 4.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the reactor coolant not be solid when criticality is achieved.

References:

1. FSAR Table 3.2-1

2. FSAR Figure 3.2-9

A.1 A.2

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

ITS 3.1.8 A.2

Applicability:

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

Objectives:

To ensure:

1. Core subcriticality after reactor trip.

2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

Add Completion Time Req. Act A.1, A.3

~~3.10.1~~ **Shutdown Reactivity**

LCO 3.1.8.b ~~3.10.1.1~~ Whenever T_{avg} > 200°F (including Mode 2 during Physics Tests) the shutdown margin shall ~~be ≥ 1.3% Δk/k~~ LA.1

~~3.10.1~~.2 When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit.

3.10.2 **Power Distribution Limits** A.4

3.10.2.1 At all times, ~~except during low power physics tests~~, the hot channel factors defined in the basis must meet the following limits:

SEE ITS 3.2.1 ITS 3.2.2

$$F_Q(Z) \le (F_Q^{RTP}/p) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \le (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \le 0.5$$

$$F_{\Delta H}^{N} \le F_{\Delta H}^{RTP}(1 + PF_{\Delta H}(1-P))$$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

Amendment No. ~~13~~, ~~49~~, ~~48~~, ~~61~~, ~~73~~, ~~88~~, ~~103~~, 112

Add Req. Act A.2 M.2

Add LCO 3.1.8.c
Add Condition B and associated Req. Actions M.1

SEE ITS 3.2.3

3.10.2.8 Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

3.10.2.9 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

SEE ITS 3.2.4

3.10.3 Quadrant Power Tilt Limits

A.4

3.10.3.1 When ever the indicated quadrant power tilt ratio exceeds 1.02, ~~except for physics tests~~ within two hours the tilt condition shall be eliminated or the following actions shall be taken:

a) Restrict core power level and reset the power range high flux setpoint three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0,

and

b) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be reset to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action.

A.4

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ration exceeds 1.09 and there is simultaneous indication of a misaligned control rod, restrict core power level 3% of rated value for every percent of indicated power tilt ratio exceeding 1.0 and realign the rod within two hours. If the rod is not realigned within two hours or if there is no simultaneous indication of a misaligned rod, the reactor shall be brought to the hot shutdown condition within 4 hours. If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt.

SEE ITS 3.1.7 3.10.3.3	The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.
SEE ITS 3.2.4 3.10.3.4	The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.
3.10.4	Rod Insertion Limits
SEE ITS 3.1.5 3.10.4.1	The shutdown rods shall be fully withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of Specification 3.10.1).
SEE ITS 3.1.6 3.10.4.2	When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.
SEE ITS 3.1.6 ITS 3.1.4 3.10.4.3	Control bank insertion shall be further restricted if: a) The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown. b) A rod is inoperable (Specification 3.10.7).
LCO 3.1.8 Exemption from LCO 3.1.5 and LCO 3.1.6. ~~3.10.4.4~~	Control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin required by Specification 3.10.1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one control rod inserted.

3.10.7 Inoperable Rod Limitations

SEE ITS 3.1.4

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5 or fails to meet the requirements of 3.10.8.

LCO 3.1.8

~~3.10.7.2~~ Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

SEE ITS 3.1.4

3.10.7.3 If any rod has been declared inoperable, then the potential ejected rod worth, associated transient power distribution peaking factors and the accident listed in Table 3.10-1 shall be analyzed within 5 days, or the reactor brought to the hot shutdown condition using normal operating procedures. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time to each control rod shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry.

TABLE 3.5-2 (Sheet 1 of 3)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

	NO. FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*
SEE ITS 3.3.1	1. Manual Reactor Trip	2	1	1	0	Maintain hot shutdown
	2. Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
		4	2	2	1	For zero power physics tests only (A.5)
	3. Overtemperature ΔT	4	2	3	2	Maintain hot shutdown
	4. Overpower ΔT	4	2	3	2	Maintain hot shutdown
SEE ITS 3.3.1	5. Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown
	6. Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
	7. Pressurizer-Hi Water Level	3	2	2	1	Maintain hot shutdown
	8. Low Flow One Loop (Power ≥ P-8)	3/loop	2/loop (any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops (Power < P-8 and ≥ P-10)	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	Maintain hot shutdown

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Technical Specification 3.1.8:
"PHYSICS TESTS Exceptions MODE 2"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.1.8 - PHYSICS TESTS Exceptions - MODE 2

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.1.1 establishes requirements for SDM and no exception is provided for physics tests. CTS 3.10.1.2 specifies that if the shutdown margin (SDM) requirements in CTS 3.10.1.1 are not met during physics tests, then boration must be initiated. No completion time for the required action is specified, so a completion time of zero is assumed in accordance with CTS 3.0. Under the same conditions, ITS 3.1.8, Required Action A.1, specifies that boration must be initiated within 15 minutes.

This change is needed because 15 minutes provides a reasonable time for an operator to align systems and initiate injection of boron. This is an administrative change because the new requirements are consistent is a reasonable interpretation of the existing requirements and ensure that the appropriate action is pursued without delay and in a controlled manner. Therefore, this change has no significant adverse impact on safety.

A.4 CTS 3.10.2.1 specifies that the power distribution limits, $F_Q(Z)$ and $F^N_{\Delta H}$, are not required to be met during physics tests. Additionally, CTS 3.10.3.1 specifies that quadrant power tilt ratio limits are not applicable during physics tests.

ITS LCO 3.1.8 does not include exceptions from LCOs governing power distribution limits. This change is acceptable because ITS LCO 3.1.8 provides exemptions for physics tests performed in Mode 2 only and the referenced LCOs are applicable only in Mode 1. Therefore, there is no need for ITS LCO 3.1.8 to provide any exemptions from ITS LCO 3.2.1, Heat Flux Hot Channel Factor. ITS LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), ITS LCO 3.2.3, Axial Flux Difference (AFD), or ITS LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR). This is an administrative change with no adverse impact on safety.

A.5 CTS Table 3.5-2, Item 2, specifies the minimum required channels and minimum level of redundancy for the nuclear flux power range trip. This requirement includes an exemption for physics testing that permits disabling one channel of the trip function so that the instrument can be used to support physics testing. This allowance is maintained in ITS LCO 3.3.1.

MORE RESTRICTIVE

M.1 Various CTS LCOs provide exemptions from LCOs for the performance of physics testing and, although some of these exemptions specify low power physics tests, no reactor power level limit is specified as a prerequisite for using these exemptions from LCOs.

ITS LCO 3.1.8 maintains the ability to take exemptions from selected LCOs but includes the restriction that these exemptions are applicable only if reactor power is maintained ≤ 5% RTP during the tests. In conjunction with this change, ITS LCO 3.1.8, Condition B and associated Required Actions, requires that reactor trip breakers are opened immediately to prevent operation of the reactor beyond its design limits if reactor power goes above 5% during physics tests. Additionally, ITS SR 3.1.8.3 is added to verify every 30 minutes that reactor power is less than 5% during physics tests.

This change is needed because all of the physics tests for which these exemptions are designed can be performed in Mode 2. This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring a more conservative limit for performing physics tests. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.10.1.1 establishes requirements for SDM and no exception is provided for physics tests. CTS 3.10.1.2 specifies that if the shutdown margin (SDM) requirements in CTS 3.10.1.1 are not met, then boration must be initiated. There is no specific requirement to terminate physics testing exceptions. Under the same conditions, ITS 3.1.8, Required Action A.2, requires that the physics test exemptions from ITS LCOs provided by ITS LCO 3.1.8 be suspended within 1 hour. This change is needed because 1 hour provides a reasonable time to restore from the physics test exemptions and/or initiate Actions under the applicable LCO. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while ensuring Technical Specification exceptions needed to support physic testing are terminated when the required conditions are not met.

TECHNICAL CHANGES - LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 3.10.1.1 requires that SDM be "≥ 1.3% Δk/k" with no explicit exemption for physics tests. ITS LCO 3.1.1 and ITS 3.1.8 require that SDM be maintained within the limits specified in the Core Operating Limits Report(COLR). This change allows the specific limits for shutdown margin to be removed from the ITS and relocated to the Core Operating Limits Report (COLR). This change is needed because the specific value for SDM is a cycle-specific variable. Therefore, by maintaining the SDM value in the COLR, the core reload design can be completed after shutdown when the actual end of cycle burnup is known. This saves redesign efforts that occur if actual burnup differs from the projected value.

This change is acceptable because ITS LCO 3.1.1 and ITS LCO 3.1.8 maintain the requirement to meet SDM requirements and ITS 5.6.5, Core Operating Limits Report (COLR), includes detailed requirements that ensure SDM limits will be properly established and maintained. Requirements established by ITS 5.6.5 include the following:

a. The analytical methods used to determine the core operating limits (including the SDM) must be those previously reviewed and approved by the NRC. The approved documents that document this approved methodology must be listed in ITS 5.6.5 and can be changed only with a TS change.

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

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications. Additionally, an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

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Technical Specification 3.1.8:
"PHYSICS TESTS Exceptions MODE 2"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.1.8 - PHYSICS TESTS Exceptions - MODE 2

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

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Technical Specification 3.1.8:
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PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.1.8

This ITS Specification is based on NUREG-1431 Specification No. **3.1.10**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-004.1 R1	**009 R1**	RELOCATE VALUE FOR SHUTDOWN MARGIN TO COLR	Approved by NRC	Incorporated	**T.1**
WOG-004.4	**012 R0**	DELETE LCO 3.1.9 AND 3.1.11 (PHYSICS TESTS EXCEPTIONS)	See Next Rev.	Superceded-See Next Rev	**N/A**
WOG-004.4 R1	**012 R1**	DELETE LCO 3.1.9 AND 3.1.11 (PHYSICS TESTS EXCEPTIONS)	Approved by NRC	Incorporated	**T.3**
WOG-004.6 R2	**014 R2**	ADD AN LCO ITEM AND SR TO MODE 2 PHYSICS TESTS EXCEPTIONS TO VERIFY THAT THERMAL POWER <= 5% RTP.	See Next Rev.	Superceded-See Next Rev	**N/A**
WOG-004.6 R3	**014 R3**	ADD AN LCO ITEM AND SR TO MODE 2 PHYSICS TESTS EXCEPTIONS TO VERIFY THAT THERMAL POWER <= 5% RTP.	Superceded-See Next Rev	Superceded-See Next Rev	**N/A**

Technical Specification 3.1.8:
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WOG-004.6 R4	**014 R4**	ADD AN LCO ITEM AND SR TO MODE 2 PHYSICS TESTS EXCEPTIONS TO VERIFY THAT THERMAL POWER <= 5% RTP.	Approved by NRC	Incorporated	**T.4**
WOG-044 R1	**108 R1**	ELIMINATE THE 12 HOUR COT ON POWER RANGE AND INTERMEDIATE RANGE CHANNELS FOR PHYSICS TEST EXCEPTIONS	Approved by NRC	Incorporated	**T.5**
WOG-044 RO	**108 R0**	ELIMINATE THE 12 HOUR COT ON POWER RANGE AND INTERMEDIATE RANGE CHANNELS FOR PHYSICS TEST EXCEPTIONS	See Next Rev	Superceded-See Next Rev	**N/A**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**

T.2
T.3

3.1 REACTIVITY CONTROL SYSTEMS

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

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

<3.1.C.1> 3 LCO 3.1.~~4~~, "Moderator Temperature Coefficient (MTC)";
<3.10.7.2> 4 LCO 3.1.~~5~~, "Rod Group Alignment Limits";
<3.10.4.4> 5 LCO 3.1.~~6~~, "Shutdown Bank Insertion Limits";
<3.10.4.4> 6 LCO 3.1.~~7~~, "Control Bank Insertion Limits"; and
<3.1.C.1><DOC A5> LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

<3.1.C3> a. RCS lowest loop average temperature is ≥ ~~531~~ 540 °F; and

<3.10.1.1> b. SDM is ≥ ~~1.6% Δk/k~~

Insert: 3.1-23-01

Insert: 3.1-23-02

T.1

APPLICABILITY: MODE 2 during PHYSICS TESTS.
<DOC M.1>

T.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit. <3.10.1.2> <DOC A.3>	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	AND	
<3.10.1.2> <DOC M.2>	A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit. <DOC M.1>	B.1 Open reactor trip breakers.	Immediately

(continued)

INSERT: 3.1-23-01

within the limits specified in the COLR; and (T.1)

INSERT: 3.1-23-02

<DOC M.1> c. THERMAL POWER is ≤ 5% RTP. (T.4)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.C.3.a C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
3.1.C.3.a 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~~rior to initiation of PHYSICS TESTS
SR 3.1.~~10~~ 8.2	Verify the RCS lowest loop average temperature is ≥ ~~[531]~~ 540 °F.	30 minutes
SR 3.1.~~10~~ 8.~~3~~ 4	Verify SDM ~~is ≥ 1.6% Δk/k~~	24 hours

T.5 T.4 T.1

Insert: 3.1-24-01

Insert: 3.1-24-02

INSERT: 3.1-24-01

within the limits specified in the COLR.

INSERT: 3.1-24-02:

T.4

SR 3.1.8.3 Verify THERMAL POWER is ≤ 5% RTP.	30 minutes

B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response; and

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. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include

(continued)

BASES

BACKGROUND (continued)

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. 4)~~ ~~in MODE 2~~ are listed ~~below~~: in Reference 4.

a. Critical Boron Concentration—Control Rods Withdrawn;

b. Critical Boron Concentration—Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

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 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, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration—Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron

(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 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.~~5~~ 4, "Rod Group Alignment Limits"; LCO 3.1.~~6~~ 5, "Shutdown Bank Insertion Limit"; or LCO 3.1.~~7~~ 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 Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.~~5~~ 4, LCO 3.1.~~6~~ 5, or LCO 3.1.~~7~~ 6.

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

(continued)

BASES

BACKGROUND (continued)

performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.~~5~~ 4, LCO 3.1.~~6~~ 5, or LCO 3.1.~~9~~ 6. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

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~~ These PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. ~~Tables [14.1-1 and 14.1-2] summarize the zero, low power, and power tests.~~ Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.~~4~~ 4, "Moderator Temperature Coefficient (MTC)," LCO 3.1.~~5~~ 5, LCO 3.1.~~6~~ 6, LCO 3.1.~~7~~ 7, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ~~≥ 531°F, and~~ ≥540°F ~~SDM is ≥ [1.6]% Δk/k~~.

Insert: B 3.1-64-01

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~~ Rod Cluster Control Assemblies (RCCAs) or control rods (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. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

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~~ MTC ~~and minimum temperature for criticality~~ to be outside ~~their~~ its specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified

(continued)

INSERT: B 3.1-64-01

and SDM is kept within the limits specified in the COLR for low power physics tests.

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.~~4~~ 3, LCO 3.1.~~5~~ 4, LCO 3.1.~~6~~ 5, LCO 3.1.~~7~~ 6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ ~~[531]~~ 540 °F; and

Insert: B 3.1-65-01

b. SDM is ~~≥ [1.6]% Δk/k~~ T.1 T.4

Insert: B 3.1-65-02

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. ~~Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1."~~

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.

B.1

as indicated on power range instrument

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.

(continued)

INSERT: B 3.1-65-01

within the limit specified in the COLR;

INSERT: B 3.1-65-02

c. THERMAL POWER is ≤ 5% RTP.

BASES

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 531 540 °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 531 540 °F could violate the assumptions for accidents analyzed in the safety analyses.

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 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.~~10~~ 8.1 RPS The Frequency is specified in LCO 3.3.1. PA

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor ~~Trip~~ Protection System (RTS) Instrumentation." ~~A CHANNEL OPERATIONAL TEST is~~ normally performed on each power range and intermediate range channel ~~within 12 hours~~ 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. T.5 ~~The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.~~ T.5

SR 3.1.~~10~~ 8.2

Verification that the RCS lowest loop T_{avg} is ≥ 531 540 °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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.~~10~~ 8.2 (continued)

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

Insert: B 3.1-67-01

SR 3.1.~~10~~ 8.~~3~~ 4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation ~~because~~ when the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

(continued)

INSERT: B 3.1-67-01:

Verification that THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analysis. Verification of 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.

BASES

REFERENCES (continued)

5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.

6. WCAP-11618, including Addendum 1, April 1989.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.1.8:
"PHYSICS TESTS Exceptions MODE 2"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.1.8 - PHYSICS TESTS Exceptions – MODE 2

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-09 (WOG-04.1), Rev.1, which relocates value for shutdown margin during physics tests to COLR if the plant cycle specific analysis documented in the COLR specifically supports this relaxation. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates Generic Change TSTF-136, Rev. 1 (WOG-59), Rev.1, which combines ITS 3.1.1,Shutdown Margin (SDM) – T_{avg} > 200°F, and ITS 3.1.2, Shutdown Margin (SDM) – T_{avg} ≤ 200°F, into ITS 3.1.1, Shutdown Margin (SDM). This change is necessary because ITS 3.1.1 and ITS 3.1.2

became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin during physics tests to COLR. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T3. This change incorporates Generic Change TSTF-012 (WOG-4.4), Rev.1, which deletes LCO 3.1.9, "Physics Tests Exceptions - Mode 1" and LCO 3.1.11, "Shutdown Margin (SDM) Test Exceptions" and renumbers LCO 3.1.10 to 3.1.9. Elimination of LCO 3.1.9 is acceptable because the physics tests that LCO 3.1.9 permits are RCCA pseudo ejection test, RCCA pseudo drop and misalignment test, and xenon stability measurements. These physics tests were performed only in initial startup testing programs for some plants. Elimination of LCO 3.1.11 is acceptable because the physics test that LCO 3.1.11 permits is the rod worth measurement in the N-1 condition. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception can be deleted. LCO 3.0.7 is revised to reflect the elimination of the Test Exception LCOs. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.4 This change incorporates Generic Change TSTF-014 (WOG-4.6), Rev.4, which adds an LCO requirement and SR to Mode 2 Physics Tests Exceptions to verify that Thermal Power <= 5% RTP during physics testing. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.5 This change incorporates Generic Change TSTF-108, Rev.1 (WOG-044), which eliminates the 12 hour Channel Operational Test (COT) on power range and intermediate range channels "within 12 hours prior to initiation of Physics Tests" regardless of whether the COT has been performed within its required frequency for RTS. This change eliminates those surveillances. This change is acceptable because performance of a COT on power range and intermediate range channels is required by LCO 3.3.1, RTS Instrumentation, every 92 days (SR 3.3.1.7 and SR 3.3.1.8). The 92 day required frequency has been determined to be sufficient for verification that the power range and intermediate range monitors are properly functioning. SR 3.1.8.1 requires a COT within 12 hours prior

to initiation of Physics Tests regardless of whether the COT has been performed within its required frequency. Initiation of Physics Tests does not impact the ability of the monitors to perform their required function, does not affect the trip setpoints or RTS trip capability, and does not invalidate previous surveillances. Therefore, an additional surveillance required to be performed "prior to" this event is an unnecessary performance of a surveillance. Additionally, this surveillance is not related to any LCO requirement and has no appropriate condition to enter upon failure to meet the surveillance. Therefore, deletion of the surveillance, relying on the COT surveillance specified within the RTS Instrumentation LCO, enhances the proper utilization of the ITS.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3

Improved

Technical Specifications

Conversion Submittal

Volume 4

New York Power Authority

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q(Z)$ exceeds limit.	15 minutes
	AND		
	A.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours
	AND		
	A.3	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q(Z)$ was last verified AND 31 EFPD thereafter

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36.

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

BASES

LCO (continued)

$$F_Q(Z) \leq \frac{FQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{FQ}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

where: FQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The current IP3 specific values of FQ and K(Z) are given in the COLR.

An $F_Q(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q(Z) = F_Q^M(Z)\ 1.0815$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty. This correction factor for the measured value of total peaking factor $F_Q^M(Z)$ is for the three percent needed to account for manufacturing tolerances and this value is further increased by five percent to account for measurement error.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures exceeding 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$

BASES

LCO (continued)

limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

BASES

ACTIONS (continued)

A.3

Verification that $F_Q(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If Required Actions A.1 through A.3 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 is modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q(Z)$ is within specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which it was last verified to be within specified limits. Because $F_Q(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the

BASES

SURVEILLANCE REQUIREMENTS (continued)

Frequency condition requiring verification of $F_Q(Z)$ following a power increase of more than 10%, ensures that it was verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

Verification that $F_Q(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q(Z) = F_Q^M(Z)$ 1.0815 (Ref. 4). $F_Q(Z)$ is then compared to its specified limits.

The limit with which $F_Q(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because the highest peaking factors (i.e., the ratio of local power density to the core average power density) generally decrease as core average power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

REFERENCES

1. 10 CFR 50.46, 1974.

2. FSAR 14.2.6.

3. 10 CFR 50, Appendix A.

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties".

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-1	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-2	112	112	IPN 96-063	Leakage Limits for RCS and SIS	
3.10-8	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-8a	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-9	175	175	No TSCRs	No TSCRs for this Page	N/A
3.10-10	180	180			
F 3.10-1	112	112	No TSCRs	No TSCRs for this Page	N/A
F 3.10-2	143	143	No TSCRs	No TSCRs for this Page	N/A
F 3.10-3	14	14	No TSCRs	No TSCRs for this Page	N/A

A.1

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

A.2

Applicability:

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

Objectives:

To ensure:

1. Core subcriticality after reactor trip.

2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

SEE ITS 3.1.1

3.10.1 Shutdown Reactivity

3.10.1.1 Whenever T_{avg} > 200°F the shutdown margin shall be ≥ 1.3% Δk/k.

3.10.1.2 When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit.

~~3.10.2~~ Power Distribution Limits

Mode 1 A.3

LCO 3.2.1 Applicability

~~3.10.2.1~~ At all times ~~except during low power physics tests~~, the hot channel factors defined in the basis must meet the following limits: A.4

LCO 3.2.1

shall be within limits in COLR LA.1

$F_Q(Z) \le (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) \le (F_Q^{RTP}/0.5) \times K(Z)$ for $P \le 0.5$

SEE ITS 3.2.2

$F_{\Delta H}^N \le F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

LA.1

A.1

M.1

Add Note to Surveillances

COLR, Z is the core height location of F_Q, F_Q^{RTP} is the F_Q limit at Rated Thermal Power (RTP) specified in the COLR, $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR. LA.1 A.5

SR 3.2.1.1 ~~3.10.2.2~~ ~~Following initial core loading~~, subsequent reloading and at regular effective full power monthly intervals thereafter, power distribution maps, ~~using the movable detector system~~, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison, LA.1

Prior to > 75% RTP AND 12 hrs after pwr increase M.1

3.10.2.2.1 The measurement of total peaking factor F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

~~3.10.2.2.2~~ When $F_{\Delta H}^N$ is measured, no additional allowances are necessary prior to comparison with the limits of section 3.10.2. An error allowance of 4% has been included in the limits of section 3.10.2. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, ~~the~~ reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q ~~or $F_{\Delta H}^N$~~ limit to measured value, whichever is less. ~~If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.~~

M.2 within 15 min

L.1 within 72 hours

Req Act A.1
Req Act A.2

A.L

A.9

SEE ITS 3.2.3

3.10.2.3 The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux differences must be updated each effective full power month by linear interpolation using the most recent measured value and a value of 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference of all but one operable excore channel shall be maintained within the band specified in the COLR about the target flux difference.

Add Req. Action A.3 A.7

Add Req Action B.1 A.8

3.10.9 **Rod Position Monitor**

SEE ITS 3.1.4

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per 8 hours and after a load change greater than 10 percent of rated power.

3.10.10 **Reactivity Balance**

SEE ITS 3.1.2

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3.10.11 ~~**Notification**~~

~~Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.~~

A.10

~~**Basis**~~

~~Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR in normal operation or in short term transients.~~

~~In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant~~

A.1

A.1

accident analysis based on the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident analyses. To aid in specifying the limits on power distribution, the following hot channel factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

A.1

F_Q^E Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

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

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

An upper bound envelope of F_Q^{RTP} specified in the COLR times the normalized peaking factor axial dependence of K(Z) specified in the COLR has been determined consistent with Appendix K criteria and is satisfied for OFA transition mixed cores [3] by all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analysis based on this upper bound normalized envelope, K(Z), specified in the COLR demonstrates that the peak clad temperature is below the peak clad temperature limit of 2200°F.[2]

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

In the specified limit of $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq F_{\Delta H}^{RTP}/1.04$, where $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^N$ limit at Rated Thermal Power specified in the COLR. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

(e.g. rod misalignment) affect $F_{\Delta H}^{N}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^{N}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}^{N}$ is less readily available. When a measurement of $F_{\Delta H}^{N}$ is taken, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the group step counter demand position (operating at greater than 85% of rated thermal power with no accounting for peaking factor margin), or 18.75 inches (operating at less than or equal to 85% of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment greater than 15 inches with consideration of instrumentation error and 18 steps indicated misalignment corresponds to 18.75 inches with instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

A.1

FIGURE 3.10-1

HAS BEEN

DELETED

A.1

DELETED

A.1

~~Figure 3.10-2 Hot Channel Factor Normalized Operating Envelope~~

A.1

Figure 3.10 - 3

DELETED

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.2.1 specifies that the power distribution limit $F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is applicable "at all times." ITS LCO 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$, is applicable in Mode 1 (i.e., when thermal power is greater than 5% RTP). This is an administrative change because CTS 3.10.2.2 allows unlimited operation for physics testing when $F_Q(Z)$ limits are not met and, consistent with

ITS 3.1.8, physics testing is performed in Mode 2 only. Therefore, the implied CTS applicability for $F_Q(Z)$ is Mode 1. This change is acceptable because $F_Q(Z)$ is a local peaking factor limit and there are very substantial margins for $F_Q(Z)$ limits when in Mode 2 assuming ITS LCO 3.1.4, Rod Group Alignment Limits, are met and thermal power is less than 5% RTP. This is an administrative change with no adverse impact on safety because the power distribution limit specified in CTS 3.10.2.1 and ITS LCO 3.2.1 cannot be exceeded except when in Mode 1 if other Applicable LCOs are being met.

A.4 CTS 3.10.2.1 specifies that the power distribution limit $F_Q(Z)$ is not required to be met "during low power physics tests." ITS LCO 3.2.1 does not state this exception because the limits on power distribution are applicable only in Mode 1 (i.e., when thermal power is greater than 5% RTP) and, as specified in ITS LCO 3.1.8, Physics Tests Exceptions, physics tests may be performed in Mode 2 only. The applicability requirements for ITS LCO 3.2.1 (Mode 1) and ITS LCO 3.1.8 (Mode 2) eliminates the need for an exemption from ITS LCO 3.2.1 for physics testing. Therefore, this is an administrative change with no adverse impact on safety.

A.5 CTS 3.10.2.2 includes "following 'initial' core loading" as one of the required SR Frequencies for verifying $F_Q(Z)$ limits are met.
ITS SR 3.2.1.1 maintains the requirement for periodic verification of $F_Q(Z)$; however, ITS SR 3.2.1.1 does not specify following initial core loading as a required Frequency. Deletion of the requirement to perform these tests "following initial core loading" is an administrative change with no impact on safety because initial fuel loading was a one time event that has been completed.

A.6 CTS 3.10.2.2.2 specifies Actions if limits for either $F_Q(Z)$ or $F^N_{\Delta H}$ are exceeded. ITS LCO 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$), establishes the Required Actions if limits for $F_Q(Z)$ are exceeded. ITS LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), establishes the Required Actions if limits for $F^N_{\Delta H}$ are exceeded. This is an

administrative change with no significant adverse impact on safety because there are no changes to the existing requirements except as identified and justified in ITS 3.2.1 and ITS 3.2.2.

A.7 CTS 3.10.2.2.2 requires a proportional reduction in reactor power and high flux trip setpoints if limits for $F_Q(Z)$ are not met. However, there are no explicit provisions for returning to full power if appropriate conditions can be established.

ITS LCO 3.2.1, Required Actions A.1 and A.2, maintain the requirements for a proportional reduction in reactor power and trip setpoints if limits for $F_Q(Z)$ are not met. However, ITS 3.2.1, Required Action A.3, specifies that reactor power and high flux trip setpoint reductions may be restored after satisfactory performance of ITS SR 3.2.1.1 which verifies that FQ(Z) has been restored to within its limit before increasing thermal power above the limit imposed by Required Action A.1.

This is an administrative change with no significant adverse impact on safety because requiring the performance of ITS SR 3.2.1.1 prior to increasing power is consistent with a reasonable interpretation of the existing requirements.

A.8 CTS 3.10.2.2.2 specifies Actions if $F_Q(Z)$ limits are not met; however, no Actions are specified if these Actions are not completed. Under the same conditions (Required Actions A.1 through A.3 are not performed within the specified completion time), ITS 3.2.1, Required Action B.1, establishes an explicit requirement that the reactor be in Mode 2 (outside the LCO Applicability, see ITS 3.2.1, DOC A.3) within 6 hours. This is an administrative change with no significant adverse impact on safety because it is a reasonable interpretation of the existing requirements (See ITS 3.2.1, DOC A.9).

A.9 CTS 3.10.2.2.2 Actions if $F_Q(Z)$ limits are not met includes the requirement that the reactor must be brought to hot shutdown with return to power authorized only for physics testing if subsequent incore

mapping cannot demonstrate that the hot channel factors are met within a 24-hour period.

Under the same conditions (ITS 3.2.1, Required Action A.1, not completed, i.e., power not reduced sufficiently to restore $F_Q(Z)$ to within limits), ITS 3.2.1, Required Action B.1, requires that the reactor be in Mode 2 (outside the LCO Applicability, see ITS 3.2.1, DOC A.3) within 6 hours (versus a reactor shutdown required by CTS). This is an administrative change because both CTS and ITS allow continued reactor operation for physics testing when $F_Q(Z)$ limits are not met except that ITS 3.2.1, Required Action B.1, explicitly limits this physics testing to Mode 2 (i.e., < 5% RTP). Additionally, ITS LCO 3.2.1 eliminates the requirement to complete a reactor shutdown before the initiation of physics testing. This is an administrative change with no significant adverse impact on safety because ITS 3.2.1, Required Action B.1 is consistent with the intent of the CTS.

A.10 CTS 3.10.11 specifies that any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10, Control Rod and Power Distribution Limits, shall be reported to the Nuclear Regulatory Commission within 30 days. ITS LCO 3.2.1 does not include an explicit requirement for the submittal of a special report for any event requiring plant shutdown on trip setpoint reduction. This change is needed because requirements for reportable events are included in 10 CFR 50.72 and 10 CFR 50.73 and are not repeated in the ITS to avoid the potential for contradictions. This change is acceptable because there is no change to the existing requirements and future changes are appropriately controlled. Additionally, adequate administrative controls exist to ensure this requirement is understood and properly implemented. Therefore, this is an administrative change with no adverse impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.10.2.2 specifies that $F_Q(Z)$ must be confirmed to be within required limits following refueling and every full power month

thereafter. ITS SR 3.2.1.1 maintains the requirement to confirm $F_Q(Z)$ is within required limits following refueling and every 31 effective full power days thereafter. However, ITS SR 3.2.1.1 requires that the post refueling verification is completed "prior to exceeding 75% rated thermal power" on the first startup following refueling. Additionally, ITS SR 3.2.1.1 requires that $F_Q(Z)$ is verified within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the Thermal Power at which $F_Q(Z)$ was last verified.

The first change, verify $F_Q(Z)$ prior to exceeding 75% RTP after refueling, is needed because peaking factors generally decrease as power level is increased. Therefore, performing SR 3.2.1.1 in Mode 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved. The second change, verify $F_Q(Z)$ within 12 hours after achieving equilibrium conditions whenever power is increased ≥ 10% RTP since the last determination of $F_Q(Z)$, is needed because it ensures that $F_Q(Z)$ values are being reduced sufficiently with the power increase to stay within the LCO limits. These SR Frequencies are modified by a Note that permits Thermal Power to be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained.

These more restrictive changes are acceptable because they do not introduce any operation that is un-analyzed while requiring that verifications of thermal limits be completed early enough to ensure that $F_Q(Z)$ is within required limits before reaching rated thermal power. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.10.2.2 specifies that if FQ(Z) limits are not met, then reactor power must be reduced so as not to exceed a fraction of rated power equal to the ratio of the $F_Q(Z)$ limit to measured value; however, no completion time is specified.

ITS LCO 3.2.1, Required Action A.1, maintains the requirement for a proportional reduction in reactor power if FQ(Z) limits are not met; however, a Completion Time of 15 minutes is specified. This change is needed because it eliminates ambiguity regarding the need for a prompt reduction in reactor power when limits for $F_Q(Z)$ are not met. This

change is acceptable because the Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.2.2.2 requires a proportional reduction of the high neutron flux trip setpoint whenever a hot channel factor exceeds its specified limit; however, no completion time is specified for either action. ITS LCO 3.2.1, Required Action A.2 (as modified by Generic Change TSTF-95), maintains the requirement for a proportional reduction of the high neutron flux trip setpoint whenever a hot channel factor exceeds its specified limit; however, a Completion Time of 72 hours is specified. This change extends the completion time from the several hours that would be needed under CTS to perform the required adjustment of setpoints to the 72 hours allowed by ITS. The 72 hour Completion Time is sufficient because the reduction in power required within 15 minutes by Required Action A.1 ensures requirements are met for steady state operation and there is a low probability of a severe transient during 72 hour period for setpoint adjustment. Therefore, this change does not have a significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.10.2.1 requires that $F_Q(Z)$ be maintained within the limits specified in the COLR and supports this requirement with the following information: mathematical formula for calculating the power distribution limits $F_Q(Z)$; tolerances for manufacturing and measurement errors; and, a statement that power distribution maps are made using the moveable detector system.

ITS LCO 3.2.1 maintains the requirement that $F_Q(Z)$ be maintained within the limits specified in the COLR; however, the supporting information is relocated to the COLR.

DISCUSSION OF CHANGES
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

This change allows the specific limits for $F_Q(Z)$ and associated supporting information to be removed from the ITS and relocated to the Core Operating Limits Report (COLR). This change is needed because the specific value for $F_Q(Z)$ is a cycle-specific variable.

This change is acceptable because ITS LCO 3.2.1 maintains the requirement to meet $F_Q(Z)$ limits and ITS 5.6.5, Core Operating Limits Report (COLR), includes detailed requirements that ensure core operating limits will be properly established and maintained. Requirements established by ITS 5.6.5 include the following:

a. The analytical methods used to determine the core operating limits must be those previously reviewed and approved by the NRC. The approved documents that document this approved methodology must be listed in ITS 5.6.5 and can be changed only with a TS change.

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

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications. Additionally, an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
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Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   This change extends the completion time for adjusting power range neutron flux-high trip setpoint when $F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is exceeded to 72 hours instead of the implied CTS requirement to perform the adjustment of setpoints as soon as possible. This change will not result in a significant increase in the probability of an accident previously evaluated because $F_Q(Z)$ is an operating restriction that is an initial condition of a design basis accident or transient analysis and is not assumed as the initiator of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because a reduction in reactor power is still required within 15 minutes and this action ensures there are sufficient margins to thermal limits.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because the proportional reduction in thermal power and the power range neutron flux-high trip setpoint compensates for the increase in the value for the total peaking factor assumed as an initial condition in the accident analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The 72 hour completion time for setpoint reduction does not involve a significant reduction in a margin of safety because reactor power has already been reduced within a 4 hour completion time which increases margins to thermal limits and satisfies assumptions in the safety analysis. The 72 hour completion time also recognizes that setpoint reduction is a sensitive operation that may inadvertently trip the Reactor Protection System which warrants additional time to perform the evolution.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.2.1

This ITS Specification is based on NUREG-1431 Specification No. **3.2.1B**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-022	**095 R0**	REVISE COMPLETION TIME FOR REDUCING POWER RANGE HIGH TRIP SETPOINT FROM 8 HOURS TO 72 HOURS	Approved by NRC	Incorporated	**T.1**
WOG-025	**097 R0**	REVISE NOTE TO SR 3.2.1.2, FQ MEASUREMENT	Approved by NRC	SR 3.2.1.2 not incorporated.	**N/A**
WOG-026	**098 R2**	RELOCATE THE FQ(Z) PENALTY FACTOR TO THE COLR	Approved by NRC	SR 3.2.1.2 not incorporated.	**N/A**
WOG-027	**099 R0**	EXTEND THE COMPLETION TIME FOR FQ(W) NOT WITHIN LIMITS FROM 2 HOURS TO 4 HOURS	Approved by NRC	Incorporated	**T.3**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**
WOG-101		CLARIFY COMPLETION TIME AND FREQUENCY WORDING	TSTF Review	Not Incorporated	**N/A**

3.2 POWER DISTRIBUTION LIMITS

3.2.1~~β~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

<3.10.2.1>
<DOC LA.1>

LCO 3.2.1~~β~~ $F_Q(Z)$~~, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$~~ shall be within the limits specified in the COLR. DB.1

<3.10.2.1>
<DOC A.3>
<DOC A.4>
<DOC A.5>

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.10.2.2.2> <DOC M.2> A. $F_Q^{\otimes}(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^{\otimes}(Z)$ exceeds limit.	15 minutes
	AND	
<3.10.2.2.2> <DOC L.1>	A.2 Reduce Power Range Neutron Flux—High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	~~8~~ 72 hours T.1
	AND	
	~~A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.~~	~~72 hours~~ CLB.1 DB.1
	AND	
<DOC A.7>	A.~~4~~ 3 Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. $F_Q^U(Z)$ not within limits.~~	~~B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^U(Z)$ exceeds limit.~~	~~2 hours~~ ~~4~~ ~~T.3~~ DB.1
<DOC A.8> <DOC A.9> ~~C.~~ B. Required Action and associated Completion Time not met.	~~C.1~~ B. Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

<DOC M.1>

----------NOTE----------
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
<3.10.2.2> <DOC M.1> <DOC A.7> SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within {12} hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 ----------NOTE---------- If $F_Q^W(Z)$ is within limits and measurements indicate $\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by a factor of [1.02] and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate $\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has not increased. ----------	
Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND (continued)

CLB.1
DB.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within [12] hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

CLB.1

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1~~5~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) ~~(F_Q Methodology)~~

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.~~6~~ 7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations~~, such as load following~~.

~~To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.~~ DB.1

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed ~~280 cal/gm~~ (Ref. 2); and DB2

Insert: B3.2-2-01

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36. PA.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

INSERT: B 3.2-2-01

DB.2

225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The current IP3 specific

~~For this facility, the actual~~ values of CFQ and K(Z) are given in the COLR; ~~however, CFQ is normally a number on the order of [2.32], and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1.~~ PA.1

~~For Relaxed Axial Offset Control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^W(Z)$ must meet the preceding limits on $F_Q(Z)$.~~ DB.1

An $F_Q^W(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q^W(Z) = F_Q^M(Z) \; [1.0815]$$

where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

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~~$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.~~ DB.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

INSERT: B 3.2-13-01

DB.1

This correction factor for the measured value of total peaking factor $F_Q^M(Z)$ is for the three percent needed to account for manufacturing tolerances and this value is further increased by five percent to account for measurement error.

BASES

LCO (continued)

~~The expression for $F_Q^W(Z)$ is:~~

~~$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$~~

~~where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.~~

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. exceeding

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q(Z)$ exceeds its limit, maintains an acceptable absolute power density. ~~$F_Q^C(Z)$ is $F_Q^W(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^W(Z)$ is the measured value of $F_Q(Z)$.~~ The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux—High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. 72 T.1

A.3

~~Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.~~ CLB.1

A.~~4~~ 3

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

~~If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.~~ CLB.1

(continued)

BASES

ACTIONS (continued)

B ~~E.1~~

If Required Actions A.1 through A.~~4~~ 3 ~~or B.1~~ are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 ~~and SR 3.2.1.2 are~~ is modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ ~~and $F_Q^W(Z)$ are~~ is within ~~their~~ specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which ~~they were~~ it was last verified to be within specified limits. Because $F_Q^C(Z)$ ~~and $F_Q^W(Z)$~~ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^C(Z)$ ~~and $F_Q^W(Z)$ are~~ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^C(Z)$ ~~and $F_Q^W(Z)$~~ following a power increase of more than 10%, ensures that ~~they are~~ it was verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ ~~and $F_Q^W(Z)$~~. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.1

Verification that $F_Q^W(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^W(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^W(Z) = F_Q^M(Z)$ [1.0815] (Ref. 4). $F_Q^W(Z)$ is then compared to its specified limits.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Insert: B 3.7-17-01

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^W(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

the highest

Core average

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^W(Z)$, another evaluation of this factor is required [12] hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^W(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

~~SR 3.2.1.2~~

~~The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the~~

(continued)

INSERT: B 3.2-17-01

(i.e., the ratio of local power density to the core average power density)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F_Q^W(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right],$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by a factor of [1.02], or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, [12] hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

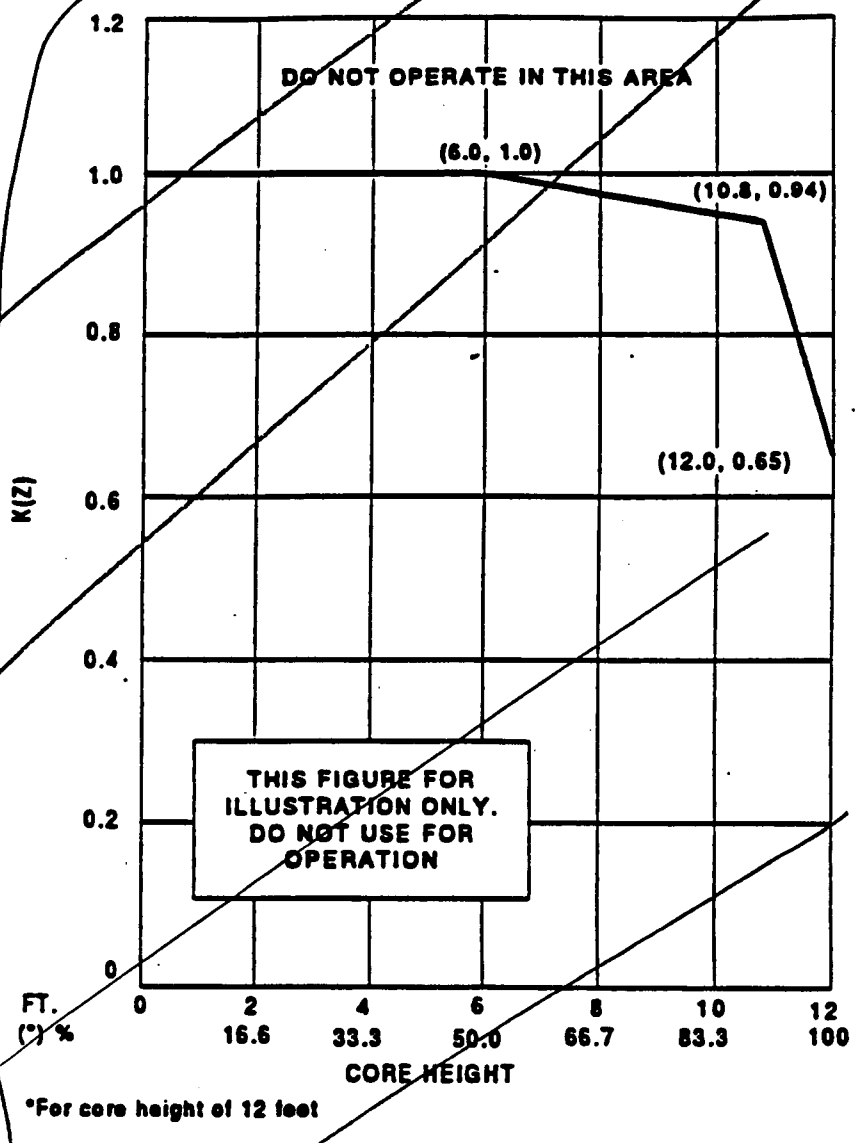
The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974. FSAR 14.2.6
2. Regulatory Guide 1.77, Rev. 0, May 1974.
3. 10 CFR 50, Appendix A, GDC 26.
4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," ~~June 1988~~.

X.1



*For core height of 12 feet

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG-1431, Rev 1, Section 3.2.1, was modified as needed to reflect that the IP3 Heat Flux Hot Channel Factor ($F_Q(Z)$) limits assume the use of the constant axial offset method for axial flux difference limits. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DB.2 IP3 Updated FSAR 14.2.6 establishes the requirement that a rod ejection will maintain average fuel pellet enthalpy at the hot spot below 225 cal/gm for non-irradiated fuel and 200 cal/gm for irradiated fuel. This limit is based on a review of experimental data and is intended to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves following a rod ejection.

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-95 (WOG-22), Rev.0, which revises the completion time for reducing power range high trip setpoint from 8 to 72 hours. This change was made because a completion time of 72 hours will allow time to perform a second flux map to confirm the results, or determine that the condition was temporary, without implementing an unnecessary trip setpoint change, during which there is increased potential for a plant transient and human error. Additionally, following a significant power reduction, at least 24 hours are required to re-establish steady state xenon prior to taking a flux map, and approximately 8 to 12 hours to obtain a flux map and analyze the data. Finally, the setpoint adjustment is estimated to take approximately 4 hours per channel (review of plant condition supportive of removing channels from service, tripping of bistables, setpoint adjustments, and channel restoration), adding 2 hours for necessary initial preparations (procedure preparation, calibration equipment checks, obtaining tools and approvals), it is reasonable to expect a total of 18 hours. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates numbering changes needed to support Generic Change TSTF-136 (WOG-59), Rev.0, which combined LCO 3.1.1 and LCO 3.1.2, This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.3 This change incorporates Generic Change TSTF-99 (WOG-27), Rev.0, which extends the Completion Time for Fq(w) not within limits from 2 hours to 4 hours. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

X.1 NUREG-1431, Rev 1, Section 3.2.1, was modified to delete Figure B 3.2.1B-1 because IP3 maintains the Heat Flux Hot Channel Factor ($F_Q(Z)$) limits in the Core Operating Limits Report.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.2:
"Nuclear Enthalpy Rise Hot Channel Factor"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55% RTP.	72 hours
	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ----------	
	Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to 1.3 using the W3 CHF correlation. All DNB limited transient events are assumed to

BASES

BACKGROUND (Continued)

begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.3 using the W3 CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of 10 CFR 50.36.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least additional

BASES

LCO (continued)

heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase a small amount for every 1% RTP reduction in THERMAL POWER as specified in the COLR.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore

BASES

ACTIONS

A.1.1 (continued)

$F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux-High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and

BASES

ACTIONS

A.1.2.1 and A.1.2.2 (continued)

there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on

BASES

ACTIONS

B.1 (continued)

operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. FSAR 14.2.6.

2. 10 CFR 50, Appendix A.

3. 10 CFR 50.46.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.2:
"Nuclear Enthalpy Rise Hot Channel Factor"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-1	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-2	112	112	IPN 96-063	Leakage Limits for RCS and SIS	
3.10-8	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-9	175	175	No TSCRs	No TSCRs for this Page	N/A
3.10-10	180	180			

A.1

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

A.2

Applicability:

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

Objectives:

To ensure:

1. Core subcriticality after reactor trip.

2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

SEE ITS 3.1.1	3.10.1	Shutdown Reactivity
	3.10.1.1	Whenever T_{avg} > 200°F the shutdown margin shall be ≥ 1.3% Δk/k.
	3.10.1.2	When the conditions of specification 3.10.1.1 are not met, initiate boration to restore shutdown margin within limit.
	3.10.2	Power Distribution Limits
LCO 3.2.2 Applicability	3.10.2.1	Mode 1 A.3 A.4 ~~At all times~~, ~~except during low power physics tests~~, the hot channel factors defined in the basis must meet the following limits:
SEE ITS 3.2.1		$F_Q(Z) \leq (F_Q^{RTP}/p) \times K(Z)$ for $P > 0.5$
		$F_Q(Z) \leq (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$
LCO 3.2.2		Shall be within limits in COLR $F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP}(1 + PF_{\Delta H}(1-P))$ LA.1

Where P is the fraction of full power at which the core is operating, K(Z) is the fraction specified in the

A.1

Add Req Action A.1.1
Add Note to Condition A — M.4

LA.1

COLR, Z is the core height location of F_Q. F_Q^{RTP} is the F_Q limit at Rated Thermal Power (RTP) specified in the COLR, $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

A.5

SR 3.2.2.1 ~~3.10.2.2~~ Following initial core loading, subsequent reloading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

prior to >75% RTP — M.1

LA.1

SEE ITS 3.2.1

3.10.2.2.1 The measurement of total peaking factor F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 When $F_{\Delta H}^{N}$ is measured, no additional allowances are necessary prior to comparison with the limits of section 3.10.2. An error allowance of 4% has been included in the limits of section 3.10.2. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q or $F_{\Delta H}^{N}$ limit to measured value, whichever is less. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

LCO 3.2.2, Cond A
Req Act A.1.2.1
Req Act A.1.2.2
Req Act A.2

LA.1

within 4hr — L.1

within 72hr — M.2 — A.6

Mode 2 in 6 hours — L.2

SEE ITS 3.2.3

~~3.10.2.3~~ The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux differences must be updated each effective full power month by linear interpolation using the most recent measured value and a value of 0 percent at the end of the cycle life.

~~3.10.2.4~~ Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference of all but one operable excore channel shall be maintained within the band specified in the COLR about the target flux difference.

Add Required Action A.3 — M.3

SEE ITS 3.1.4

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per 8 hours and after a load change greater than 10 percent of rated power.

SEE ITS 3.1.2

3.10.10 Reactivity Balance

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3.10.11 Notification

Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

A.7

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR in normal operation or in short term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant

A.1

A.1

F_Q^E Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

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

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

An upper bound envelope of F_Q^{RTP} specified in the COLR times the normalized peaking factor axial dependence of K(Z) specified in the COLR has been determined consistent with Appendix K criteria and is satisfied for OFA transition mixed cores [3] by all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analysis based on this upper bound normalized envelope, K(Z), specified in the COLR demonstrates that the peak clad temperature is below the peak clad temperature limit of 2200°F.[2]

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

In the specified limit of $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq F_{\Delta H}^{RTP}/1.04$, where $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^N$ limit at Rated Thermal Power specified in the COLR. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

(e.g. rod misalignment) affect $F_{\Delta H}^{N}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^{N}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}^{N}$ is less readily available. When a measurement of $F_{\Delta H}^{N}$ is taken, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

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

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

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the group step counter demand position (operating at greater than 85% of rated thermal power with no accounting for peaking factor margin), or 18.75 inches (operating at less than or equal to 85% of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment greater than 15 inches with consideration of instrumentation error and 18 steps indicated misalignment corresponds to 18.75 inches with instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

A.1

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.2:
"Nuclear Enthalpy Rise Hot Channel Factor"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety because neither are required by 10 CFR 50.36, and neither define nor impose any specific requirements.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.2.1 specifies that the power distribution limit $F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is applicable "at all times." ITS LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), is applicable in Mode 1 (i.e., when thermal power is greater than 5% RTP). This is an administrative change because CTS 3.10.2.2 allows unlimited operation

for physics testing when $F^N_{\Delta H}$ limits are not met and, consistent with ITS 3.1.8, physics testing is performed in Mode 2 only. Therefore, the implied CTS applicability for $F^N_{\Delta H}$ is Mode 1. This change is acceptable because $F^N_{\Delta H}$ is a radial peaking factor limit and there are very substantial margins for $F^N_{\Delta H}$ limits when in Mode 2 assuming ITS LCO 3.1.4, Rod Group Alignment Limits, is met and thermal power is less than 5% RTP. This is an administrative change with no adverse impact on safety because the power distribution limit specified in CTS 3.10.2.1 and ITS LCO 3.2.2 cannot be exceeded except when in Mode 1 if other Applicable LCOs are being met.

A.4 CTS 3.10.2.1 specifies that the power distribution limit $F^N_{\Delta H}$ is not required to be met "during low power physics tests." ITS LCO 3.2.2 does not state this exception because the limits on power distribution are applicable only in Mode 1 (i.e., when thermal power is greater than 5% RTP) and, as specified in ITS LCO 3.1.8, Physics Tests Exceptions, physics tests may be performed in Mode 2 only. The applicability requirements for ITS LCO 3.2.2 (Mode 1) and ITS LCO 3.1.8 (Mode 2) eliminates the need for an exemption from ITS LCO 3.2.2 for physics testing. Therefore, this is an administrative change with no adverse impact on safety.

A.5 CTS 3.10.2.2 includes "following initial core loading" as one of the required SR Frequencies for verifying $F^N_{\Delta H}$ limits are met. ITS SR 3.2.2.1 maintains the requirement for periodic verification of $F^N_{\Delta H}$; however, ITS SR 3.2.2.1 does not specify following initial core loading. Deletion of the requirement to perform these tests "following initial core loading" is an administrative change with no impact on safety because initial fuel loading was a one time event that has been completed.

A.6 CTS 3.10.2.2.2 specifies Actions if limits for either $F_Q(Z)$ or $F^N_{\Delta H}$ are exceeded. ITS LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), establishes the Required Actions if limits for $F^N_{\Delta H}$ are exceeded. ITS LCO 3.2.1, Heat Flux Hot Channel Factor ($F_Q(Z)$), establishes the

Required Actions if limits for $F_Q(Z)$ are exceeded. This is an administrative change with no significant adverse impact on safety because there are no changes to the existing requirements except as identified and justified in ITS 3.2.1 and ITS 3.2.2.

A.7 CTS 3.10.11 specifies that any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10, Control Rod and Power Distribution Limits, shall be reported to the Nuclear Regulatory Commission within 30 days. ITS LCO 3.2.2 does not include an explicit requirement for the submittal of a special report for any event requiring plant shutdown on trip setpoint reduction. This change is needed because requirements for reportable events are included in 10 CFR 50.72 and 10 CFR 50.73 and are not repeated in the ITS to avoid the potential for contradictions. This change is acceptable because there is no change to the existing requirements and future changes are appropriately controlled. Additionally, adequate administrative controls exist to ensure this requirement is understood and properly implemented. Therefore, this is an administrative change with no adverse impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.10.2.2 specifies that $F^N_{\Delta H}$ must be confirmed to be within required limits following refueling. ITS SR 3.2.2.1 maintains the same requirement except that confirmation must be completed "prior to exceeding 75% rated thermal power" on the first startup following refueling. This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while requiring that verification of thermal limits be completed early enough to ensure that $F^N_{\Delta H}$ is within required limits before reaching rated thermal power. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.10.2.2.2 requires a proportional reduction of the reactor power and the high neutron flux trip setpoint if a hot channel factor (i.e., $F^N_{\Delta H}$) exceeds its specified limit.

Under the same conditions, ITS 3.2.2, Required Action A.1, requires reducing thermal power to < 50% RTP (versus the proportional reduction required by CTS) and reducing the Power Range Neutron Flux–High trip setpoints to ≤ 55% RTP (versus the proportional reduction required by CTS). This change is needed because reducing thermal power to < 50% RTP increases the DNB margin and will prevent the DNBR limit being violated during steady state operation. The reduction in trip setpoint to < 55% RTP ensures that continuing operation remains at the acceptable low power level. This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while providing greater assurance that DNBR limits will not be violated during steady state or transient operation when $F^N_{\Delta H}$ are not met. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.10.2.2.2 requires a proportional reduction in reactor power and trip setpoints if limits for $F^N_{\Delta H}$ are not met and requires subsequent verification that these actions were effective. However, there are no explicit requirements for returning to full power if appropriate conditions can be established.

ITS LCO 3.2.2, Required Actions A.1 and A.2, maintain the requirements for a reduction in reactor power and trip setpoints (See ITS 3.2.1, DOC M.2) if limits for $F^N_{\Delta H}$ are not met and require subsequent verification that these actions were effective. However, ITS LCO 3.2.2, Required Action A.3, specifies that reactor power and trip setpoint reductions may be restored only after a determination that $F^N_{\Delta H}$ is within limits prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP.

This change is needed because it provides greater assurance that DNBR limits will not be violated following restoration to full power after $F^N_{\Delta H}$ limits were not met. This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed. Therefore, this change has no adverse impact on safety.

M.4 CTS 3.10.2.2.2 requires a proportional reduction in reactor power and

trip setpoints if limits for $F^N_{\Delta H}$ specified in COLR are not met and requires subsequent verification that these actions were effective; however, there is no explicit option for restoration of the LCO requirement.

ITS LCO 3.2.2, Required Actions A.1.2.1, A.1.2.2 and A.2, maintain the requirements for a reduction in reactor power and trip setpoints (See ITS 3.2.1, DOC M.2) if limits for $F^N_{\Delta H}$ are not met and requires subsequent verification within 4 hours that these actions were effective. This change is needed because ITS Condition A includes a Note that requires performance of ITS SR 3.2.2.1 (Required Actions A.2 and A.3) even if restoration of the LCO requirements is completed within the specified Completion Time without the need for a power reduction. CTS 3.10.2.2 includes the same requirement to re-verify thermal limits are met within 24 hours with an implicit requirement this action is needed whether or not a power reduction was required. Under the ITS format, an explicit statement of the option to restore compliance with the LCO is needed to establish the requirement to re-verify thermal limits are met within 24 hours whether or not a power reduction was required. This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.2.2.2 requires a proportional reduction in reactor power and trip setpoints if limits for $F^N_{\Delta H}$ are not met. However, no completion time is specified.

ITS LCO 3.2.2, Required Actions A.1.2.1 and A.1.2.2, maintain the requirements for a reduction in reactor power and trip setpoints (See ITS 3.2.1, DOC M.2) if limits for $F^N_{\Delta H}$ are not met; however, Required Action A.1.2.1 specifies a Completion Time of 4 hours for power reduction and Required Action A.1.2.2 specifies a Completion Time of 72 hours for trip setpoint reduction. This is a less restrictive change because CTS 3.10.2.2.2 implies these Actions must be initiated immediately.

This 4 hour Completion Time to reduce power is needed because it provides sufficient time to attempt restoration (in accordance with Required Action A.1.1) from conditions such as a misaligned rod and provides an acceptable time to reach the required power level from full power operation if restoration is not successful. This change is acceptable because the DNBR limit is not likely to be violated in steady state operation because factors, such as static control rod misalignment, are considered in the safety analyses and the probability of a transient during this 4 hour period is low.

This 72 hour Completion Time to reduce trip setpoints is needed because there is no urgent need to reduce the trip setpoints because power reduction satisfies safety analysis considerations and setpoint reduction is a sensitive operation that may inadvertently trip the Reactor Protection System. This change is acceptable because power reduction satisfies safety analysis considerations and setpoint reduction is needed only to ensure that continuing operation remains at an acceptable low power level with adequate DNBR margin. Administrative controls are sufficient to maintain power level until setpoint reductions are completed. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.10.2.2.2 requires a reduction in reactor power and trip setpoints if limits for $F^N_{\Delta H}$ are not met and requires subsequent verification that these actions were effective. If subsequent verification demonstrates that the reduction in reactor power does not result in $F^N_{\Delta H}$ limits being met, then CTS 3.10.2.2.2 requires that the reactor be brought to a hot shutdown condition with return to power authorized only for physics testing.

ITS LCO 3.2.2, Required Actions A.1.2.1, A.1.2.2 and A.2, maintain the requirements for a reduction in reactor power and trip setpoints (See ITS 3.2.1, DOC M.2) if limits for $F^N_{\Delta H}$ are not met and requires subsequent verification that these actions were effective. However, if subsequent verification demonstrates that the reduction in reactor power does not result in $F^N_{\Delta H}$ limits being met, then ITS 3.2.2, Required Action B.1, requires that the reactor be in Mode 2 (outside the LCO

Applicability, see ITS 3.2.1, DOC A.3) within 6 hours (versus a reactor shutdown required by CTS). This is a less restrictive change because CTS requires a reactor shutdown if $F^N_{\Delta H}$ is not restored to within limits by a reduction to 50% RTP and ITS only requires that the plant be placed outside the LCO Applicability.

This change is needed because the LCO is no longer required to be met when in Mode 2 (i.e., outside the LCO Applicability, see ITS 3.2.1, DOC A.3). This change is acceptable because $F^N_{\Delta H}$ is a radial peaking factor limit and there are very substantial margins for $F^N_{\Delta H}$ limits when in Mode 2 assuming ITS LCO 3.1.4, Rod Group Alignment Limits, is met and thermal power is less than 5% RTP. This change has no adverse impact on safety because the power distribution limit specified in CTS 3.10.2.1 and ITS LCO 3.2.2 cannot be exceeded except when in Mode 1 if other Applicable LCOs are being met.

REMOVED DETAIL

LA.1 CTS 3.10.2.1 requires that $F^N_{\Delta H}$ be maintained within the limits specified in the COLR and supports this requirement with the following information: mathematical formula for calculating the power distribution limits $F^N_{\Delta H}$; tolerances for manufacturing and measurement errors; and, a statement that power distribution maps are made using the moveable detector system.

ITS LCO 3.2.1 maintains the requirement that $F^N_{\Delta H}$ be maintained within the limits specified in the COLR; however, the supporting information is relocated to the COLR.

This change allows the specific limits for $F^N_{\Delta H}$ to be removed from the ITS and relocated to the Core Operating Limits Report (COLR). This change is needed because the specific value for $F^N_{\Delta H}$ is a cycle-specific variable. Therefore, by maintaining the $F^N_{\Delta H}$ limits in the COLR, the core reload design can be completed after shutdown when the actual end of cycle burnup is known. This saves redesign efforts that occur if actual burnup differs from the projected value.

DISCUSSION OF CHANGES

ITS SECTION 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

This change is acceptable because ITS LCO 3.2.2 maintains the requirement to meet $F^N_{\Delta H}$ limits and ITS 5.6.5, Core Operating Limits Report (COLR), includes detailed requirements that ensure core operating limits will be properly established and maintained. Requirements established by ITS 5.6.5 include the following:

a. The analytical methods used to determine the core operating limits must be those previously reviewed and approved by the NRC. The approved documents that document this approved methodology must be listed in ITS 5.6.5 and can be changed only with a TS change.

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

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications. Additionally, an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.2:
"Nuclear Enthalpy Rise Hot Channel Factor"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   This change increases the completion time for reducing the high neutron flux trip setpoint whenever Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) exceeds its specified limit to 72 hours instead of the implied CTS requirement to perform the adjustment of setpoints as soon as possible. This change will not result in a significant increase in the probability of an accident previously evaluated because $F^N_{\Delta H}$ is an operating restriction that is an initial condition of a design basis accident or transient analysis and is not assumed as the initiator of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because once power is reduced to < 50% within the 4 hour completion time of Required Action A.1.2.1, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. Setpoint reduction only ensures that the low power operation with the associated higher thermal margins is maintained.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions once power is reduced to < 50% within the 4 hour completion time of Required Action A.1.2.1, the safety analysis assumptions are satisfied. Therefore, these changes will not create the possibility of a new or

different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because once power is reduced to < 50% within the 4 hour completion time of Required Action A.1.2.1, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. Setpoint reduction only ensures that the low power operation with the associated higher thermal margins is maintained. The 72 hour completion time also recognizes that setpoint reduction is a sensitive operation that may inadvertently trip the Reactor Protection System. Therefore, this change does not have a significant adverse impact on safety.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

This change eliminates the requirement to shutdown the reactor if $F^N_{\Delta H}$ is not restored to within limits by a reduction to 50% RTP and requires that the plant be placed outside the LCO Applicability (i.e., placed in Mode 2). This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because there is no effect on the initiator of any analyzed event. Additionally, $F^N_{\Delta H}$ is a radial peaking factor limit and there are very substantial margins for $F^N_{\Delta H}$ limits when in Mode 2 assuming ITS LCO 3.1.4, Rod Group Alignment Limits, is met and thermal power is less than

5% RTP. This change has no adverse impact on safety because the power distribution limit specified in CTS 3.10.2.1 and ITS LCO 3.2.2 cannot be exceeded except when in Mode 1 if other Applicable LCOs are being met.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions once power is reduced to < 5% (i.e., Mode 2), the safety analysis assumptions are satisfied. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because $F^N_{\Delta H}$ is a radial peaking factor limit and there are very substantial margins for $F^N_{\Delta H}$ limits when in Mode 2 assuming ITS LCO 3.1.4, Rod Group Alignment Limits, is met and thermal power is less than 5% RTP. This change has no adverse impact on safety because the power distribution limit specified in CTS 3.10.2.1 and ITS LCO 3.2.2 cannot be exceeded except when in Mode 1 if other Applicable LCOs are being met.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.2:
"Nuclear Enthalpy Rise Hot Channel Factor"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.2.2

This ITS Specification is based on NUREG-1431 Specification No. **3.2.2**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-022	**095 R0**	REVISE COMPLETION TIME FOR REDUCING POWER RANGE HIGH TRIP SETPOINT FROM 8 HOURS TO 72 HOURS	Approved by NRC	Incorporated	**T.1**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.2**
WOG-094		ELIMINATE UNNECESSARY ACTIONS TO RESTORE COMPLIANCE WITH THE LCO	TSTF Review	Not Incorporated	**N/A**
WOG-101		CLARIFY COMPLETION TIME AND FREQUENCY WORDING	TSTF Review	Not Incorporated	**N/A**

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

<3.10.2.1> <DOC LA.1> LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

<3.10.2.1> <DOC A.3> <DOC A.4> APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.4> A. ---------NOTE--------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- <3.10.2.2.2> <DOC L.1> <DOC M.2> $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
<3.10.2.2.2> <DOC L.1> <DOC M.2>	A.1.2.2 Reduce Power Range Neutron Flux—High trip setpoints to ≤ 55% RTP.	8 hours 72 T.1
	AND	
<3.10.2.2.2>	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.3> A. (continued)	A.3 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ----------	
	Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
<3.10.2.2.2> <DOC L.2> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<3.10.2.2> <DOC M.D> SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to [1.3] using the [W3] CHF correlation. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm [Ref. 1]; and

Insert: B3.2-22-01

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of [1.3] using the [W3] CHF correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

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225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel

BASES

APPLICABLE SAFETY ANALYSES (continued)

this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.~~0~~6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least additional heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced

(continued)

BASES

LCO (continued)

thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER. (as specified in the COLR) (a small amount) (DB.1)

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding

(continued)

BASES

ACTIONS

A.1.1 (continued)

75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux—High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

THERMAL POWER

PA.1

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

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T.1

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which

(continued)

BASES

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is $\geq$ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. ~~Regulatory Guide 1.77, Rev. [0], May 1974~~
2. 10 CFR 50, Appendix A, ~~GDC 26~~
3. 10 CFR 50.46.

FSAR 14.2.6.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.2:
"Nuclear Enthalpy Rise Hot Channel Factor"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DB.2 IP3 Updated FSAR 14.2.6 establishes the requirement that a rod ejection will maintain average fuel pellet enthalpy at the hot spot below 225 cal/gm for non-irradiated fuel and 200 cal/gm for irradiated fuel. This limit is based on a review of experimental data and is intended to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves following a rod ejection.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-95 (WOG-22), which revised the completion time for reducing Power Range High trip setpoint from 8 to 72 hours. This change was made because a completion Time of 72 hours

will allow time to perform a second flux map to confirm the results, or determine that the condition was temporary, without implementing an unnecessary trip setpoint change, during which there is increased potential for a plant transient and human error. Additionally, following a significant power reduction, at least 24 hours are required to re-establish steady state xenon prior to taking a flux map, and approximately 8 to 12 hours to obtain a flux map, and analyze the data. Finally, the setpoint adjustment is estimated to take approximately 4 hours per channel (review of plant condition supportive of removing channels from service, tripping of bistables, setpoint adjustments, and channel restoration), adding 2 hours for necessary initial preparations (procedure preps, calibration equipment checks, obtaining tools and approvals), it is reasonable to expect a total of 18 hours.

T.2 This change incorporates Generic Change TSTF-136 (WOG-59), Rev.0, which combines ITS 3.1.1,Shutdown Margin (SDM) – T_{avg} > 200°F, and ITS 3.1.2, Shutdown Margin (SDM) – T_{avg} ≤ 200°F, into ITS 3.1.1, Shutdown Margin (SDM). This change is necessary because ITS 3.1.1 and ITS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin during physics tests to COLR.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.3:
"AXIAL FLUX DIFFERENCE (AFD)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

b. May deviate outside the target band with THERMAL POWER < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

---------- NOTES ----------

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

2. With Thermal Power ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

3. With Thermal Power < 50% RTP and > 15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP.	15 minutes
C. ---------NOTE--------- Required Action C.1 must be completed whenever Condition C is entered. ---------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within target band for each OPERABLE excore channel.	7 days
SR 3.2.3.2	----------NOTE---------- Assume logged values of AFD exist during the preceding time interval. ---------- Verify AFD is within target band and log AFD for each OPERABLE excore channel.	-----NOTE------ Only required to be performed if AFD monitor alarm is inoperable ---------- Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP AND Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.3	Update target flux difference of each OPERABLE excore channel by: a. Determining the target flux difference in accordance with SR 3.2.3.4, or b. Using linear interpolation between the most recently measured value, and either the predicted value for the end of cycle or 0% AFD.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter
SR 3.2.3.4	----------NOTE---------- The initial target flux difference after each refueling may be determined from design predictions. ---------- Determine, by measurement, the target flux difference of each OPERABLE excore channel.	Once within 31 EFPD after each refueling AND 92 EFPD thereafter

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, CAOC, involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 190 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

BASES

BACKGROUND (continued)

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology entails:

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36.

LCO

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 1). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as $\%\Delta$ flux or $\%\Delta I$.

The AFD LCO establishes the limits for how much and for how long the measured AFD may deviate from a predetermined (i.e., target) AFD. The amount that the measured AFD may deviate from the target AFD is called the "target band" which is specified in the COLR. If the measured AFD is within the "target band," then there are no restrictions on plant operations.

If the measured AFD cannot be consistently maintained within the "target band" but can be maintained within the "acceptable operation limits," then reactor power must be reduced to < 90% RTP. However, even with power reduced, the measured AFD must be maintained within the target band for 23 out of every 24 hours (i.e., the cumulative penalty deviation time cannot be exceeded); otherwise additional power reductions are required.

BASES

LCO (continued)

If the measured axial flux difference cannot be maintained within the "acceptable operation limits" or the cumulative penalty deviation time for operating outside the target band is exceeded, then reactor power must be reduced to < 50% RTP. There are no restrictions on measured AFD when reactor power is < 50% RTP; however, the measured AFD must be within the "target band" for a specified period of time (i.e., the cumulative penalty deviation time must be within a specified limit) before reactor power can be increased ≥ 50% RTP.

The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup.

With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

Target band and AFD acceptable operation limits are specified in the COLR.

The LCO is modified by four Notes. Note 1 states the conditions necessary for declaring the AFD outside of the target band. Notes 2 and 3 describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the

BASES

LCO (continued)

xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part b of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR. This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part b of this LCO (i.e., THERMAL POWER 50% RTP). The cumulative penalty time is the sum of penalty times from Parts b and c of this LCO.

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part c of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

For surveillance of the power range channels performed according to SR 3.3.1.6, Note 4 allows deviation outside the target band for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system.

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1).

Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

BASES

LCO (continued)

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent

BASES

ACTIONS

C.1 (continued)

of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer provides an alarm when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.3.2

With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP, the AFD is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. At power levels < 90% RTP, but > 15% RTP, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

SR 3.2.3.2 is modified by a Note that states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time. The AFD should be monitored more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3

This Surveillance requires that the target flux difference is updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.4. Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update.

SR 3.2.3.4

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.4 (continued)

A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore excore calibrations that may have occurred in the interim.

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

REFERENCES

1. FSAR, Chapter 7.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.3:
"AXIAL FLUX DIFFERENCE (AFD)"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
3.10-2	112	112	IPN 96-063	Leakage Limits for RCS and SIS	
3.10-3	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-4	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-11	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-12	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-13	103	103	No TSCRs	No TSCRs for this Page	N/A
3.11-1	122	122	No TSCRs	No TSCRs for this Page	N/A

COLR, Z is the core height location of F_Q, F_Q^{RTP} is the F_Q limit at Rated Thermal Power (RTP) specified in the COLR, $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

SEE ITS 3.2.1 and ITS 3.2.2

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

3.10.2.2.1 The measurement of total peaking factor F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 When $F_{\Delta H}^N$ is measured, no additional allowances are necessary prior to comparison with the limits of section 3.10.2. An error allowance of 4% has been included in the limits of section 3.10.2. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated power equal to the ratio of the F_Q or $F_{\Delta H}^N$ limit to measured value, whichever is less. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

SR 3.2.3.4 ~~3.10.2.3~~ The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per 31 EFPD after refuel equivalent full power quarter. L.3 The target flux differences must be updated each effective full power month by linear interpolation using the most recent measured value and a value of 0 percent at the end of the cycle life. A.10

SR 3.2.3.3

~~3.10.2.4~~ Note 4 ~~Except during physics tests~~ during excore calibration procedures ~~and except as modified by Items 3.10.2.5 through 3.10.2.7 below~~ the indicated axial flux difference ~~of all but one operable excore channel~~ shall be maintained within the band specified in the COLR about the target flux difference. M.1 total of 16 hours A.5

LCO 3.2.3.a

LCO 3.2.3, Note 1

Add LCO 3.2.3 applicability L.2

A.1

LCO 3.2.3, Note 1

~~3.10.2.5~~

LCO 3.2.3 Cond A Cond B

~~3.10.2.5.1~~

At a power level greater than 90% of rated power,

If the indicated axial flux difference of more than one operable excore channel deviates from its target band, either such deviation shall be immediately eliminated or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

within 15 minutes — A.3

following 15 minutes — A.3

~~3.10.2.6~~

At a power level no greater than 90 percent of rated power, and ≥ 50% RTP

< — A.4

~~3.10.2.6.1~~

LCO 3.2.3.b and Note 2

The indicated axial flux difference (AFD) may deviate from its target band specified in the COLR for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by that specified in the COLR at 90% power and increasing by the value specified in the COLR for each 2 percent of rated power below 90% power. A two hour deviation is permissible during tests performed as part of the augmented startup program. (1)

LA.1

A.6

LCO 3.2.3, Note 1

~~3.10.2.6.2~~

LCO 3.2.3 Cond C, Req Act C.1

If Item 3.10.2.6.1 is violated by more than one operable excore channel, then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

A.7

within 30 minutes

L.1

Add Note to Cond C — A.8

~~3.10.2.6.3~~

LCO 3.0.4

A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference of all but one operable excore channel being within their target band.

A.9

~~3.10.2.7~~

LCO 3.2.3.c

At a power level no greater than 50 percent of rated power,

<

~~3.10.2~~.7.1

The indicated axial flux difference may deviate from its target band.

~~3.10.2.7.2~~

LCO 3.2.3.b Note 1

A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference of all but one operable excore channel not being outside their target bands for more than two hours (cumulative) out of the preceding 24-hour period. One-half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one-hour cumulative (two-hour cumulative during augmented startup tests) (1) maximum the flux difference may deviate from its target band of a power level ≤ 90% of rated power.

A.5

Note 3

A.6

~~3.10.2.8~~ SR 3.2.3.2 Note to SR Freq.

Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals ~~for the first 24 hours and~~ half-hourly thereafter. M.2

3.10.2.9 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 When ever the indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, within two hours the tilt condition shall be eliminated or the following actions shall be taken:

a) Restrict core power level and reset the power range high flux setpoint three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0,

and

SEE ITS 3.2.4

b) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be reset to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ration exceeds 1.09 and there is simultaneous indication of a misaligned control rod, restrict core power level 3% of rated value for every percent of indicated power tilt ratio exceeding 1.0 and realign the rod within two hours. If the rod is not realigned within two hours or if there is no simultaneous indication of a misaligned rod, the reactor shall be brought to the hot shutdown condition within 4 hours. If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt.

A.1

4. Axial Power Distribution Control Procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F_{\Delta H}^{N}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The referenced value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that F_Q upper bound envelope of F_Q^{RTP} times K(Z) (specified in the COLR) is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup

A.1

proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and the AFD deviation specified in the COLR is permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control ar not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band. However, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the AFD range specified in the COLR.

A.1

If, for any reason, flux difference is not controlled within the AFD limit specified in the COLR for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the control rods to produce the required indicated flux difference.

For FSAR Section 14.1 events, the core is protected from overpower and a minimum DNBR of the applicable safety limit DNBR by an automatic protection system. Compliance with operating procedures is assumed as a precondition for FSAR Section 14.1 events. However, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation, as this phenomenon is caused by some asymmetric perturbation, e.g., rod misalignment, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could build up. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be commensurate with design and safety limits for DNB protection and linear heat generation rate as

3.11 MOVABLE INCORE INSTRUMENTATION

A.2

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the incore instrumentation system, for the recalibration of the excore axial off-set detection system.

Specification

SEE RELOCATED CTS

A. A minimum of 2 thimbles per quadrant and sufficient movable incore detectors shall be operable during recalibration of the excore axial off-set detection system.

B. Power shall be limited to 90% of rated power if recalibration requirements for the excore axial off-set detection system, identified in Table 4.1-1, are not met.

A.10

SEE RELOCATED CTS

C. During the incore/excore calibration procedure, all full core flux maps will be made only when at least 38 of the movable detector guide thimbles are operable.

Basis

The Movable Incore Instrumentation System[1] has six drives, six detectors, and 58 movable detector guide thimbles in the core. Fifty (50) of these thimbles were provided as part of the original design basis of the plant. The other eight thimbles are supplemental thimbles that were connected during the 8/9 refueling outage. The eight supplemental thimbles are maintained to the same standards as the original 50 thimbles. These eight supplemental thimbles can be used to satisfy the 38 thimble requirement for flux mapping. An appropriate evaluation will be performed prior to the initial use of the supplemental thimbles to satisfy technical specification requirements for flux mapping. The eight supplemental thimbles improve the reliability of the Movable Incore Instrumentation System. Each of the six movable incore detectors can be routed to sixteen or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

A.1

To calibrate the excore detectors, it is only necessary that the Movable Incore Instrumentation System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.3:
"AXIAL FLUX DIFFERENCE (AFD)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.2.3 - AXIAL FLUX DIFFERENCE (AFD)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety because neither are required by 10 CFR 50.36, and neither define nor impose any specific requirements.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.2.5.1 specifies that if the indicated axial flux difference deviates from its target band, then either a) the deviation shall be eliminated immediately; or, b) reactor power shall be reduced to < 90% RTP.

ITS 3.2.3, Required Actions A.1 and B.1, maintains these requirements

except that Required Action A.1 provides an explicit limit of 15 minutes (versus initiate action immediately) for attempts to restore AFD to within limits and Required Action B.1 provides an explicit Completion Time of an additional 15 minutes to reduce power to < 90% RTP if restoration is not successful.

This change is needed because it eliminates ambiguity about acceptable amounts of time for completing these actions. This change is acceptable because xenon distributions change very little in the 15 minutes permitted to attempt restoration of AFD limits and, if these attempts fail, an additional 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

This is an administrative change with no significant adverse impact on safety because the ITS Completion Times of 15 minutes for Required Action A.1 and an additional 15 minutes for Required Action B.1 is a reasonable interpretation of the existing requirements.

A.4 CTS 3.10.2.5 specifies requirements for AFD when thermal power is > 90% RTP; CTS 3.10.2.6 specifies requirements for AFD when thermal power is < 90% RTP; and, CTS 3.10.2.7 specifies requirements for AFD when thermal power is < 50% RTP. ITS LCO 3.2.3, Parts a, b and c and associated Notes, are presented to eliminate potential ambiguity rules for accumulation of penalty time when < 90% and > 50% RTP and when < 50% RTP. This is an administrative change with no significant adverse impact on safety because the change eliminates a potential ambiguity and there is no change to the existing requirements.

A.5 CTS 3.10.2.4 specifies that the indicated axial flux difference of all but one operable excore channel shall be maintained within the band specified in the COLR except as modified by CTS 3.10.2.5 through CTS 3.10.2.7. Specifically, CTS 3.10.2.5 specifies requirements for AFD when thermal power is > 90% RTP; CTS 3.10.2.6 specifies requirements for AFD when thermal power is < 90% RTP; and, CTS 3.10.2.7 specifies requirements for AFD when thermal power is < 50% RTP. ITS LCO 3.2.3, Parts a, b and c and associated Notes, are presented to eliminate the

need for internal cross references. This is an administrative change with no significant adverse impact on safety.

A.6 CTS 3.10.2.6.1 and CTS 3.10.2.7.2 specify that a 2-hour deviation from AFD requirements is permissible during tests performed as part of the augmented startup program. ITS 3.2.3 does not include this exception because this allowance applied to a one time only event and is no longer required. Elimination of this allowance is an administrative change with no adverse impact on safety.

A.7 CTS 3.10.2.6.1 specifies AFD limits when thermal power is < 90% RTP and CTS 3.10.2.6.2 specifies that if these limits are not met (i.e., AFD targets exceeded for greater than 1 hour in previous 24 hours), then power must be reduced to < 50% RTP (See ITS 3.2.3, DOC L.1). No Completion Time is specified so the action must be initiated immediately.

ITS 3.2.3, Required Action C.1, maintains the requirement to reduce thermal power to < 50% RTP if AFD limits for < 90% RTP are not met (i.e., AFD targets exceeded for greater than 1 hour); however, a Completion Time of 30 minutes (versus initiate action immediately) for completion of the power reduction. This change is needed because it eliminates ambiguity about the time allowed to complete the required power reduction. This change is acceptable because this Completion Time ensures that the reactor is < 50% RTP within 60 minutes after the determination that the reactor is outside the AFD target. This ensures the plant is at a power level at which the AFD is not a significant accident analysis parameter before changes in a xenon result in additional significant changes to the power distribution. This is an administrative change with no significant adverse impact on safety because the 30 minute Completion Time for power reduction is a reasonable interpretation of the existing requirement.

A.8 CTS 3.10.2.6.1 specifies AFD limits when thermal power is < 90% RTP and CTS 3.10.2.6.2 specifies that if these limits are not met (i.e., AFD targets exceeded for greater than 1 hour in previous 24 hours), then

power must be reduced to < 50% RTP (See ITS 3.2.3, DOC L.1). No direction is provided if AFD is restored to the target band before the power reduction is completed; however, the CTS 3.10.2.6.2 is not met until AFD has exceeded the target band for less than 1 hour in previous 24 hours.

ITS 3.2.3, Required Action C.1, maintains the requirement to reduce thermal power to < 50% RTP if AFD limits for < 90% RTP are not met (i.e., AFD targets exceeded for greater than 1 hour in previous 24); however, the Note to Condition C clarifies that LCO 3.2.3 is not met until AFD has exceeded the target band for less than 1 hour in the previous 24 hours (i.e., LCO 3.2.3 is not met as soon as AFD is restored to the target band).

The addition of the note to ITS 3.2.3, Condition C, is an administrative change with no significant adverse impact on safety because these additions clarify the existing requirements.

A.9 CTS 3.10.2.6.3 specifies that a power increase to > 90% RTP is contingent upon the indicated axial flux difference being within their target band. This statement is not included in ITS LCO 3.2.3 because it is a restatement of the LCO. Deletion of CTS 3.10.2.6.3 is an administrative change with no significant adverse impact on safety because there is no change to the existing requirements.

A.10 CTS 3.11.B specifies that power shall be limited to 90% RTP if calibration requirements for the excore axial offset detection system, identified in Table 4.1-1, are not met. CTS Table 4.1-1, Item 1, requires monthly calibrations of upper and lower power range detector chambers for axial offset. ITS LCO 3.2.4 does not establish any requirements for the calibration of the excore detectors used to determine AFD. This change is needed and is acceptable because ITS LCO 3.3.1, Reactor Trip System (RTS) Instrumentation, establishes requirements, including calibration, for the Operability of the power range detectors that are at least equivalent to those imposed by CTS 3.11.B. Additionally, proper calibration of the instruments used to verify that AFD limits are met is an intrinsic requirement for

satisfactory performance of ITS SR 3.2.3.1 through ITS SR 3.2.3.4. Therefore, deletion of calibration requirements for excore detectors is an administrative change with no significant adverse impact on safety because there is no change to existing requirements, except as identified and justified in conversion package ITS 3.3.1.

MORE RESTRICTIVE

M.1 CTS 3.10.2.4 provides the allowance that AFD does not have to be maintained within the specified band during excore calibration procedures.

ITS LCO 3.2.3, Note 4, maintains this allowance by allowing operation with AFD outside the target band without accumulating penalty deviation time during power range detector calibrations performed in accordance with ITS SR 3.3.1.6. However, ITS LCO 3.2.3 limits this allowance to 16 hours during each SR interval (i.e., 92 days) and ITS LCO 3.2.3 requires that AFD be maintained within acceptable operation limits.

This change is needed because deviation in the AFD will result of doing the NIS calibration. This more restrictive change is acceptable because it does not introduce any new allowance while both limiting the duration and extent to which the plant can be operated with AFD outside the target band for performing required SRs. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.10.2.8 specifies that if the AFD monitor alarm is not functional, then verifications that AFD target band is being met must include logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

ITS SR 3.2.3.2 maintains the requirement for verification and logging AFD requirements are met; however, ITS SR 3.2.3.2 establishes the following more restrictive Frequencies: a) once within 15 minutes and every 15 minutes thereafter when ≥ 90% RTP; and, b) once within 1 hour and every 1 hour thereafter when < 90% RTP.

This change is needed because SR Frequency must be sufficient to identify a Condition and take Required Actions before a significant adverse consequence is possible. This more restrictive change is acceptable because it does not introduce any new allowance while providing greater assurance of timely identification that AFD is not within the required target band and when the AFD monitor alarm is inoperable. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.10.2 does not establish any requirement or operator verification that AFD is within required limits when the AFD monitor alarm is functional. ITS SR 3.2.3.1 requires verification that AFD is within the required target band for each Operable excore channel every 7 days.

This change is needed because it requires periodic verification that the AFD, as indicated by the NIS excore channels, is within the target band and consistent with the status of the AFD monitor alarm. The SR Frequency of 7 days is acceptable because AFD is controlled by the operator and monitored by the process computer and any deviations of the AFD from the target band that is not alarmed is expected to be readily noticed by the operators. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.2.6.2 requires that if AFD deviates from the target band for more than 1 hour in any 24-hour period when operating > 50% RTP and < 90% RTP, then power must be reduced to < 50% RTP and the high neutron flux trip setpoint must be reduced to < 55% RTP.

Under the same conditions (AFD deviates from the target band for more than 1 hour in any 24-hour period), ITS LCO 3.2.3, Required Action C.1, maintains the requirement to reduce power to <50% RTP; however, there is no requirement to reduce the high neutron flux trip setpoint.

This change is needed because requiring neutron flux setpoint reduction imposes a burden on plant operation with no commensurate safety benefit. Not requiring a proportional reduction in the power range high flux trip

setpoint when reactor power is restricted to < 50% RTP to compensate for AFD limits not met is acceptable because AFD limits are required to ensure the following: a) to ensure Heat Flux Hot Channel Factor (FQ(Z)) is not exceeded during either normal operation or in the event of xenon redistribution following power changes; and, to limit the range of power distributions that are assumed as initial conditions in analyzing events. Adequate compensation for failure to meet AFD limits is provided by steady state power reduction because AFD limits are used to establish limiting initial conditions for events and not plant response. Additionally, administrative controls are adequate to ensure power reductions specified by ITS LCO 3.2.3, Required Action C.1, are implemented and maintained. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.10.2 does not explicitly establish when AFD limits are applicable. CTS 3.10.2.7 allows unlimited operation when < 50% RTP even if AFD limits are not met; however, as specified in CTS 3.10.2.7.2, operation > 50% RTP may be limited by AFD penalties that are accumulated based any power history (i.e., > 0% RTP) during the previous 24 hours. Therefore, the implied applicability is whenever the reactor is critical.

ITS 3.2.3, Required Action C.1, maintains the allowance for unlimited operation when < 50% RTP even if AFD limits are not met; however, ITS LCO 3.2.3, Note 3, specifies that operation > 50% RTP may be limited by AFD penalties that are accumulated based on power history > 15% RTP during the previous 24 hours. Therefore, ITS LCO 3.2.3 is Applicable in Mode 1 with Thermal Power > 15% RTP even though unlimited operation < 50% RTP is permitted when AFD limits are not met.

This change is needed because accumulating AFD penalties based on operation < 15% RTP could restrict plant power ascension with no commensurate safety benefit. This change, accumulating AFD penalties based only on operation > 15% RTP, is acceptable because the xenon distribution resulting from AFD not within limits when < 15% RTP is not sufficient to affect the power distribution when > 50% RTP. Therefore, monitoring it is not necessary (i.e., accumulate AFD deviation penalty time) when operating < 15% RTP. Therefore, this change has no adverse impact on safety.

L.3 CTS 3.10.2.3 requires that the reference equilibrium AFD for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. Although an initial determination of the target flux difference following refueling is not explicitly required, target flux difference is determined any time that equilibrium xenon conditions have been established with the control rod banks withdrawn to the normal full power operating position.

ITS SR 3.2.3.4 maintains the requirement to determine, by measurement, the target flux difference of each Operable excore channel every 92 EFPDs; however, ITS SR 3.2.3.4 allows the initial measurement of target flux difference to be deferred until 31 days after refueling with the interim target flux difference after each refueling determined from design predictions.

This change is needed because full power equilibrium xenon conditions are required to perform this measurement and significant time may elapse following refueling may elapse before these conditions are established. This change eliminates ambiguity by establishing a specific allowance for the initial performance of this measurement. This change is acceptable because significant errors in core design predictions are expected to be identified during startup tests and initial low power operation. Therefore, significant operation based on erroneous design predictions will be prevented. This change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.10.2 specifies that AFD must be maintained within the band specified in the Core Operating Limits Report; however, CTS 3.10.2.6.1 includes a partial description of the band specified in the COLR when power is < 90% RTP. Specifically, when < 90% RTP, the AFD target band is equal to the target band when > 90% RTP but increased by another value specified in the COLR for each 2% of rated power below 90% RTP.

DISCUSSION OF CHANGES
ITS SECTION 3.2.3 - AXIAL FLUX DIFFERENCE (AFD)

ITS LCO 3.2.3 maintains the requirement to maintain AFD within the target band within the target flux difference specified in the COLR. CTS descriptions and modifications to the limits specified in COLR are being relocated to the COLR. This change is needed because it establishes a single location for maintaining the AFD limits.

This change is acceptable because ITS LCO 3.2.3 maintains the existing requirement to maintain AFD within the limits specified in the COLR and ITS 5.6.5, Core Operating Limits Report (COLR), includes detailed requirements that ensure AFD limits will be properly established and maintained. Requirements established by ITS 5.6.5 include the following:

a. The analytical methods used to determine the core operating limits must be those previously reviewed and approved by the NRC. The approved documents that document this approved methodology must be listed in ITS 5.6.5 and can be changed only with a TS change.

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

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications. Additionally, an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.3:
"AXIAL FLUX DIFFERENCE (AFD)"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.2.3 - AXIAL FLUX DIFFERENCE (AFD)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   This change eliminates the requirement to reduce the high neutron flux trip setpoint to < 55% RTP when AFD limits are not met (i.e., AFD deviates from the target band for more than 1 hour in any 24 hour period).

   This change will not result in a significant increase in the probability of an accident previously evaluated because violation of power distribution limits is not the initiator of any analyzed event.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because AFD limits are required to ensure the following: a) to ensure Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes; and, to limit the range of power distributions that are assumed as initial conditions in analyzing events. Adequate compensation for failure to meet AFD limits is provided by steady state power reduction because AFD limits are used to establish limiting initial conditions for events and not plant response. Additionally, administrative controls are adequate to ensure power reductions specified by ITS LCO 3.2.3, Required Action C.1, are implemented and maintained.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant

operation are consistent with the current safety analysis assumptions because AFD limits are required to ensure the following: a) to ensure Heat Flux Hot Channel Factor (FQ(Z)) is not exceeded during either normal operation or in the event of xenon redistribution following power changes; and, to limit the range of power distributions that are assumed as initial conditions in analyzing events. Adequate compensation for failure to meet AFD limits is provided by steady state power reduction because AFD limits are used to establish limiting initial conditions for events and not plant response. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because AFD limits are required to ensure the following: a) to ensure Heat Flux Hot Channel Factor (FQ(Z)) is not exceeded during either normal operation or in the event of xenon redistribution following power changes; and, to limit the range of power distributions that are assumed as initial conditions in analyzing events. Adequate compensation for failure to meet AFD limits is provided by steady state power reduction because AFD limits are used to establish limiting initial conditions for events and not plant response. Additionally, administrative controls are adequate to ensure power reductions specified by ITS LCO 3.2.3, Required Action C.1, are implemented and maintained.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.2.3 - AXIAL FLUX DIFFERENCE (AFD)

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   This change modifies the Applicability requirements for the LCO governing Axial Flux Difference to Mode 1 when > 15% RTP (versus whenever the reactor is critical). Specifically, ITS 3.2.3 maintains the CTS 3.10.2 allowance for unlimited operation if AFD limits are not met when < 50% RTP; however, ITS LCO 3.2.3 specifies that operation > 50% RTP may be limited by AFD penalties that are accumulated based on power history > 15% RTP (versus > 0% RTP in CTS) during the previous 24 hours. Therefore, ITS LCO 3.2.3 is Applicable in Mode 1 with Thermal Power > 15% RTP even though unlimited operation < 50% RTP is permitted when AFD limits are not met. This change is needed because accumulating AFD penalties based on operation < 15% RTP could restrict plant power ascension with no commensurate safety benefit.

   This change, accumulating AFD penalties based only on operation > 15% RTP, will not result in a significant increase in the probability or consequences of an accident previously evaluated because the xenon distribution resulting from AFD not within limits when < 15% RTP is not sufficient to affect the power distribution when > 50% RTP. Therefore, monitoring AFD limits (i.e., accumulate AFD deviation penalty time) is not necessary when operating < 15% RTP.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant

operation are consistent with the current safety analysis assumptions because the xenon distribution resulting from AFD not within limits when < 15% RTP is not sufficient to affect the power distribution when > 50% RTP. Therefore, monitoring AFD limits (i.e., accumulate AFD deviation penalty time) is not necessary when operating < 15% RTP. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the xenon distribution resulting from AFD not within limits when < 15% RTP is not sufficient to affect the power distribution when > 50% RTP. Therefore, monitoring AFD limits (i.e., accumulate AFD deviation penalty time) is not necessary when operating < 15% RTP.

LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   ITS SR 3.2.3.4 maintains the CTS requirement to determine, by measurement, the target flux difference of each Operable excore channel every 92 EFPDs; however, ITS SR 3.2.3.4 allows the initial measurement of target flux difference to be deferred until 31 days after refueling with the interim target flux difference after each refueling determined from design predictions.

   This change will not result in a significant increase in the probability

of an accident previously evaluated because depending on design predictions instead of measuring the target flux difference of each Operable excore channel for 31 EFPDs following refueling has no effect on the initiators of any analyzed events.

This change will not result in a significant increase in the consequences of an accident previously evaluated because significant errors in core design predictions are expected to be identified during startup tests and initial low power operation. Therefore, significant operation based on incorrect design predictions will be prevented.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because significant errors in core design predictions are expected to be identified during startup tests and initial low power operation. Therefore, significant operation based on incorrect design predictions will be prevented. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because significant errors in core design predictions are expected to be identified during startup tests and initial low power operation. Therefore, significant operation based on incorrect design predictions will be prevented.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.3:
"AXIAL FLUX DIFFERENCE (AFD)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.2.3

This ITS Specification is based on NUREG-1431 Specification No. **3.2.3A**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-011	**024 R1**	DELETE THE DETAILS ON UPDATING THE TARGET FLUX DIFFERENCE	Approved by NRC	TSTF to Rewrite	**N/A**
WOG-053	**112 R1**	DELETE CONDITION D IN TECH SPEC 3.2.3.A	Approved by NRC	Incorporated	**T.2**
WOG-075 R1	**164 R1**	AFD NOTES REARRANGED	Approved by NRC	Incorporated	**T.1**

<CTS> 3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

<3.10.2.4> a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

<3.10.2.4> <3.10.2.5.1> <3.10.2.6.2>

NOTE

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band. T.1

<3.10.2.6> <3.10.2.6.1> <3.10.2.7.2> b. May deviate outside the target band with THERMAL POWER < 90% RTP but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

Insert: 3.2-12-01

<3.10.2.6.1>

NOTE

2. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

<3.10.2.7> <3.10.2.7.1> c. May deviate outside the target band with THERMAL POWER < 50% RTP.

NOTE

3. Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band. T.1

<3.10.2.7.2> <DOC L.2>

Insert: 3.2-12-02

<DOC L.2> APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

NOTE

<3.10.2.4> <DOC H.1>

4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits. T.1

NUREG-1431 Markup Inserts
ITS SECTION 3.2.3 - AXIAL FLUX DIFFERENCE (AFD)(CAOC Methodology)

INSERT: 3.2-12-01

With Thermal Power ≥ 50% RTP, T.1

INSERT: 3.2-12-02

L.2 With Thermal Power < 50% RTP and > 15% RTP, T.1

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.10.2.5> <3.10.2.5.1> <DOC A.3>	A. THERMAL POWER ≥ 90% RTP. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes
<3.10.2.5.1> <DOC A.3>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP.	15 minutes
<3.10.2.6.2> <DOC L.1> <DOC A.6> <DOC A.7> <DOC A.8>	C. ---------NOTE--------- Required Action C.1 must be completed whenever Condition C is entered. ---------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. OR THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ---------NOTE--------- Required Action D.1 must be completed whenever Condition D is entered. ---------- Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

T.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days

<DOC M.3>

target band

(continued)

PA.1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<3.10.2.8> <DOC M.2>	SR 3.2.3.2	----------NOTE---------- Assume logged values of AFD exist during the preceding time interval. ---------- Verify AFD is within limits (target band) and log AFD for each OPERABLE excore channel.	-----NOTE------ Only required to be performed if AFD monitor alarm is inoperable ---------- (PA.1) Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP AND Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP
<3.10.2.3>	SR 3.2.3.3	Update target flux difference of each OPERABLE excore channel by: a. Determining the target flux difference in accordance with SR 3.2.3.4, or b. Using linear interpolation between the most recently measured value, and either the predicted value for the end of cycle or 0% AFD.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.2.3.4	----------NOTE---------- The initial target flux difference after each refueling may be determined from design predictions. ----------	
	Determine, by measurement, the target flux difference of each OPERABLE excore channel.	Once within 31 EFPD after each refueling AND 92 EFPD thereafter

<3.10.2.3>
<DOC L.3>

<DOC L.3>

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3A AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, CAOC, involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

190

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

Insert from Page B 3.2-30

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

PA.1

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology ~~(Refs. 1, 2, and 3)~~ entails: DB.1

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events. This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 PA.1

(continued)

BASES (continued)

LCO

Move to Page B 3.2-28

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

PA.1

1

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 4). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

Insert: B3.2-30-01

T.1

The four Notes

1

~~Part A of this~~ LCO is modified by ~~a~~ Note ~~that~~ states the conditions necessary for declaring the AFD outside of the target band. The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup.

With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

Insert from Page B 3.2-31

2 and 3

~~Parts B and C of this LCO are modified by~~ Notes ~~that~~ describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part ~~B~~ of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may

(continued)

PA.1

INSERT: B 3.2-30-01

The AFD LCO establishes the limits for how much and for how long the measured AFD may deviate from a predetermined (i.e., target) AFD. The amount that the measured AFD may deviate from the target AFD is called the "target band" which is specified in the COLR. If the measured AFD is within the "target band," then there are no restrictions on plant operations.

If the measured AFD cannot be consistently maintained within the "target band" but can be maintained within the "acceptable operation limits," then reactor power must be reduced to < 90% RTP. However, even with power reduced, the measured AFD must be maintained within the target band for 23 out of every 24 hours (i.e., the cumulative penalty deviation time cannot be exceeded); otherwise additional power reductions are required.

If the measured axial flux difference cannot be maintained within the "acceptable operation limits" or the cumulative penalty deviation time for operating outside the target band is exceeded, then reactor power must be reduced to < 50% RTP. There are no restrictions on measured AFD when reactor power is < 50% RTP; however, the measured AFD must be within the "target band" for a specified period of time (i.e., the cumulative penalty deviation time must be within a specified limit) before reactor power can be increased ≥ 50% RTP.

BASES

LCO (continued)

be operated outside of the target band but within the acceptable operation limits provided in the COLR (Note 2). This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part B of this LCO (i.e., THERMAL POWER ≥ 50% RTP ~~but < 90% RTP~~). The cumulative penalty time is the sum of penalty times from Parts B and C of this LCO. (T.1)

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. Note 3 allows the accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band and reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

Insert from Page B3.2-32

Move to Page B 3.2-30

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

Insert: B 3.2-31-01

~~Figure B 3.2.3A-1 shows a typical~~ target band and ~~typical~~ AFD acceptable operation limits.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1).

Between 15% RTP and 90% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

(continued)

INSERT: B 3.2-31-01

are specified in the COLR.

BASES

APPLICABILITY (continued)

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

Move to Page B3.2-31

Note 4 allows

For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is ~~permitted~~ for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. ~~This calibration is performed every 92 days~~.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour

(continued)

BASES

ACTIONS

C.1 (continued)

if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Actions C.1 and C.2 must be completed whenever this Condition is entered.

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the

(continued)

BASES

ACTIONS

D.1 (continued)

~~previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.~~

~~Condition D is modified by a Note that requires Action D.1 be completed whenever this Condition is entered.~~

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm ~~message immediately~~ if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% RTP. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm ~~message~~ when the cumulative penalty deviation time is > 1 hour in the previous 24 hours. provides

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

SR 3.2.3.2

With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP, the AFD is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. At power levels < 90% RTP, but > 15% RTP, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

SR 3.2.3.2 is modified by a Note that states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time. The AFD should be monitored ~~and logged~~ more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3

This Surveillance requires that the target flux difference is updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.4.

Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update. ~~because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.~~

SR 3.2.3.4

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.4 (continued)

values that account for changes in incore excore calibrations that may have occurred in the interim.

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package," January 31, 1980.

3. C. Eicheldinger to D. B. Vassallo (Chief of Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975.

1 ~~4~~. FSAR, Chapter ~~[15]~~. 7

X.1

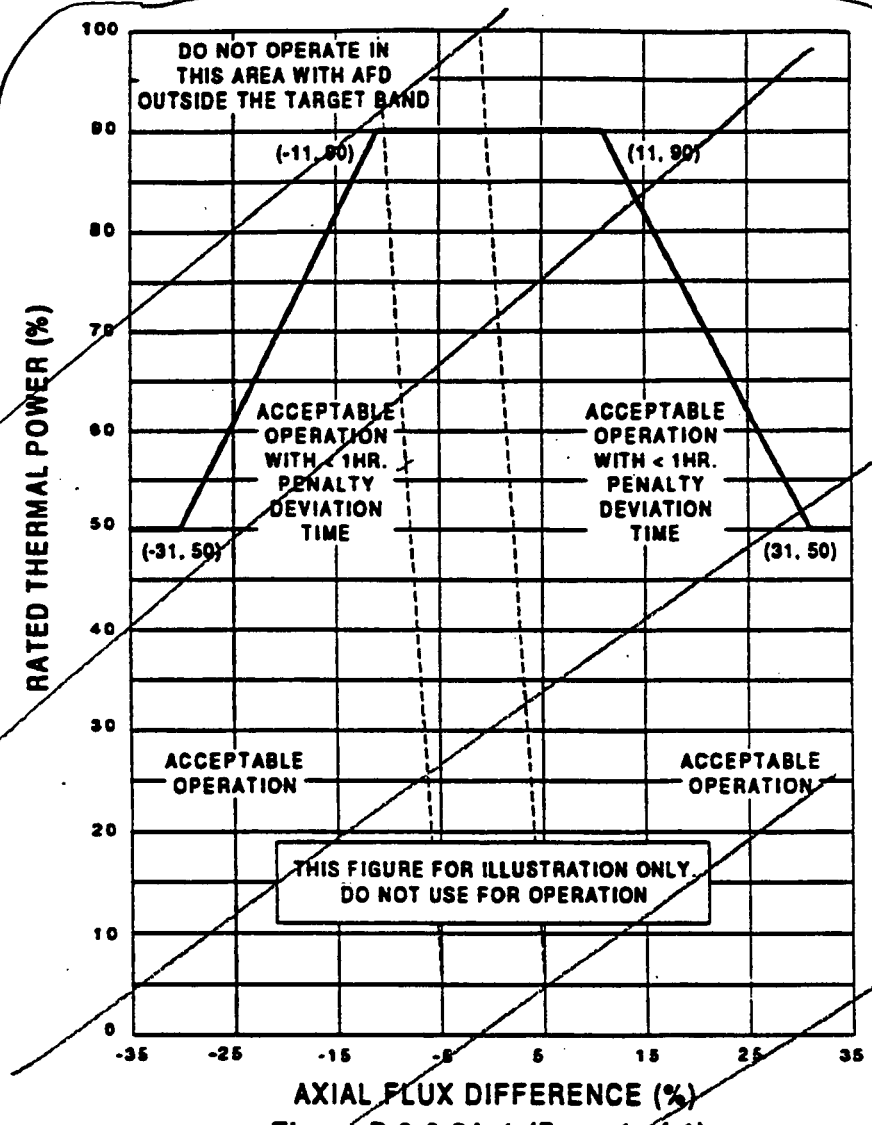

Figure B 3.2.3A-1 (Page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits and Target Band Limits as a Function of RATED THERMAL POWER

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.3:
"AXIAL FLUX DIFFERENCE (AFD)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.3 - AXIAL FLUX DIFFERENCE (AFD)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-164 (WOG-75), Rev.1, which rearranges the AFD LCO Notes and one Applicability Note into one "Notes" list in the LCO. This change improves clarity and ensure requirements are fully understood and consistently applied. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates Generic Change TSTF-112 (WOG-53), Rev.1, which deletes LCO 3.2.3, Condition D, because this Condition is not needed to ensure that power will not be increased > 50% RTP until the total commutative deviation penalty time is less than 60 minutes in the previous 24 hour period. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

X.1 ISTS 3.2.3 specifies that AFD in % flux difference units shall be maintained within the limits specified in the COLR; however, the bases for ISTS 3.2.3 includes "typical AFD limits" and associated figures. ITS deletes these references to the typical values because the IP3 Core Operating Limits Report is readily available to IP3 control room operators and other users of the Technical Specifications. Therefore, the inclusion of "typical" values and figures from the COLR provides no benefit and is a potential source of operator error.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
---------NOTE--------- Required Actions A.4, A.5 and A.6 must be completed whenever Condition A is entered. ----------	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours
	AND		
A. QPTR not within limit.	A.2	Determine QPTR and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	AND		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter
	AND		
	A.4	Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.5	--------NOTE--------- Perform Required Action A.5 only after Required Action A.4 is completed. ----------	
		Calibrate excore detectors to show 1.00 QPTR.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
	A.6	--------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. ----------	
		Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after reaching RTP OR Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ---------- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	----------NOTE---------- Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. ---------- Verify QPTR is within limit using the movable incore detectors.	24 hours

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 2);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel (Ref. 3); and

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 4).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs

BASES

APPLICABILITY (continued)

still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2

After completion of Required Action A.1, the QPTR may still exceed the specified limit. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small

BASES

ACTIONS

A.3 (continued)

changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are recalibrated to show a 1.00 QPTR prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by a Note that states that the QPT is not normalized until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

BASES

ACTIONS (continued)

A.6

Once the flux tilt is normalized (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show 1.00 tilt ratio (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show 1.00 tilt ratio and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other indications and alarms available to the operator in the control room. For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 24 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or at least two thimbles per quadrant.

The symmetric thimble flux map can be used to measure symmetric thimble "tilt." This can be compared to a reference symmetric

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map.

REFERENCES

1. 10 CFR 50.46.
2. FSAR Section 14.1.6.
3. FSAR Section 14.2.6.
4. FSAR Section 3.1.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS pages and associated TSCRs annotated for this ITS Specification :

CTS Page No.	Effective Amendment	Annotated Amendment	TSCR No.	TSCR Description	ITS Status of TSCR
1-5	97	97	No TSCRs	No TSCRs for this Page	N/A
3.10-4	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-5	112	112	No TSCRs	No TSCRs for this Page	N/A
3.10-8	181	181	No TSCRs	No TSCRs for this Page	N/A
3.10-14	103	103	No TSCRs	No TSCRs for this Page	N/A
3.10-15	112	112	No TSCRs	No TSCRs for this Page	N/A

SEE ITS 1.0

1.11 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio shall be the ratio of the maximum upper excore dector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

SR 3.2.4.1 Note 1

can and <75% RTP A.9

SEE ITS 3.0

1.12 SURVEILLANCE INTERVAL

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

1.13 OPERATION IN A DEGRADED MODE

The plant is said to be operating in a degraded mode when it is operating with one or more systems listed herein inoperable as permitted by the Technical Specifications. If inoperable components or systems are subsequently made operable, the action statements requiring plant shutdown no longer apply.

SEE ITS 1.0

1.14 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY

Noble gas $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 10 minutes, making up at least 95% of the total activity in the coolant.

1.15 DOSE EQUIVALENT I-131

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

Add SR 3.2.4.1 — M.3

3.10.2.8 — Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

SEE ITS 3.2.3

SR 3.2.4.2 and Note ~~3.10.2.9~~ — If the core is operating above 75% power with one (one or more — A.10) excore nuclear channel out of service, then core quadrant power balance shall be determined once a day (24 hours) using movable incore detectors (at least two thimbles per quadrant). — LA.2

LCO 3.2.4 ~~3.10.3~~ — Quadrant Power Tilt Limits (Mode 1 with > 50% RTP — A.3)

LCO 3.2.4 ~~3.10.3.1~~ — When ever (A.4) the indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, within two hours (A.5) the tilt condition shall be eliminated or the following actions shall be taken:

Req Act A.1

a) Restrict core power level (within 2 hours — L.1) and reset the power range high flux setpoint three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0, — L.2

and (Add Req Act A.2 — every 12 hours — M.1)

Req. Act A.2

b) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be reset to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action. — L.2

~~3.10.3.2~~ Req Act A.1 — Except for physics tests (A.7) if the indicated quadrant power tilt ration exceeds 1.09 and there is simultaneous indication of a misaligned control rod (A.8), restrict core power level 3% of rated value for every percent of indicated power tilt ratio exceeding 1.0 (L.1 — in 2 hours) and realign the rod (A.6) within two hours. If the rod is not realigned within two hours or if there is no simultaneous indication of a misaligned rod, the reactor shall be brought to the hot shutdown condition within 4 hours. If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt. — L.2

Add Required Action A.3 — M.2

Amendment No. 34, 103

Add Required Actions A.4, A.5 and A.6 — L.3

Add Condition B and associated Req Act — A.11

SEE ITS 3.1.4

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 ~~The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.~~ LA.1

3.10.4 Rod Insertion Limits

3.10.4.1 The shutdown rods shall be fully withdrawn as specified in the COLR when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin of Specification 3.10.1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits specified in the COLR.

3.10.4.3 Control bank insertion shall be further restricted if:

SEE ITS 3.1.5 and ITS 3.1.6

a) The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown.
b) A rod is inoperable (Specification 3.10.7).

3.10.4.4 Control rod insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin required by Specification 3.10.1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one control rod inserted.

SEE ITS 3.1.4

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per 8 hours and after a load change greater than 10 percent of rated power.

SEE ITS 3.1.2

3.10.10 Reactivity Balance

The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall, at least consider reactor coolant system boron concentration, control rod position, reactor coolant system average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, and samarium concentration. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core condition prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3.10.11 Notification

A8

~~Any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.~~

~~Basis~~

~~Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analysis. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable safety limit DNBR in normal operation or in short term transients.~~

~~In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant~~

A.1

A.1

described below.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level.

The two hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the moveable detector system. For a tilt condition ≤ 1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of three percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event a tilt condition of ≤ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for low power physics testing. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition.

A.1

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

A sufficient shutdown margin insures that: 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at end of life (EOL), with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident resulting in uncontrolled RCS cooldown. In the analysis of this accident, a minimum shutdown margin of 1.3 % $\Delta k/k$ is required to control the reactivity transient. Accordingly, the shutdown margin requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

The action to be taken when shutdown margin is out of limit is to borate using the best available source. In the determination of the required combination of boration flow rate and boron concentration, there is no unique Design Basis Event which must be satisfied. It is imperative to raise the boron concentration of the Reactor Coolant System as soon as possible. Therefore, the operator should begin boration with the best possible source available for the plant condition.

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

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

The specifications of Section 3.10.5 ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. Operability of the control rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits. Permitted control rod misalignments (as indicated by the analog rod position indicators within one hour after control rod motion) fall into four separate categories, which are:

a) ±18 steps of the group step counter demand position (if sufficient peaking factor margin exists and the power level is greater than 85% of rated thermal power);

b) to within +17, -12 steps ramping to ±17 steps of the group step counter demand position (depending on the group step counter demand position greater than

Amendment No. 34, 103, 112.

TSCR 97-010

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.10.3 does not identify applicability requirements for the QPTR; however, CTS 3.10.3.1.b and CTS 3.10.3.2 specify that if QPTR limits are not met, then operation is permitted up to 50% for an indefinite period while performing corrective action. This is an implied applicability because it indicates that no QPTR requirements exist below 50% RTP.

ITS LCO 3.2.4 specifies that QPTR limits are Applicable in Mode 1 with

thermal power > 50% RTP. This is an administrative change with no impact on safety because it is consistent with the Applicability implied by CTS 3.10.3.1.b and CTS 3.10.3.2. This change is acceptable because QPTR is a power distribution limit and there is insufficient energy being transferred to the reactor coolant to require limits on QPTR when the plant is ≤ 50% RTP assuming LCOs governing control rod insertion and control rod alignment are being met. Additionally, LCOs limiting $F^N_{\Delta H}$ and FQ(Z), which limit peaking factors, still apply when below 50% RTP. Therefore, this is an administrative change with no significant adverse impact on safety.

A.4 CTS 3.10.3.1 specifies that the QPTR limit is not required to be met during physics tests. ITS LCO 3.2.2 does not state this exception because the QPTR limits are applicable only in Mode 1 when thermal power is greater than 50% RTP and, as specified in ITS LCO 3.1.8, Physics Tests Exceptions, physics test exceptions are permitted in Mode 2 only. The applicability requirements for ITS LCO 3.2.4 (Mode 1 when thermal power is greater than 50% RTP) and ITS LCO 3.1.8 (Mode 2) eliminates the need for an exemption from ITS LCO 3.2.4 for physics testing. Therefore, this is an administrative change with no adverse impact on safety.

A.5 CTS 3.10.3.1 specifies that if QPTR limits are not met, then the tilt condition must be eliminated or the specified Actions started within 2 hours. ITS LCO 3.2.4, Required Action A.1, specifies the actions required within 2 hours if a tilt condition exists; however, ITS LCO 3.2.4 does not include an explicit statement that the option exists to restore compliance with the LCO within the Completion Time of the LCO Actions. An explicit statement of the option to restore compliance with the LCO is not needed in the ITS because ITS LCO 3.0.2 establishes this option for all ITS LCOs unless specifically disallowed. This is an administrative change with no significant adverse impact on safety because there is no change to the existing requirements.

A.6 CTS 3.10.3.2 specifies actions if QPTR exceeds 1.09 with a concurrent misaligned rod. These actions include a requirement to realign the

misaligned rod within 2 hours (See ITS 3.2.4, DOC L.2).

ITS LCO 3.2.4 Conditions and Required Actions do not establish any requirements associated with a misaligned control rod. ITS LCO 3.1.4, Rod Group Alignment Limits, and ITS LCO 3.1.6, Control Bank Insertion Limits, establish requirements for improperly positioned control rods.

This change is needed because the basis for rod alignment limits is maintaining sufficient scram insertion reactivity to meet SDM requirements and the bases for the QPTR limit is to limit local power peaking. Although a misaligned rod can result in exceeding a core power distribution limit, the Actions for a misaligned rod are to ensure sufficient SDM is maintained and the Actions for exceeding QPTR limits are to prevent exceeding a local power limit. When a mispositioned rod causes QPTR limits to be exceeded, then actions for both a mispositioned rod (ITS LCO 3.1.4 and 3.1.6) and exceeding the QPTR limit (ITS 3.2.4) are both required.

Eliminating the requirements to correct a mispositioned rod (potential cause of exceeding QPTR limits) from the QPTR LCO is an administrative change with no significant adverse impact on safety because ITS LCO 3.1.4 and ITS LCO 3.1.6 maintain appropriate requirements for a mispositioned rod.

A.7 CTS 3.10.3.2 specifies that Actions for QPTR exceeding 1.09 are not required to be met during "physics tests." ITS LCO 3.2.4 does not state this exception because QPTR limits are applicable only in Mode 1 with thermal power greater than 50% RTP and, as specified in ITS LCO 3.1.8, Physics Tests Exceptions, physics tests exceptions are permitted in Mode 2 only. The applicability requirements for ITS LCO 3.2.4 (Mode 1 with thermal power > 50% RTP) and ITS LCO 3.1.8 (Mode 2) eliminate the need for an exemption from ITS LCO 3.2.4 for physics testing. Therefore, this is an administrative change with no adverse impact on safety.

A.8 CTS 3.10.11 specifies that any event requiring plant shutdown on trip setpoint reduction because of Specification 3.10, Control Rod and Power Distribution Limits, shall be reported to the Nuclear Regulatory

Commission within 30 days. ITS LCO 3.2.4 does not include an explicit requirement for the submittal of a special report for any event requiring plant shutdown on trip setpoint reduction. This change is needed because requirements for reportable events are included in 10 CFR 50.72 and 10 CFR 50.73 and are not repeated in the ITS to avoid the potential for contradictions. This change is acceptable because there is no change to the existing requirements and future changes are appropriately controlled. Additionally, adequate administrative controls exist to ensure this requirement is understood and properly implemented. Therefore, this is an administrative change with no adverse impact on safety.

A.9 CTS Definition 1.11, Quadrant Power Tilt Ratio, specifies that when one excore detector is inoperable, then the remaining three (excore) detectors shall be used for computing the average; however, CTS 3.10.2.9 specifies that if one excore detector is out of service operating above 75% RTP, then core quadrant power balances shall be determined using movable incore detectors. Although IP3 complies with both CTS 1.11 and CTS 3.10.2.9, the intent is that CTS 3.10.2.9 supercedes CTS 1.11 because incore detectors provide more accurate indication of QPTR than use of 3 of the 4 excore detectors and QPTR measurements are more critical at higher power levels.

ITS SR 3.2.4.1 and ITS SR 3.2.4.2 maintain the requirement in CTS 3.10.2.9 and require use of incore detectors to determine QPTR when operating > 75% RTP with one excore detector out of service. This is an administrative change with no significant adverse impact on safety because there is no change to the existing requirements.

A.10 CTS 3.10.2.9 specifies that if one excore detector is out of service when operating above 75% RTP, then QPTR must be determined using incore detectors. ITS SR 3.2.4.1, Note 2, and ITS SR 3.2.4.2 allow QPTR to be verified using incore detectors with any number of excore detectors inoperable. This change is acceptable because incore detectors are adequate to determine QPTR accurately and provide adequate assurance that QPTR is being maintained within specified limits. This is an administrative change with no significant adverse impact on safety

because ITS LCO 3.3.1, Reactor Trip System (RTS) Instrumentation, would not allow continued operation with more than one inoperable excore detector. Therefore, this is an administrative change with no significant adverse impact on safety.

A.11 CTS 3.10.3.1.b requires a power reduction to ≤ 50% RTP if QPTR limits are not met within 24 hours for testing and evaluation (See ITS 3.2.4, DOC A.3). ITS 3.2.4, Required Action B.1, requires that reactor power be reduced to ≤ 50% RTP if Required Actions A.1 or A.2 are not completed within the specified Completion Times. This is an administrative change with no significant adverse impact on safety because ITS 3.2.4, Required Action B.1, is consistent with the intent of CTS 3.10.3 to restrict power to ≤ 50% RTP if QPTR limits are not met.

MORE RESTRICTIVE

M.1 CTS 3.10.3.1.a requires a proportional power reduction if QPTR limits are not met and CTS 3.10.3.1.b requires reducing thermal power to < 50% RTP if the tilt condition is not eliminated within 24 hours. There are no explicit requirements for interim verification of QPTR limits and/or additional power reductions during this initial 24 hour period.

ITS 3.2.4, Required Action A.1, maintains the requirement for a proportional power reduction if QPTR limits are not met; however, ITS 3.2.4, Required Action A.2, requires verification within 12 hours that QPTR limits are met and requires additional proportional power reductions if QPTR limits are not met (See ITS 3.2.4, DOCs L.1 and L.2).

This change is more restrictive because ITS LCO 3.2.4 requires that QPTR be re-verified (and appropriate action taken if QPTR is still not met) within 12 hours (and every 12 hours thereafter) instead of the one re-verification within 24 hours. This change is needed because the requirement to re-verify QPTR within 12 hours (and every 12 hours thereafter) and reduce power by 3% for each 1% that QPTR exceeds 1.0 provides a more timely response and provides greater assurance that QPTR will be restored to within limits quickly. This more restrictive change

is acceptable because it does not introduce any operation that is un-analyzed while requiring a more conservative response than is currently required for the restoration of thermal limits. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.10.3 establishes requirements for QPTR because it can be continuously monitored and limiting QPTR ensures that the margins for uncertainty for $F_Q(Z)$ (ITS LCO 3.2.1) and $F^N_{\Delta H}$ (ITS LCO 3.2.2), which cannot be continuously monitored, are not exceeded. However, CTS 3.10.3 does not include any requirements for verification that exceeding QPTR limits is not causing ITS LCO 3.2.1 ($F_Q(Z)$) and ITS LCO 3.2.2 ($F^N_{\Delta H}$) to be not met (See ITS 3.2.4, DOC L.2).

Under the same Conditions (QPTR limits not met), ITS LCO 3.2.4, Required Action A.3 requires accelerated verification that ITS LCO 3.2.1 ($F_Q(Z)$) and ITS LCO 3.2.2 ($F^N_{\Delta H}$) are being met (i.e., within 24 hours and every 7 days thereafter until QPTR limits are met versus every 31 EFPDs).

This change is needed because ITS 3.2.4, Required Actions A.1 and A.2, uses an iterative process to restore QPTR to within limits (i.e., determine QPTR within 12 hours and reduce power proportionally if QPTR limits are exceeded). Therefore, QPTR could remain outside specified limits for a considerable period of time. ITS 3.2.4, Required Action A.3, ensures that exceeding QPTR limits is not causing ITS LCO 3.2.1 ($F_Q(Z)$) and ITS LCO 3.2.2 ($F^N_{\Delta H}$) to be not met, in which case, Required Actions for ITS LCO 3.2.1 and ITS LCO 3.2.2 are applicable.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed and requires verification that the restoration from exceeding QPTR limits is not allowing ITS LCO 3.2.1 ($F_Q(Z)$) and ITS LCO 3.2.2 ($F^N_{\Delta H}$) to be not met. A Completion Time of 24 hours for the initial verification takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. A Completion Time of 7 days for periodic re-verification takes into consideration the relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.10.3 establishes requirements for QPTR; however, there are no CTS requirements for the periodic verification that QPTR limits are met because CTS 3.10.3.4 requires the operation of the tilt deviation alarm (See ITS 3.2.4, DOC LA.1) set to alarm within the QPTR LCO limit.

ITS SR 3.2.4.1 adds the requirement to verify every 7 days that QPTR is within limits. ITS SR 3.2.4.1, in conjunction with ITS SR 3.2.4.2, maintains the requirement in CTS 3.10.2.9 to use incore detectors to verify QPTR limits are met if one excore detector is inoperable when operating > 75% RTP. This change establishes the requirement that QPTR is periodically verified and is needed because ITS LCO 3.4.2 does not require Operability of the flux deviation alarm (See ITS 3.2.4, DOC LA.1). This change is acceptable because it ensures the QPTR limits are met at all times and periodically verified. The Frequency of 7 days is acceptable because the TRM maintains requirements for the flux deviation alarm and accelerated verification of QPTR if the alarm is not functional (See ITS 3.2.4, DOC LA.1) and the low probability that this alarm can remain inoperable without detection. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.10.3.1.a specifies that the power level must be reduced by three times the amount that QPTR exceeds 1.0 and that the power range high flux trip setpoint must be reduced by the same amount. No completion time is specified so these Actions must be initiated immediately.

Under the same conditions (QPTR limits not met), ITS LCO 3.2.4, Required Action A.1, allows 2 hours to complete the power reduction and does not require any adjustment to the power range high flux trip setpoint.

Allowing 2 hours to reduce power when QPTR limits are not met is acceptable because of the low probability of an event that would be exacerbated by QPTR not within required limits.

Not requiring a proportional reduction in the power range high flux trip setpoint when QPTR limits are exceeded is acceptable because the reductions in reactor power specified in ITS 3.2.4, Required Actions A.1

and A.2, are sufficient to ensure that the assumptions that are enforced by limiting QPTR are re-established. Additionally, the requirements to re-verify QPTR every 12 hours (Required Action A.2) and verify $F^N_{\Delta H}$ and $F_Q(Z)$ within 24 hours (Required Action A.3) ensure that appropriate restrictions on power level are maintained. Therefore, these changes have no significant adverse impact on safety.

L.2 CTS 3.10.3.1.b specifies that thermal power must be reduced to < 50% RTP if QPTR limits are not restored within 24 hours. Additionally, CTS 3.10.3.2 specifies actions if QPTR exceeds 1.09 and the concurrent misaligned rod cannot be aligned within 2 hours or there is no concurrent indication of a misaligned rod. In this condition, CTS requires that the reactor is in hot shutdown within 4 hours and subsequent operation is limited to 50% RTP (See ITS 3.2.4, DOCs A.6 and M.2).

Under the same conditions (QPTR exceeds 1.09 and/or not restored within 24 hours), ITS 3.2.4, Required Actions A.1 and A.2, uses an iterative process to restore QPTR to within limits (i.e., reduce power 3% for each 1% QPTR exceeds limits within 2 hours and then determine QPTR within 12 hours and reduce power proportionally if QPTR limits are still exceeded). Therefore, QPTR could remain outside specified limits for a considerable period of time and plant operation > 50% RTP could continue.

This change is acceptable because ITS 3.2.4, Required Action A.1, (i.e., a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 within 2 hours) conservatively reduces total core power by significantly more than the peak local power exceeds its limit. This provides a high degree of assurance that Required Action A.1 will restore QPTR to within required limits. Additionally, ITS 3.2.4, Required Action A.2, requires verification within 12 hours that QPTR limits are met and requires additional proportional power reductions if QPTR limits are still not met. Therefore, this change has no significant adverse impact on safety.

L.3 CTS 3.10.3.1.b specifies that power level must be reduced to < 50% RTP if QPTR limits cannot be re-established within 24 hours and operation above 50% RTP is prohibited until the QPTR limits are re-established. Under the same conditions, ITS 3.2.4, Required Actions A.4, A.5 and A.6, provide specific steps for performing an evaluation of the QPTR limit and accomplishing a carefully controlled return to 100% RTP. This change is acceptable because when the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing thermal power to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses. Additionally, Required Actions A.5 and A.6 ensure that the excore detectors are re-calibrated to detect any subsequent significant changes in QPTR and follow-up re-verifications that the core power distribution at RTP is consistent with the safety analysis assumptions. This change is acceptable because the combination of the re-evaluation of QPTR and the subsequent verifications of $F^N_{\Delta H}$ and $F_Q(Z)$ ensure that the return to power does not result in violations of any thermal limits for an extended period of time.

REMOVED DETAIL

LA.1 CTS 3.10.3.4 specifies requirements for the tilt deviation alarm including the required setpoint and the compensatory action (i.e., increased monitoring by operations personnel) when either or both of the alarms are not operable. ITS LCO 3.2.4 (ISTS 3.2.4 as modified by TSTF-110 (WOG-49), Revision 1) does not establish any requirements for the tilt deviation alarm and the details contained in CTS 3.10.3.4 are relocated to the Technical Requirements Manual (TRM).

This change is needed because the tilt deviation alarm is not an essential element in ensuring the QPTR limits are met. Specifically,

QPTR can be maintained by monitoring excore and/or incore flux detectors.

This change is acceptable because ITS LCO 3.2.4 maintains the requirement that QPTR be maintained within required limits whenever thermal power is > 50% RTP. Moving requirements for the tilt alarm to the TRM is acceptable because a prompt change in quadrant power tilt (e.g., from a dropped rod) result in other indications of abnormality and is not usually identified by the tilt deviation monitor. Additionally, the 7 day frequency for ITS SR 3.2.4.1, the determination of QPTR, is adequate to detect any relatively slow changes in QPTR. Furthermore, requirements for the tilt deviation alarm will be maintained in the TRM which will require more frequent verification of QPTR if the deviation monitor is not functional.

The Quality Assurance Plan will be revised to specify that requirements in the TRM are part of the facility as described in the FSAR and that changes to the TRM can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the TRM and future changes to the TRM will be controlled in accordance with 10 CFR 50.59.

LA.2 CTS 3.10.2.9 specifies that if one excore detector is out of service when operating above 75% RTP, then core quadrant power balances shall be determined using movable incore detectors using "at least two thimbles per quadrant." ITS SR 3.2.4.2 maintains the requirement in CTS 3.10.2.9 for the use of incore detectors to determine QPTR when operating > 75% RTP with an excore detector out of service. However, the stipulation that "at least two thimbles per quadrant" are needed to satisfy this requirement is relocated to the Bases for ITS SR 3.2.4.2.

This change is acceptable because ITS SR 3.2.4.2 maintains the requirement to use incore detectors when an excore detector is inoperable.

Maintaining the stipulation that "at least two thimbles per quadrant" are needed to satisfy this requirement in the ITS Bases is acceptable

because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 4:

No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   CTS 3.10.3.1.a specifies that power level must be reduced by three times the amount that QPTR exceeds 1.0 and that the power range high flux trip setpoint must be reduced by the same amount. No completion time is specified so these Actions must be initiated immediately.

   Under the same conditions (QPTR limits not met), ITS LCO 3.2.4, Required Action A.1, allows 2 hours to complete the power reduction and does not require any adjustment to the power range high flux trip setpoint

   This change will not result in a significant increase in the probability of an accident previously evaluated because QPTR is an operating restriction that is an initial condition of a design basis accident or transient analysis and is not assumed as the initiator of any accident previously evaluated.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because of the low probability of an event that would be exacerbated by QPTR not within required limits. Not requiring a proportional reduction in the power range high flux trip setpoint when QPTR limits are exceeded is acceptable because the reductions in reactor power specified in ITS 3.2.4, Required Actions A.1 and A.2, are sufficient to ensure that the assumptions that are enforced by limiting QPTR are re-established. Additionally, the requirements to re-verify QPTR every 12 hours (Required Action A.2) and verify $F^N_{\Delta H}$ and $F_Q(Z)$ within 24 hours (Required

Action A.3) ensure that appropriate restrictions on power level are maintained. Additionally, there is a low probability of an event during the 2 hour period allowed to complete the required power reduction.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the reductions in reactor power specified in ITS 3.2.4, Required Actions A.1 and A.2, are sufficient to ensure that the assumptions that are enforced by limiting QPTR are re-established. Additionally, the requirements to re-verify QPTR every 12 hours (Required Action A.2) and verify $F^N_{\Delta H}$ and $F_Q(Z)$ within 24 hours (Required Action A.3) ensure that appropriate restrictions on power level are maintained.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

CTS 3.10.3.1.b specifies that thermal power must be reduced to < 50% RTP if QPTR limits are not restored within 24 hours. CTS 3.10.3.2 specifies actions if QPTR exceeds 1.09 and the concurrent misaligned rod cannot be aligned within 2 hours or there is no concurrent indication of a misaligned rod. In this condition, CTS requires the reactor is in hot shutdown within 4 hours and subsequent operation is limited to 50% RTP.

Under the same conditions (QPTR exceeds 1.09 and/or not restored within 24 hours), ITS 3.2.4, Required Actions A.1 and A.2, uses an iterative process to restore QPTR to within limits (i.e., reduce power 3% for each 1% QPTR exceeds limits within 2 hours and then determine QPTR within 12 hours and reduce power proportionally if QPTR limits are still exceeded). Therefore, QPTR could remain outside specified limits for a considerable period of time and plant operation > 50% RTP could continue.

This change will not result in a significant increase in the probability of an accident previously evaluated because QPTR is an operating restriction that is an initial condition of a design basis accident or transient analysis and is not assumed as the initiator of any accident previously evaluated.

This change will not result in a significant increase in the consequences of an accident previously evaluated because ITS 3.2.4, Required Action A.1, conservatively reduces total core power by significantly more than the peak local power exceeds its limit. This provides a high degree of assurance that Required Action A.1 will restore QPTR to within required limits. Additionally, ITS 3.2.4, Required Action A.2, requires verification within 12 hours that QPTR limits are met and requires additional proportional power reductions if QPTR limits are still not met.

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

The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions.

Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because ITS 3.2.4, Required Action A.1, conservatively reduces total core power by significantly more than the peak local power exceeds its limit. This provides a high degree of assurance that Required Action A.1 will restore QPTR to within required limits. Additionally, ITS 3.2.4, Required Action A.2, requires verification within 12 hours that QPTR limits are met and requires additional proportional power reductions if QPTR limits are still not met.

LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

This change provides specific steps for performing an evaluation of the QPTR limit and accomplishing a carefully controlled return to 100% RTP following a condition where QPTR limits are not met. This change will not result in a significant increase in the probability of an accident previously evaluated because QPTR is an operating restriction that is an initial condition of a design basis accident or transient analysis and is not assumed as the initiator of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because when the QPTR exceeds its limit, it does not necessarily mean a safety concern exists.

It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing thermal power above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses. Additionally, Required Actions A.5 and A.6 ensure that the excore detectors are re-calibrated to detect any subsequent significant changes in QPTR and follow-up re-verifications that the core power distribution at RTP is consistent with the safety analysis assumptions. This change is acceptable because the combination of the re-evaluation of QPTR and the subsequent verifications of $F^N_{\Delta H}$ and $F_Q(Z)$ ensure that the return to power do not result in violations of any thermal limits for an extended period of time.

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

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the re-evaluation required before increasing thermal power above the limit of Required Action A.1 ensures that the reactor core conditions are consistent with the assumptions in the safety analyses. Additionally, Required Actions A.5 and A.6 ensure that the excore detectors are re-calibrated to detect any subsequent significant changes in QPTR and follow-up re-verifications ensure that the core power distribution at RTP is consistent with the safety analysis assumptions. This change is acceptable because the combination of the re-evaluation of QPTR and the subsequent verifications of $F^N_{\Delta H}$ and $F_Q(Z)$

ensure that the return to power do not result in violations of any thermal limits for an extended period of time.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

Status of NUREG 1431 Generic Changes for ITS 3.2.4

This ITS Specification is based on NUREG-1431 Specification No. **3.2.4**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-012	**025 R0**	REVISE ACTIONS TERMINOLOGY REGARDING QPTR TO MATCH ACTIONS BEING TAKEN	Rejected by NRC	Not Incorporated	**N/A**
WOG-045	**109 R0**	CLARIFY THE QPTR SURVEILLANCES	Approved by NRC	Incorporated	**T.3**
WOG-049 R2	**110 R2**	DELETE SR FREQUENCIES BASED ON INOPERABLE ALARMS	Approved by NRC	Incorporated	**T.2**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	Approved by NRC	Incorporated	**T.1**
WOG-095, R2		ALLOW TIME FOR STABILIZATION AFTER REDUCING POWER DUE TO QPTR OUT OF LIMIT	TSTF Review	Not Incorporated	**N/A**
WOG-101		CLARIFY COMPLETION TIME AND FREQUENCY WORDING	TSTF Review	Not Incorporated	**N/A**

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WOG-105	REQUIRE STATIC AND TRANSIENT FQ MEASUREMENT	TSTF Review	Not Incorporated	**N/A**

⟨CTS⟩ 3.2 POWER DISTRIBUTION LIMITS

⟨3.10.3⟩ 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

⟨3.10.3.1⟩ LCO 3.2.4 The QPTR shall be ≤ 1.02.

⟨3.10.3.1⟩ APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

Insert: 3.2-18-01

PA.1

⟨DOC A.3⟩
⟨DOC A.4⟩

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.10.3.1⟩ ⟨3.10.3.1.a⟩ ⟨3.10.3.2⟩ ⟨DOC L.1⟩ A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours
	AND	
⟨3.10.3.1.a⟩ ⟨DOC L.1⟩ ⟨DOC M.1⟩	A.2 Determine QPTR. ~~Perform SR 3.2.4.1~~ and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	AND	
⟨DOC M.2⟩	A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter
	AND	
⟨DOC L.3⟩	A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	

T.3

(continued)

Typical

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

PA.1

INSERT: 3.2-18-01

A. ---------NOTE---------
Required Actions A.4, A.5 and A.6 must be completed whenever Condition A is entered.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 --------NOTE--------- Perform Required Action A.5 only after Required Action A.4 is completed. ----------	
<DOC L.3>	Calibrate excore detectors to show zero QPTR. 1.00	Prior to increasing THERMAL POWER above the limit of Required Action A.1 (PA.1)
	AND	
	A.6 --------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. ----------	
<DOC L.3>	Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after reaching RTP OR Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
<DOC A.11> <3.10.3.1.b> <3.10.3.2> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<DOC M.3> <DOC LA2> <1.11> <DOC A.9> SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance ~~if adequate Power Range Neutron Flux channel inputs are not OPERABLE.~~ T.3 ---------- Verify QPTR is within limit by calculation.	7 days ~~AND~~ ~~Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable~~ T.2 CLB.1
<3.10.2.9> <DOC A.10> SR 3.2.4.2	Not ----------NOTE---------- ~~Only~~ required to be performed ~~if~~ until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. T.3 ---------- Verify QPTR is within limit using the movable incore detectors.	~~Once within~~ 12 hours T.3 CLB.1 ~~AND~~ ~~12 hours thereafter~~ 24 T.3

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made ~~during startup testing,~~ after refueling, and periodically during power operation.

Bank

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control ~~Rod~~ Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition; (Ref. 2)

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

Insert: B3.2-43-01

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

Typical

INSERT: B 3.2-43-01

225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel

BASES

APPLICABLE SAFETY ANALYSES (continued)

Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

(continued)

BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2

After completion of Required Action A.1, the QPTR ~~alarm~~ may still ~~be in its alarmed state~~ exceed the specified limit. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow. PA.1 T.2

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are recalibrated to show a zero QPTR prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR.

1.00

PA.1

Required Action A.5 is modified by a Note that states that the QPT is not zeroed out until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

normalized

PA.1

A.6

normalized

Once the flux tilt is zeroed out (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the

(continued)

BASES

ACTIONS

A.6 (continued)

core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

1.00

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

ratio

1.00 tilt ratio

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 ~~if more than one input from Power Range Neutron Flux channels are inoperable~~ T.3

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

Insert: B 3.2-48-01

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

T.2

SR 3.2.4.2 Until 24 hours after

This Surveillance is modified by a Note, which states that it is required only when the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

not

T.3

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.

at least two thimbles per quadrant

LA.2

measure

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full

(continued)

INSERT: B 3.2-48-01

T.2

Takes into account other indications and alarms available to the operator in the control room.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

incore

core flux map, to generate an incore QPTR. Therefore, QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. ~~Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.~~

REFERENCES

1. 10 CFR 50.46.
2. ~~Regulatory Guide 1.77, Rev [0], May 1974.~~
3. ~~10 CFR 50, Appendix A, GDC 26.~~

Insert:
B 3.2-49-01

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: B 3.2-49-01

2. FSAR Section 14.1.6.

3. FSAR Section 14.2.6.

4. FSAR Section 3.1.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 IP3 ITS differs from NUREG-1431 regarding the Frequency for performing QPTR determinations with incore detectors when an excore detector is inoperable. CTS 3.10.2.9 specifies that if one excore detector is out of service when > 75% RTP, then QPTR must be determined once a day (i.e., 24 hours) using incore detectors. Under the same conditions (one excore detector inoperable), NUREG-1431, Rev.1, SR 3.2.4.2 requires verification of QPTR using the incore detectors every 12 hours. (Note that SR 3.2.4.2 applies to one or more excore detectors inoperable while the CTS is limited to one excore detector inoperable. This difference is not significant because ITS LCO 3.3.1, Reactor Trip System (RTS) Instrumentation, would not allow continued operation with more than one inoperable excore detector.

Maintaining the current licensing basis is needed because IP3 experience indicates that using the incore probes every 12 hours creates a significant burden on plant operators and equipment without a commensurate increase in plant safety. Maintaining the CLB is acceptable because this option can only be used when 3 of the 4 excore detectors functional and the tilt deviation monitor is providing continuous indication of QPTR. Additionally, operation with an inoperable excore detector is expected to occur infrequently and then for only a limited period of time. Therefore, maintaining the CLB does not have a significant adverse impact on safety.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-136 (WOG-59), Rev.1, which combines ITS 3.1.1,Shutdown Margin (SDM) – T_{avg} > 200°F, and ITS 3.1.2, Shutdown Margin (SDM) – T_{avg} ≤ 200°F, into ITS 3.1.1, Shutdown Margin (SDM). This change is necessary because ITS 3.1.1 and ITS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin during physics tests to COLR.

T.2 This change incorporates Generic Change TSTF-110, Rev.1 (WOG-49), which deletes the Frequency for SR 3.2.3.1 which requires verification of the Axial Flux Difference once within 1 hour and every 1 hour thereafter when the AFD monitor alarm is inoperable. These actions are relocated from the Technical Specifications to plant administrative practices because the alarms do not directly relate to the LCO limits (i.e., ITS LCO 3.2.3 still requires that AFD limits be maintained even if the alarm is not operable).

T.3 This change incorporates Generic Change TSTF-109, Rev.1 (WOG-45), which clarifies the requirements for performing SR 3.2.4.2 for calculating QPTR using the incore detectors. This change is needed because Required Action A.2 is intended to result in a periodic re-check and re-adjustment of thermal power based on QPTR. However, Required Action A.2 specifically requires performance of SR 3.2.4.1 which may not be

possible if Power Range Neutron Flux channel(s) are inoperable. In this event, SR 3.2.4.2 should be performed using the incore detectors. To more correctly specify the intended Required Action, A.2 is revised to simply require "Determine QPTR" rather than specifying an SR to perform. Additionally, Note 2 to SR 3.2.4.1 (QPTR by calculation) allows performance of SR 3.2.4.2 (QPTR using incore detectors) "if adequate Power Range Neutron Flux channel inputs are not OPERABLE." Besides posing some ambiguity as to what "adequate...inputs" are, it is overly restrictive. QPTR determination using incore detectors can adequately verify the requirements for QPTR in all cases; not just when flux channels are inoperable. SR 3.2.4.2 presentation of the frequency for verifying QPTR using incore detectors is revised to be consistent with typical presentation formats that provide for a period of time after establishing conditions.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None